Alex L. Javorik
Columbia Generating Station
P.O. Box 968, PE04
Richland, WA 99352-0968
Ph. 509-377-8555 | F. 509-377-4150
aljavorik@energy-northwest.com

August 17, 2017
GO2-17-147

EA-12-049

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 ENERGY NORTHWEST'S NOTIFICATION OF FULL COMPLIANCE WITH ORDER EA-12-049, "ORDER MODIFYING LICENSES WITH REGARD TO REQUIREMENTS FOR MITIGATION STRATEGIES FOR BEYOND DESIGN BASIS EXTERNAL EVENTS"**

References:
1. Letter from E. J. Leeds (NRC) and M. R. Johnson (NRC) to All Power Reactor Licensees and Holders of Construction Permits in Active or Deferred Status, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ML12054A736)
2. Letter from E. J. Leeds (NRC) to All Operating Boiling Water Reactor Licensees with Mark I and Mark II Containments, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," dated June 6, 2012 (ADAMS ML13143A334 (Pkg.))
3. Letter GO2-15-124 from D. A. Swank (Energy Northwest) to the NRC, "Energy Northwest's Fifth Six-Month Status Update Report for the Implementation of NRC Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated August 25, 2015 (ADAMS ML15244B066)
4. Letter GO2-15-120 from W. G. Hettel (Energy Northwest) to the NRC, "Completion of Required Action by Nuclear Regulatory Commission (NRC) Order EA-12-051 Reliable Spent Fuel Pool Instrumentation," dated August 12, 2015 (ADAMS ML15245A530)
5. Letter from S. Monarque (NRC) to M. E. Reddemann (Energy Northwest), "Report for the Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051," dated June 16, 2015 (ADAMS ML15139A462)

6. Letter from C. F. Lyon (NRC) to M. E. Reddemann (Energy Northwest), "Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan for Implementation of Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," dated November 7, 2013 (ML13302C136)

Dear Sir or Madam,

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued Order EA-12-049 (Reference 1) which directed Energy Northwest to implement mitigation strategies for beyond-design-basis external events at the Columbia Generating Station (Columbia). On June 6, 2013, the NRC issued Order EA-13-109 (Reference 2) which directed Energy Northwest to install a reliable hardened containment vent capable of operation under severe accident conditions.

In Reference 3, Energy Northwest reported completion of those items associated with NRC Order EA-12-049 required to be completed prior to the restart from Refueling Outage 22. In Reference 4, Energy Northwest reported compliance with the requirements of NRC Order EA-12-051, Reliable Spent Fuel Pool Instrumentation.

The purpose of this letter is to report full compliance with NRC Order EA-12-049 as required by Section IV, Condition C.3 of Reference 1 and completion of the Phase 1 requirements of NRC Order EA-13-109. This letter also provides the required responses to the items that were identified as open or pending in June 16, 2015 audit report (Reference 5). Additionally, Energy Northwest is including the questions and answers related to Order EA-12-051 as requested by the NRC.

Attachment 1 to this letter documents the completion of those elements required to confirm acceptable implementation of NRC Order EA-12-049. The Columbia Final Integrated Plan (FIP) complies with NEI 12-06, Revision 2, with the exception of Appendix E. Appendix E, Validation Guidance, was finalized after Energy Northwest completed the validation of the Time-Critical mitigation actions. Other aspects of NEI 12-06, Revision 2, while not applicable to this Order compliance, will be utilized for upcoming submittals (e.g., mitigating strategies assessment for the reevaluated hazards using Appendix G and Appendix H) and rulemaking (e.g., references to NEI 13-06 and NEI 14-01).

Attachment 2 contains the required responses to the items identified as open or pending from the June 16, 2015 audit report. Attachment 3 contains the additional information requested in Reference 6 on spent fuel pool level instrumentation. Attachment 4 provides the Columbia FIP.

No new commitments are identified in this letter.

If you have any questions or require additional information, please contact Ms. L. L. Williams at (509) 377-8148.

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

Executed on the 17th day of August, 2017

Respectfully,

A. L. Javorik
Vice President, Engineering

Attachments As stated

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399 (email)
WA Horin – Winston & Strawn

COLUMBIA GENERATING STATION, DOCKET NO. 50-397

Energy Northwest's Compliance with the March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order EA-12-049) for the Columbia Generating Station

Introduction

In Reference 2, Energy Northwest previously reported completion of those items associated with NRC Order EA-12-049 required to be completed prior to the restart from Refueling Outage 22. During Refueling Outage 23, Energy Northwest completed Phase 1 of NRC Order EA-13-109, the installation of a reliable hardened containment vent capable of operation under severe accident conditions. Energy Northwest is now reporting full compliance with NRC Order EA-12-049 as documented in this attachment.

Milestone Schedule – Items Complete

Milestones	Target Commence Date	Target Completion Date	Activity Status
Correspondence & Reports:			
Submit 60 Day Initial Mitigation Strategies Status Report	Oct. 2012	Oct. 2012	Completed GO2-12-149
Submit Mitigation Strategies Overall Integrated Plan	Feb. 2013	Feb. 2013	Completed GO2-13-034
First Status Update Report for the Mitigation Strategies Overall Integrated Plan	Aug. 2013	Aug. 2013	Completed GO2-13-123
Second Status Update Report for the Mitigation Strategies Overall Integrated Plan	Feb. 2014	Feb. 2014	Completed GO2-14-031
Third Status Update Report for the Mitigation Strategies Overall Integrated	Aug. 2014	Aug. 2014	Completed GO2-14-131
Fourth Status Update Report for the Mitigation Strategies Overall Integrated Plan	Feb. 2015	Feb. 2015	Completed GO2-15-034
Fifth Status Update Report for the Mitigation Strategies Overall Integrated Plan	Aug. 2015	Aug. 2015	Completed GO2-15-124
Sixth Status Update Report for the Mitigation Strategies Overall Integrated Plan	Feb. 2016	Feb. 2016	Completed GO2-16-037
Seventh Status Update Report for the Mitigation Strategies Overall Integrated Plan	Aug. 2016	Aug. 2016	Completed GO2-16-125
Eighth Status Update Report for the Mitigation Strategies Overall Integrated Plan	Feb. 2017	Feb. 2017	This was submitted early in conjunction with the December 2016 HCV Update GO2-16-171
Issuance of Energy Northwest letter of compliance with NRC Order EA-12-049, Section IV.C.3	Jun. 2017	Aug. 2017	This Letter

Milestones	Target Commence Date	Target Completion Date	Activity Status
Issuance of Columbia's Final Integrated Plan	NA	NA	This Letter
Evaluations for Mitigation Strategies Phase 1, 2 & 3			
Perform Engineering Evaluations	Jun. 2013	Apr. 2015	Completed GO2-15-124
Engineering & Modifications for Mitigation Strategies Phase 1, 2 & 3			
Develop Engineering Design for Modifications	Jun. 2013	Apr. 2015	Completed GO2-15-124
Plant Modification Installation	Apr. 2014	Jun. 2015	Completed GO2-15-124
Diverse and Flexible Coping Strategies (FLEX) Support Guidelines (FSG) Program & Procedures:			
Perform FLEX procedure tabletop exercise	Dec. 2014	Apr. 2015	Completed GO2-15-124
Develop FSGs	Jul. 2013	Apr. 2015	Completed GO2-15-124
Develop testing, calibration, maintenance and surveillance procedures for portable FLEX equipment	Jan. 2014	Apr. 2015	Completed GO2-15-124
FLEX Program Procedural Changes are placed in effect	Jun. 2015	Jun. 2015	Completed GO2-15-124
Procurement & Storage Plan:			
Complete modification and installation of FLEX buildings	Oct. 2013	Jun. 2014	Completed GO2-15-034
Procure and store necessary FLEX portable equipment	Jun. 2013	Apr. 2015	Completed GO2-15-124
Test portable FLEX equipment	Mar. 2014	Apr. 2015	Completed GO2-15-124
Establish programmatic controls for portable FLEX equipment	Jan. 2014	Apr. 2015	Completed GO2-15-124
Mitigation Strategies Staffing Analysis:			
Perform Mitigation Strategies Staffing Analysis	Aug. 2014	Dec. 2014	Completed GO2-14-174 GO2-15-124
Operations & Training:			
Development of Mitigation Strategies Program training modules	Jan. 2015	Mar. 2015	Completed GO2-15-124
Mitigation Strategies Program training of station personnel	Mar. 2015	Jun. 2015	Completed GO2-15-124

Milestones	Target Commence Date	Target Completion Date	Activity Status
Operational/Functional Testing of Mitigation Strategies Program Structures, Systems, Components (SSC)	Mar. 2015	Jun. 2015	Completed GO2-15-124
Final Mitigation Strategies Program turned over to Operations	Jun. 2015	Jun. 2015	Completed GO2-15-124

HCV Phase 1 Milestone Schedule:

Milestone	**Target Completion Date**	**Activity Status**	**Comments** *(Include date changes in this column)*
Hold preliminary/conceptual design meeting	June 2014	Complete	GO2-15-175
Design Engineering Complete	May 2016	Complete	This Letter
Operation Procedure Changes Developed	Mar 2017	Complete	This Letter
Training Complete	Apr. 2017	Complete	This Letter
Installation Complete	May 2017	Complete	This Letter
Procedure Changes Active	May 2017	Complete	This Letter
Site Specific Maintenance Tasks Developed	June 2017	Complete	This Letter
Walk Through Demonstration/Functional Test	June 2017	Complete	This Letter

Order EA-12-049 Compliance Elements Summary

The elements identified below for Columbia, as well as the Overall Integrated Plan submitted in Reference 3 and revised in Reference 4, the 6-Month Status Reports (References 4 through 11), the final integrated plan (FIP) (Attachment 4) and any additional docketed correspondence, demonstrate compliance with Order EA-12-049.

The Columbia FIP is based on NEI 12-06, Revision 2, with the exception of Appendix E, which was finalized after the validation process was completed. Other aspects of NEI 12-06, Revision 2, while not applicable to compliance with this Order, will be utilized for upcoming submittals (e.g., use of reevaluated hazards, Appendix G and Appendix H) and rulemaking (e.g., references to NEI 13-06 and NEI 14-01).

Strategies - Complete

The Columbia FLEX strategies are in compliance with Order EA-12-049. There are no strategy related Open Items, Confirmatory Items, or Audit Questions/Audit Report Open Items other than those that are pending NRC review as shown in Attachment 2.

Modifications - Complete

The modifications required to support the FLEX strategies for Columbia have been fully implemented in accordance with the station design control process.

Equipment – Procuredment and Maintenance & Testing - Complete

The equipment required to implement the FLEX strategies for Columbia has been procured in accordance with NEI 12-06, Section 11.1 and 11.2, received at Columbia, initially tested/performance verified as identified in NEI 12-06, Section 11.5, and is available for use.

Periodic maintenance and testing will be conducted through the use of the Columbia preventative maintenance program.

Protected Storage - Complete

The storage facilities required to implement the FLEX strategies for Columbia have been completed and provide protection from the applicable site hazards. The equipment required to implement the FLEX strategies for Columbia is stored in its protected configuration.

Procedures - Complete

FLEX Support Guidelines (FSGs) for Columbia have been developed and integrated with existing procedures. The FSGs, and affected existing procedures, have been verified in accordance with the site procedures and are available for use.

Training - Complete

Training for Columbia has been completed in accordance with an accepted training process as recommended in NEI 12-06, Section 11.6.

Staffing - Complete

The staffing study for Columbia has been completed in accordance with 10 CFR 50.54(f), "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," Recommendation 9.3, dated March 12, 2012 (Reference 11), as documented in letter dated December 23, 2014 (Reference 13) and reviewed by the NRC staff in Reference 14.

National Safer Response Centers - Complete

Energy Northwest has established a contract with Pooled Equipment Inventory Company (PEICo) and has joined the Strategic Alliance for FLEX Emergency Response (SAFER) Team Equipment Committee for off-site facility coordination. It has been confirmed that PEICo is ready to support Columbia with Phase 3 equipment stored in the National SAFER Response Centers in accordance with the site specific SAFER Response Plan.

Validation - Complete

Energy Northwest has completed performance of validation in accordance with industry developed guidance to assure required tasks, manual actions and decisions for FLEX strategies are feasible and may be executed within the constraints identified in the Overall Integrated Plan (OIP) / FIP for Order EA-12-049. The program document, FLEX-01, contains documentation of the completed validation.

FLEX Program Document - Established

The Energy Northwest, Columbia FLEX Program Document, FLEX-01, has been developed in accordance with the requirements of NEI 12-06.

References

1. Letter GO2-12-149 from D. A. Swank (Energy Northwest) to the NRC, "Energy Northwest's Initial Status report in response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated October 25, 2012 (ADAMS ML12310A385)
2. Letter GO2-15-124 from D. A. Swank (Energy Northwest) to the NRC, "Energy Northwest's Fifth Six-Month Status Update Report for the Implementation of Nuclear Regulatory Commission (NRC) Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated August 25, 2015 (ADAMS ML15244B066)
3. Letter GO2-13-034 from A. L. Javorik (Energy Northwest) to NRC, "Energy Northwest's Response to NRC Order EA-12-049 – Overall Integrated Plan for Mitigating Strategies," dated February 28, 2013 (ADAMS ML13071A614)
4. Letter GO2-14-031, from D. A. Swank (Energy Northwest) to the NRC, "Energy Northwest's Second Six Month Status Update Report for the Implementation of NRC Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated February 27, 2014 (ADAMS – Not Available)
5. Letter GO2-13-123 from D.A. Swank (Energy Northwest) to NRC, "Energy Northwest's First Six Month Status Update Report for the Implementation of NRC Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated August 28, 2013 (ADAMS – Not Available)

6. Letter GO2-14-131 from D. A. Swank (Energy Northwest) to the NRC, "Energy Northwest's Third Six-Month Status Update Report for the Implementation of NRC Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated August 28, 2014 (ADAMS ML14254A403)
7. Letter GO2-15-034 from D. A. Swank (Energy Northwest) to the NRC, "Energy Northwest's Fourth Six-Month Status Update Report for the Implementation of NRC Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated March 2, 2015 (ADAMS ML15083A086)
8. Letter GO2-16-037, from A. L. Javorik (Energy Northwest) to the NRC, "Energy Northwest's Sixth Six-Month Status Update Report for the Implementation of Nuclear Regulatory Commission (NRC) Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated February 24, 2016 (ADAMS ML16055A271)
9. Letter GO2-16-125, from A. L. Javorik (Energy Northwest) to the NRC, "Energy Northwest's Seventh Six-Month Status Update Report for the Implementation of Nuclear Regulatory Commission (NRC) Order EA-12-049 Mitigation Strategies for Beyond Design Basis External Events," dated August 30, 2016 (ADAMS ML16243A471)
10. Letter GO2-16-171, from A. L. Javorik (Energy Northwest) to NRC, "Energy Northwest's Combined Six-Month Status update Report for the Implementation of Nuclear Regulatory Commission (NRC) Orders EA-12-049 and EA-13-109," dated December 29, 2016 (ADAMS ML16364A245)
11. Letter GO2-17-118, from A. L. Javorik (Energy Northwest) to NRC, "Energy Northwest's Second Combined Six-Month Status update Report for the Implementation of Nuclear Regulatory Commission (NRC) Orders EA-12-049 and EA-13-109," dated June 27, 2017 (ADAMS ML17178276)
12. Letter from E. J. Leeds and M. R. Johnson (NRC) to All Power Reactor Licensees, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS ML12053A340)
13. Letter GO2-14-174, from W. G. Hettel (Energy Northwest) to the NRC, "Energy Northwest's NEI 12-02 Phase 2 Staffing Assessment," dated December 23, 2014 (ADAMS ML15006A030 – Not Available)
14. Letter from M. K. Halter (NRC) to M. E. Reddemann (Energy Northwest), "Columbia Generating Station – Response Regarding Phase 2 Staffing Submittals Associated with Near-Term Task Force Recommendation 9.3 Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident," dated June 11, 2015 (TAC No. MF551) (ADAMS ML15156B285)
15. Letter GO2-15-175 from A. L. Javorik (Energy Northwest) to the NRC, "Energy Northwest's Response to NRC Order EA-13-109 – Overall Integrated Plan for

Reliable Hardened Containment Vents under Severe Accident Conditions Phases 1 and 2, Revision 1," dated December 16, 2015 (ADAMS ML15351A363)

COLUMBIA GENERATING STATION, DOCKET NO. 50-397

OPEN ITEMS FROM THE JUNE 16, 2015 REPORT FOR THE AUDIT REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051

Background

The NRC staff issued the Columbia interim staff evaluation (ISE) on January 29, 2014 (ADAMS Accession No. ML13337A365). The NRC staff conducted an onsite audit at Columbia from February 2-5, 2015, in accordance with the audit plan dated January 16, 2015 (ADAMS Accession No. ML15006A322). The NRC issued the audit report on June 16, 2015 (ADAMS Accession No. ML15139A462). The report stated that Attachment 3 of the audit report provides the status of all open audit review items that the NRC staff is evaluating from the following five sources.

a. ISE open items (OI) and confirmatory items (CI)
b. Audit questions (AQ)
c. Licensee-identified overall integrated plan (OIP) OIs
d. Spent Fuel Pool level Instrumentation (SFPLI) RAIs
e. Additional Staff Evaluation (SE) needed information

The Attachment 3 table of the audit report along with the requested information follows:

Columbia Generating Station Mitigation Strategies/Spent Fuel Pool Instrumentation Safety Evaluation Audit Items: Audit Items Currently Under NRC Staff Review, Requiring Licensee Input As Noted		
Audit Item Reference	**Item Description**	**Licensee Input Needed**
ISE OI 3.1.2.1.A	Confirm that FLEX equipment can be adequately protected and deployed in such an event and whether flooding procedures account for the use of FLEX equipment.	Licensee Open Items 43 through 45 remain open as EN is performing a local intense precipitation analysis. **Energy Northwest Response** The Columbia Flooding Hazard Reevaluation Report was submitted on October 6, 2016 (ADAMS ML16286A309) and responded to in NRC letter dated December 7, 2016 (ADAMS PKG ML16337A111) The flooding hazard reevaluation report shows that the results are either bounded by the current design basis or available physical margin exists.
ISE CI 3.2.1.4.A	The licensee has not completed calculations supporting the design of the FLEX equipment. Confirm that portable FLEX equipment is adequate to perform its credited mitigation function(s).	Licensee to evaluate head loss to the spent fuel pool while simultaneously filling SFP and RPVL. **Energy Northwest Response** See attached discussion.
ISE CI 3.2.2.D	Confirm that EN's SFP makeup strategy for Columbia provides for SFP makeup without accessing the refueling floor, as recommended in NEI 12-06, Table C-3, or that an acceptable alternate approach is developed.	Licensee to evaluate flow analysis for filling SFP through RHR B loop. **Energy Northwest Response** See attached discussion.

Columbia Generating Station Mitigation Strategies/Spent Fuel Pool Instrumentation Safety Evaluation Audit Items: Audit Items Currently Under NRC Staff Review, Requiring Licensee Input As Noted		
Audit Item Reference	**Item Description**	**Licensee Input Needed**
ISE CI 3.2.3.B	The licensee's proposed strategy for maintaining containment will rely on installation of the HCVS as required by Order EA-13-109. When complete, the licensee's calculations supporting the revised containment response and sequence of events timeline should be reviewed to confirm that the timeline is appropriate and that containment functions will be restored and maintained following an ELAP event.	The licensee needs to provide to the NRC staff the final configuration and calculations for the HCVS. **Energy Northwest Response** See attached discussion.
AQ 41	The alternate strategy for Phase 2 core cooling involves removal, replacement, and reconfiguration of several flanges and piping elbows during the ELAP event. The NRC staff requests that the licensee provide a description of the available lighting and habitability around the RHR piping where connections need to be made.	The NRC staff asked the licensee to provide further detail of the paths and the locations of the connections [sic] points as well as the validation of the ability to perform the actions. **Energy Northwest Response** See attached discussion.
AQ 52	On page 18 of 60 Columbia's OIP states that load shedding will be performed to "prolong battery life to 10 hours." On page 22 of 60 Columbia's OIP states, "The 125 VDC batteries are available for 10 hours without recharging. The 250 VDC batteries are available for 17 hours without recharging." On page 35 of 60, with reference to power for containment hardened vent valve solenoids and instrumentation, Columbia's OIP states, "This battery will be designed to support at least 24 hours of operation without any outside power source." Provide justification for the above discrepancy.	The licensee to design the containment hardened vent system battery for a cycle of 24 hours. **Energy Northwest Response** See attached discussion.

Columbia Generating Station Mitigation Strategies/Spent Fuel Pool Instrumentation Safety Evaluation Audit Items: Audit Items Currently Under NRC Staff Review, Requiring Licensee Input As Noted		
Audit Item Reference	**Item Description**	**Licensee Input Needed**
1-E	Please address the following items regarding the use of raw water sources for mitigating an ELAP event: a. Discuss the quality of the water (e.g., suspended solids, dissolved salts) that will be used for primary makeup during ELAP events, accounting for the potential for increased suspended or dissolved material in some raw water sources during events such as flooding or severe storms. b. Discuss whether instrumentation available during the ELAP event is capable of providing indication that inadequate core cooling exists for one or more fuel assemblies due to blockage at fuel assemblies' inlets or applicable bypass leakage flowpaths. c. Provide justification that the use of the intended raw water sources will not result in blockage of coolant flow across fuel assemblies' inlets and applicable bypass leakage flowpaths to an extent that would inhibit adequate core cooling. Or, if deleterious blockage at the core inlet cannot be precluded under ELAP conditions, then please discuss alternate means for assuring the adequacy of adequate core cooling in light of available indications.	Licensee to justify that the ashfall event would not plug the inlets of the fuel assemblies or that top down cooling would be used to ensure core cooling. **Energy Northwest Response** See attached discussion.

Columbia Generating Station Mitigation Strategies/Spent Fuel Pool Instrumentation Safety Evaluation Audit Items: Audit Items Currently Under NRC Staff Review, Requiring Licensee Input As Noted		
Audit Item Reference	**Item Description**	**Licensee Input Needed**
2-E	a. Discuss the design of the suction strainers used with FLEX pumps taking suction from raw water sources, including perforation dimension(s) and approximate surface area. b. Provide reasonable assurance that the strainers will not be clogged with debris (accounting for conditions following, flooding, severe storms, earthquakes or other natural hazards), or else that the strainers can be cleaned of debris at a frequency that is sufficient to provide the required flow. In the response, consider the following factors. i. The timing at which FLEX pumps would take suction on raw water relative to the onset and duration of the natural hazard. ii. The timing at which FLEX pumps would take suction on raw water relative to the timing at which augmented staffing would be available onsite. iii. Whether multiple suction hoses exist for each FLEX pump taking suction on raw water, such that flow interruption would not be required to clean suction strainers.	Licensee to analyze the suction strainer design and how far into the water it sits as well as the procedure for ensuring that flow is not interrupted to such a length of time that the fuel would remain covered. **Energy Northwest Response** See attached discussion.
10-E	Evaluation of FLEX equipment to be completed to ensure proper functioning under the design-basis temperatures and ash fall conditions during both operation and storage. This includes manual actions to transport and set up the equipment as well as storage conditions.	Licensee to complete evaluation of operating FLEX equipment under ash fall conditions. **Energy Northwest Response** See attached discussion.

Columbia Generating Station Mitigation Strategies/Spent Fuel Pool Instrumentation Safety Evaluation Audit Items: Audit Items Currently Under NRC Staff Review, Requiring Licensee Input As Noted		
Audit Item Reference	**Item Description**	**Licensee Input Needed**
11-E	Please provide an assessment of potential susceptibilities of EMI/RFI in the areas where the SFP instrument is located and how to mitigate those susceptibilities.	A strategy to mitigate EMI/RFI interference in the SFP area. **Energy Northwest Response** This response was provided in Energy Northwest letter GO2-15-120 (ADAMS ML15245A530) from W. G. Hettel (Energy Northwest) to the NRC, "Completion of Required Action By Nuclear Regulatory Commission (NRC) Order EA-12-051 Reliable Spent Fuel Pool Instrumentation," dated August 12, 2015 (ADAMS ML15245A530)
14-E	The licensee is requested to provide a summary evaluation to confirm that the temperature and pressures within containment will not exceed the environmental qualification (EQ) of electrical equipment that is being relied upon as part of their FLEX strategies. The licensee needs to ensure that the EQ profile of the required electrical equipment remains bounding for the entire duration of the event.	Provide EQ evaluation **Energy Northwest Response** See attached discussion.

ISE CI 3.2.1.4.A: Licensee to evaluate head loss to the spent fuel pool while simultaneously filling SFP and RPVL.

Response:

As stated in Section 3.1.2 of the FIP, the Columbia Phase 2 response uses one of the following on-site FLEX pumps. The supporting calculations have been added to the response.

Phase 2 FLEX Equipment				
Plant EPN No. or FLEX Tag No.	**Component Description**	**Location**	**FLEX Strategy Use**	**Supporting Calculation**
B5B-P-1	On-site High-Head Pump, B5b Pumper Truck	FLEX Storage Building 600	Motive force for Reactor coolant water makeup	ME-02-12-06 R2
FLEX-P-1	On-site Hi-Head Godwin Pump HL130M	FLEX Storage Building 82	Motive force for Reactor coolant water makeup	ME-02-12-06 R2

The size of the hose from the SFP/RPV branch to the SFP was increased from 2.5" to 4", thus reducing the required head and increasing total flow. This change provides significant margin.

ISE CI 3.2.2.D: Licensee to evaluate flow analysis for filling SFP through RHR B loop.

Response:

During Phase 2, Columbia's primary FLEX strategy for supplying water to the spent fuel pool (SFP) utilizes temporary hoses routed in the yard to the reactor building. One of the two FLEX pumps will be used to supply water from the service water spray ponds to the SFP. The spray pond water will be supplied directly into the SFP thereby allowing the overflow to cascade down the fuel pool cooling (FPC) and residual heat removal (RHR) piping to the suppression pool.

An alternate strategy has been developed to supply makeup water to the SFP when the reactor building refueling floor (606') is inaccessible. This alternate method uses the same temporary hose scenario as above but with a special connector assembly that will be installed at RHR-V-63B to inject makeup water directly into the RHR-B Loop piping system. With valve manipulations, a flow path can be established between RHR-B loop and FPC using the existing cross tie. The RHR B-loop is the only RHR loop that has an existing cross tie with FPC to provide a supplemental path for filling the SFP.

ISE CI 3.2.3.B: The licensee needs to provide to the NRC staff the final configuration and calculations for the HCVS.

Response:

The final installation of the hardened containment vent is complete. The final configuration has not changed from the sketches provided in letter GO2-15-175 dated December 16, 2015. The pipe sizing and pressure drop calculation has been completed. The final flow coefficient value for the piping and components is K = 3.36. [ME-02-13-03 R1, CMR 16448 R0]

Audit Question 41: The NRC staff asked the licensee to provide further detail of the paths and the locations of the connections points as well as the validation of the ability to perform the actions.

Response:

Water is supplied to the RHR system by installing a flanged connection hose fitting to the blind flange of 3" gate valve RHR-V-63A, RHR-V-63B, or RHR-V-63C. If the reactor building 606' elevation is accessible, the preferred connection point is RHR-V-63A based on valve location and ease of installation. If inaccessibility of RB 606' floor is anticipated, then RHR-B must be used. RHR-B loop is the only loop that has an existing cross tie with FPC to provide a supplemental flow path to the SFP for makeup and cascading flow. Reactor building lighting is not credited in these areas; operators carry flashlights during an ELAP event. The location, accessibility, and complexity for installing a RHR flange connector at each valve location is discussed below. ABN-FSG-002, *Water Makeup Strategies for RPV, SFP, DW, WW, CSTS during an Extended Loss of AC Power or other Beyond Design Basis Event*, details precautions, equipment needed, how the fire hoses are laid out and provides maps associated with the Columbia makeup water strategies.

Note that in the responses the survey maps discussed are the maps provided to the inspectors during the audit.

<u>Connection to RHR-V-63A (preferred connection point – if reactor building 606' is accessible):</u>

Location: Reactor Building 516' elevation, column K3/4.2.

Habitability/Exposure: A recent survey map near RHR-V-63A shows smear results were less than 1,000 dpm/100 cm2 and dose was 2-4 mrem/hr.

Accessibility: RHR-V-63A is located above the reactor building 501' floor in the TIP Mezzanine Room. It is a locked high radiation/contaminated area and a key is required for access. Operators will route the fire hose up the reactor building

northeast stairwell (A-5), through the locked gated doorway R312 to room R312. A permanent ladder is installed to the immediate right of the room entrance. In room R312, the fire hose must be carried up the ladder. The RHR flange connection is located roughly 5' off grated floor near the ladder manway on the mezzanine level (floor 510'-6") at elevation 515'-10.75".

Complexity of installation: To install the RHR connector to RHR-V-63A, unbolt and remove the 3" blind flange and condensate spool piece (COND-RSP-2) allowing space to install a flange connector. The fire hose is connected using a storz fitting. No ladders are required for installation work. No emergency lights are in the room; therefore flashlights are necessary to perform installation.

<u>Connection to RHR-V-63C (alternate connection point, if reactor building 606' is accessible):</u>

Location: Reactor Building 523' elevation, column H8/5.1.

Habitability/Exposure: A recent survey map near RHR-V-63C shows smear results were less than 1,000 dpm/100 cm2 and dose was 17-20 mrem/hr.

Accessibility: RHR-V-63C is located above the reactor building 522' floor in the north valve room, a locked high radiation/contaminated area. A key is required for access. Using the fire hose stored in the B.5.b storage cabinets, the fire hose is routed up either the reactor building northeast stairwell (A-5) or southwest stairwell (A-6), across the 522' general area, through locked gated doorway R404, and into room R408. RHR-V-63C is located a foot off the floor and a third of the way into the room.

Complexity of installation: To install the RHR connector to RHR-V-63C, a hanger is removed, a piping elbow is removed, the 3" blind flange is removed, and the elbow is rotated 45º and reattached to allow space for the flange connector to be installed. The fire hose is connected to the RHR adaptor using a storz fitting. Ladders are not required for work. No emergency lights are in the room, therefore flashlights are necessary.

<u>Connection to RHR-V-63B (alternate - connection point if reactor building 606' elevation is inaccessible):</u>

If inaccessibility of reactor building 606' floor is anticipated, the RHR-B loop must be used. RHR-B is the only loop that has an existing cross tie with FPC to provide a supplemental flow path to the SFP for makeup. RHR-V-63B is the only loop connection point that requires a ladder to access.

Location: Reactor Building 515' elevation, column K5/7.9.

Habitability/Exposure: A recent survey map near RHR-V-63B shows smear results were less than 1,000 dpm/100 cm2 and dose was 6-15 mrem/hr.

Accessibility: RHR-V-63B is located on the reactor building 501' in the RHR-B Valve Room (R319), a contaminated/high radiation area. Using the fire hose stored in the B.5.b storage cabinets, the fire hose is routed up the reactor building southwest stairwell (A-6) across the 501' and through the open doorway into room R319. RHR-V-63B is located to the immediate right of the room entrance approximately 14' above the floor. The fire hose is carried up the ladder and connected to the blind flange. A ladder is permanently stored outside Room R319.

Complexity of installation: To install the RHR connector to RHR-V-63B, unbolt and remove the 3" blind flange and condensate spool piece (COND-RSP-3) allowing space to install a flange connector. The fire hose is connected to the RHR adaptor using a storz fitting. Ladders are required for work. No emergency lights are in the room, therefore flashlights are necessary.

Audit Question 52: The licensee to design the containment hardened vent system battery for a cycle of 24 hours.

Response:

The HCV battery calculation E/I-02-13-03 R1 concluded that:

- The vented lead-acid cells defined by this calculation are capable of supplying the HCV system load for the 24 hours discharge cycle during an extended loss of AC power. This meets the specified 24 hours following the loss of normal power required by the NRC Order EA-13-109.
- Throughout the duty cycle, the selected battery is capable of maintaining the DC voltage at or above the 1.75 volts per cell for twenty-four hours. For the specified battery duty cycle and the cell size selected, the average cell voltage will not drop below the specified minimum (e.g. 1. 75 V) at any point in the duty cycle.

1-E: Licensee to justify the ashfall event would not plug the inlets of the fuel assemblies or that top down cooling would be used to ensure core cooling.

Response:

The strategies for Phase 2 and 3 use the service water spray ponds as a source of water. The Final Safety Analysis Report (FSAR) Section 1.2.2.5.14 states that the spray pond water is safety grade water.

Plant procedure 12.14.1, *Chemical Treatment of the Standby Service Water System*, provides controls and documentation for chemical treatment of the standby service water system (SSW). Chemical treatment consists of the addition of a corrosion

inhibitor and biological inhibitors. Blowdown and makeup are performed as necessary to maintain the control limits as defined in SWP-CHE-02.

SWP-CHE-02, *Chemical Process Management and Control*, provides the Parameter Action Levels (PAL) which apply to standby service water.

Technical Specification Surveillance Requirement 3.7.1.4 is to verify average sediment depth in each UHS spray pond is < 0.5 ft. performed in accordance with the Surveillance Frequency Control Program.

Calculation ME-02-15-04, *Potential Effects of Volcanic Ash in Spray Pond Water*, concluded that considering (1) the volume of the water in the RPV and SFP, and (2) the very small ash particle size (< 1mm), the relatively small amount of ash entering the RPV and SFP cannot reasonably be expected to cause problems with the cooling of the fuel. Therefore, no special actions are required in the RPV or SFP during or following ashfall to support the cooling of the fuel

2-E: Licensee to analyze the suction strainer design and how far into the water it sits as well as the procedure for ensuring that flow is not interrupted to such a length of time that the fuel would remain covered.

Response:

Either FLEX pump will be supplied from the service water spray ponds by dual suction lines, each with a floating strainer. Either line can be isolated to clean the strainer while allowing the other to continue to supply the pump's suction.

Time Critical Validation Plan 10 – Connect FLEX Equipment to Refill SFP, identifies that this activity starts approximately 2 hours into the event with a time constraint of 12 hours. The activity is shown to complete in 3 hours providing a 7 hour margin.

Procedure ABN-FSG-002, *Water Makeup Strategies for RPV, SFP, DW, WW, CSTS During an Extended Loss of AC Power or other Beyond Design Basis Event*, contains instructions on connecting a wye suction to either FLEX pump which allows switching from one suction strainer to the other to allow cleaning if the in-service strainer becomes fouled.

During testing of the FLEX pump, the suction strainer will remain at the surface as shown in the photograph taken during testing. From FSAR Table 9.2-1, "Ultimate Heat Sink Spray Cooling Pond Design," water depth is nominally 13 feet 6 inches at the beginning of initiation of makeup using the FLEX pumps. Even at minimum capacity, 12 feet 6 inches of water depth is available.

10-E: Licensee to complete evaluation of operating FLEX equipment under ash fall conditions.

Response:

As stated in FSAR Section 9.2.5.3, the design basis ashfall is 3 in. which bounds the Mount St. Helens eruption of May 18, 1980. In addition to the normal air filters supplied with the FLEX equipment, Energy Northwest has purchased the following oil bath filters for the protection of FLEX equipment during ashfall conditions. The oil bath filters are stored in the FLEX buildings along with twelve 5-quart containers of 5W-30 oil for use with the oil bath filters.

- DG4 Oil Bath Filter
- DG5 Oil Bath Filter
- FLEX-P-1 Oil Bath Filter
- B.5.b Pumper Truck Oil Bath Filter
- Building 82 House Generator Oil Bath Filter
- Building 600 House Generator Oil Bath Filter

Procedure ABN-ASH, Attachment 7.5, contains the instruction for installation oil bath filter and refilling/replacing the filter's oil.

The response to 1-E addresses ash fall in the service water spray ponds.

FLEX Buildings 82 and 600, has been documented to meet the requirements of ASCE 7-10 and are designed to be structurally capable of withstanding wind loading and ash fall deposit. As Columbia occasionally has snow, plans already are in place to remove ice and snow and can be used to any excessive accumulation of ash from the FLEX equipment deployment routes shown in Figure 2 of the Final Integrated Plan.

As documented in FSAR Chapter 9.4, *Heating, Ventilating, and Air Conditioning Systems*, the summer outdoor design temperature for Columbia is 105 °F (dry-bulb) with an extreme outdoor summer condition of 115°F (dry-bulb). Specifically, FLEX equipment was procured to function in weather conditions applicable to Columbia and include block heaters. The towing and debris removal equipment stored in the FLEX buildings were purchased with block heaters.

14-E: Provide EQ evaluation

Response:

The solenoid pilot valves (SPV) which actuate the automatic depressurization system safety relief valves (SRV) are environmentally qualified for loss of coolant accident (LOCA) and an anticipated transient without a SCRAM (ATWS) event as documented in QID 315008-01. Analysis of the testing shows that the equipment is capable of functioning for the duration of the ELAP event (conservatively taken as 300° Fahrenheit (F) for 72 hours).

The SRV actuators were LOCA tested for 96 hours at over 308°F as documented in QID 297009-02. The entire LOCA test duration was utilized for the accident profile, and separate testing was conducted for thermal qualified life. Therefore, the entire test profile can be applied to the ELAP profile. This bounds the entire ELAP profile (conservatively taken as 300°F for 72 hours).

COLUMBIA GENERATING STATION, DOCKET NO. 50-397

ADDITIONAL INFORMATION REQUESTED IN "INTERIM STAFF EVALUATION AND REQUEST FOR ADDITIONAL INFORMATION REGARDING THE OVERALL INTEGRATED PLAN FOR IMPLEMENTAION OF ORDER EA-12-051, RELIALE SPENT FUEL POOL INSTRUMENTATION"

RAI No.1

Please provide the information regarding compensatory measures and actions to be incorporated into station procedures controlling irradiated equipment or materials stored in the SFP, including the results of the calculation to be performed to determine the projected dose rate impact and the appropriate Level 2 value as a result of other material stored in the SFP.

Energy Northwest Response

All necessary compensatory measures and actions have been incorporated into the following documents.

SOP-FPC-LEVEL-OPS contains the following information in regards to Level 2 in the spent fuel pool.

> In accordance with NEI-12-02, the three critical levels that must be monitored in a spent fuel pool are as follows:
>
> Level 2 - Level that is adequate to provide substantial radiation shielding for a person standing on the Spent Fuel Pool Operating deck. Level 2 (593 feet 2 inches) represents the range of water level where any necessary operation in the vicinity of the SFP can be completed without significant dose consequences from direct gamma radiation from the SFP. Level 2 is based on either of the following:
>
> - 10 feet (+/-1 foot) above the highest point of any fuel rack (583 feet 2 inches) seated in the SFP, or
> - A designated level that provides adequate radiation shielding to maintain personnel radiological dose levels within acceptable limits while performing local operation in the vicinity of the SFP.

SOP-FPC-LEVEL-OPS also contains information on the operation and use of the EFP-IL display panel including the alarms displayed for Level 2.

Setting of Alarm 2	LEVEL 2 (10 ft. above top of Rack)
Level feet	10.0
Message	LOW Level Lose Fuel Pool Overflow
Warning	ALERT

SOP-FPC-LEVEL-OPS also contains a note that discusses Level 3 and that it should not be interpreted to imply that actions in initiate water make-up should be delayed until SFP water levels have reached or lowered past Level 3.

Procedure ABN-FPC-LOSS provides the actions to be taken for an unplanned loss of inventory.

Procedure PPM 6.1.1 establishes an inventory of significant material present in the spent fuel pool. This inventory has a twofold purpose. The first purpose is to document and track radioactive or irradiated equipment (excluding special nuclear material) that would be expected to be removed during a Spent Fuel Pool cleanup campaign. The second purpose is to fulfill the labeling exemption requirements of 10 CFR 20.1905. The procedure also defines when an inventory is required to be conducted.

Procedure PPM 9.2.1 establishes the accountability requirements of special nuclear material (SNM). The accountability is maintained through the inventory and control processes. The procedure controls the movement and locations of SNM and defines when an inventory is required.

Calculation CVI 1201-00,3 determined the doses to new spent fuel pool water level instrument and selected SFP operating deck areas. Case 1 of the calculation defined the total dose rate calculations performed for the operating deck general area dose rates, including the southeast stairwell location between 5' above the deck, and 5 feet below the operating deck. Various levels between and including Level 1 (normal), Level 2 (10 feet above top of fuel/top of racks), and Level 3 (top of fuel/top of racks) were evaluated. The graphic representations of dose rates are provided for Level 1, Level 2, Level 2+3 feet (just below the control blade rollers), Level 2+6 ft. just above the control blade rollers), and Level 3.

At the Level 1 water level, the maximum dose rates are about 0.035 mR/hr. in the southeast corner of the pool area, and about 0.2 mR/hr. in the southwest corner of the pool. This dose rate is primarily due to the Co-60 stellite roller sources that are assumed to completely fill the control blade hangers on the south and east walls.

At the Level 2+6', or 16 feet above the fuel which is 9.78 inches or 0.815 ft. above the stellite roller elevation, the stellite rollers are covered by some water and the fuel has more than half of the water shielding as the Level 1 case. Dose rates are much higher than the Level 1 water level case. Dose rates are about 1.0E+04 mR/hr. in the southeast corner of the pool area, and about 3.5 E+04 mR/hr in the southwest corner of the pool.

At Level 2+ 3', or 13 feet above the fuel which is 20.2 inches, or 1.68 feet below the stellite roller elevation, the stellite rollers are exposed to air, and the fuel has about half of the water shielding as the Level 1 case. The maximum dose rates of about 6.5 E+04 mR/hr. and 1.5 E +05 mR/hr. are near the southeast and southwest corners of the spent fuel pools, respectively.

At Level 2, or about 10 feet above the fuel, the stellite rollers are completely exposed to air and the fuel has less than half of the shielding of the Level 1 case. Dose rates with the water level at Level 2+3 and at Level 2 are similar. The maximum dose rates of about 6.5 E+4 mR/hr. and 1.5E+05 mR/hr occur near the southeast and southwest comers of the SFP.

RAI No.2

Please provide the following:

a) A clearly labeled sketch or marked-up plant drawing of the plan view of the SFP area, depicting the SFP inside dimensions, the planned locations/placement of the primary and back-up SFP level sensor, and the proposed routing of the cables that will extend from the sensors toward the location of the read-out/display device to meet the Order requirement to provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the SFP.

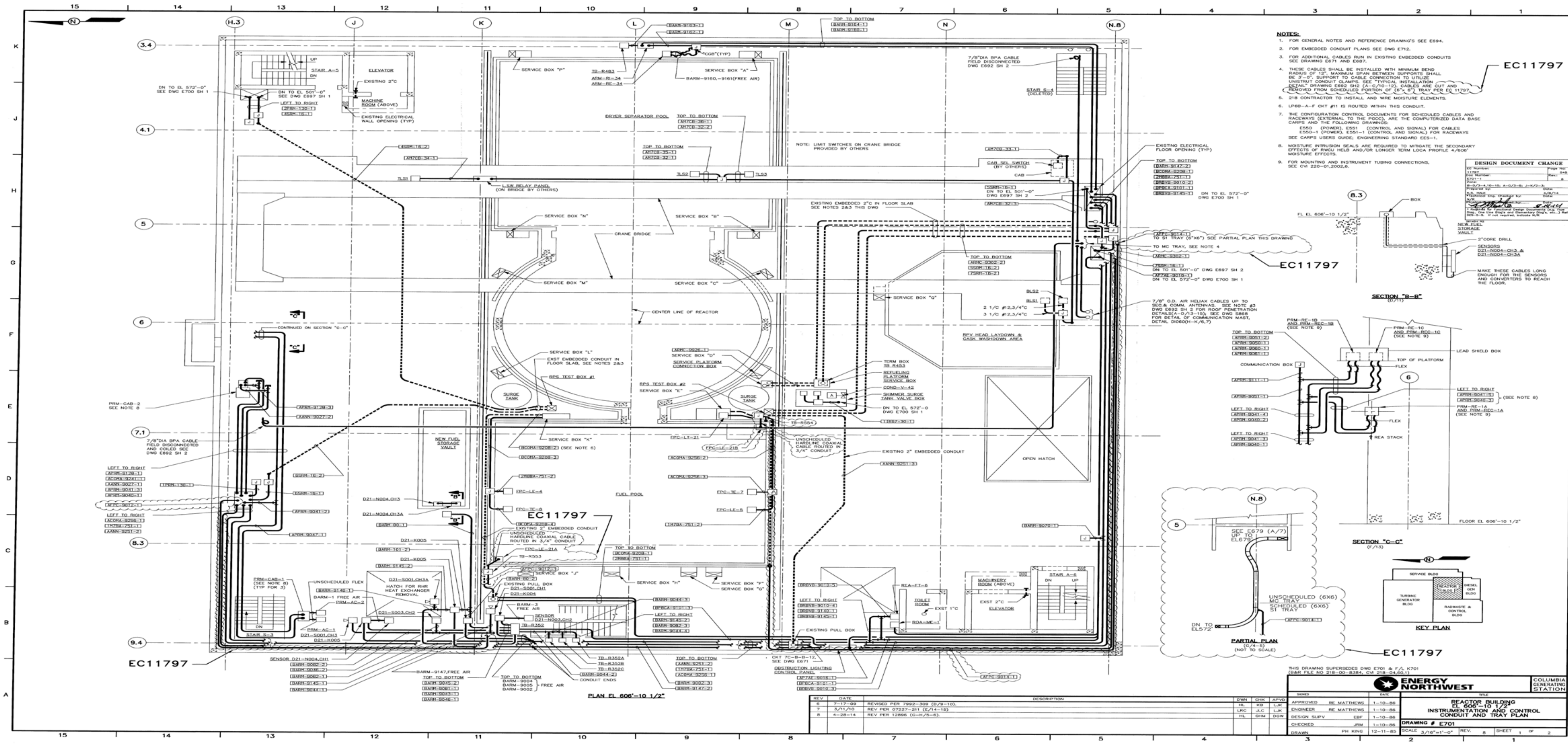
b) Please address how other hardware stored in the SFP will not create adverse interaction with the fixed instrument location(s).

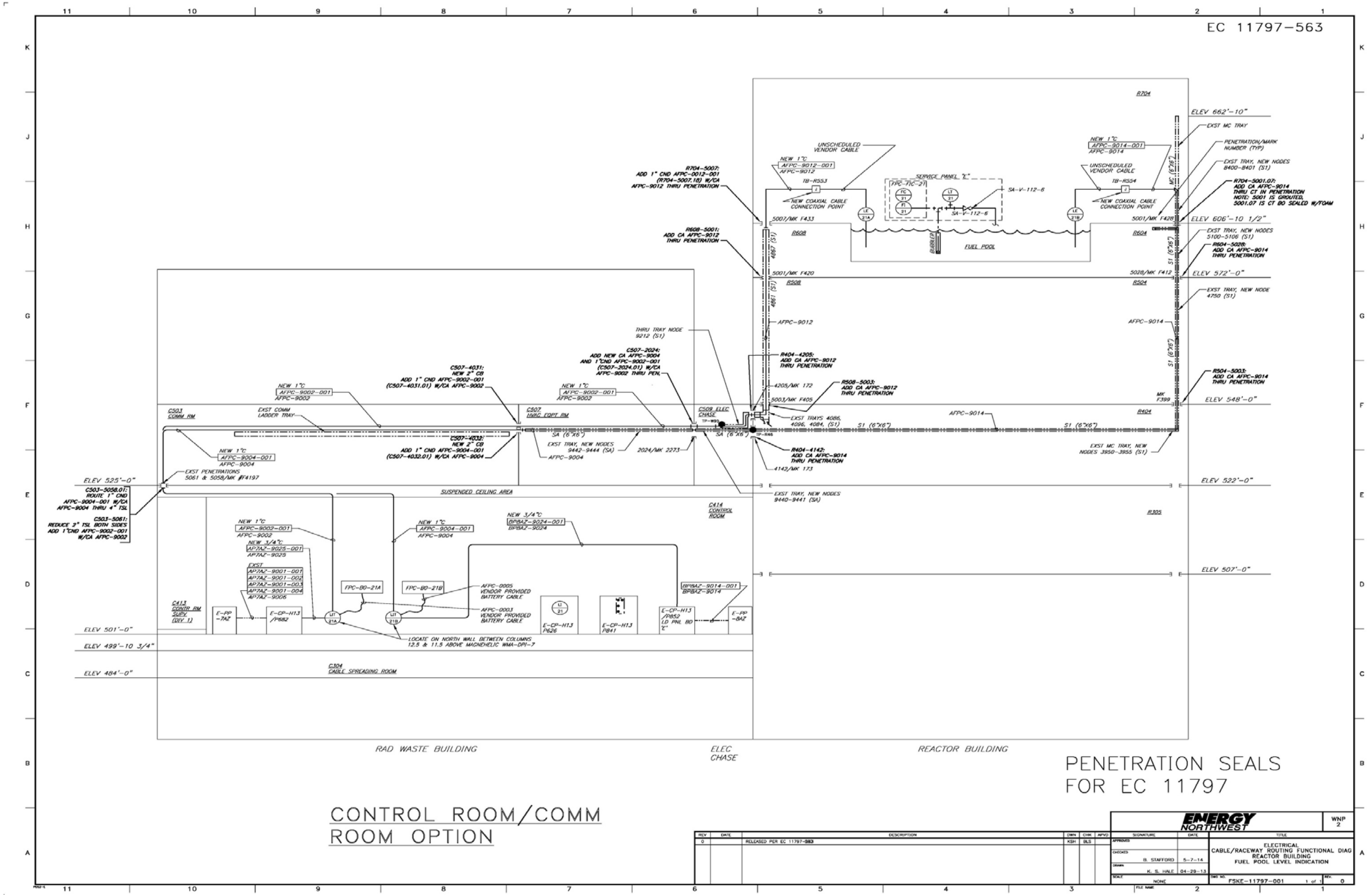
__**Energy Northwest Response**__

a) Sketches are provided below.

The spent fuel pool level instrumentation (SFPLI) has been installed in the northwest and southeast corners of the SFP. Material stored in the SFP (irradiated control blades, used local power range monitors (LPRMs) etc.) are stored in designated locations not in close proximity of the SFPLI (approximately 10 feet away). The diverse locations of the level instrumentation reduce the probability of a single interaction affecting both channels of the level instrumentation.

In addition, an inventory of significant material present in the SFP is governed by Procedure 6.1.1, Spent Fuel Pool Inventory. This procedure is used to prevent any instrument interference from material stored in the SFP.


NOTES:
1. FOR GENERAL NOTES AND REFERENCE DRAWING'S SEE E694.
2. FOR EMBEDDED CONDUIT PLANS SEE DWG E712.
3. FOR ADDITIONAL CABLES RUN IN EXISTING EMBEDDED CONDUITS SEE DRAWING E671 AND E687.
4. THESE CABLES SHALL BE INSTALLED WITH MINIMUM BEND RADIUS OF 12". MAXIMUM SPAN BETWEEN SUPPORTS SHALL BE 3'-0". SUPPORT TO CABLE CONNECTION TO UTILIZE UNISTRUT CONDUIT CLAMPS. SEE "TYPICAL INSTALLATION DETAIL" DRAWING E692 SH2 (A-C/10-12). CABLES ARE CUT AND REMOVED FROM SCHEDULED PORTION OF (6"x 6") TRAY PER EC 11797.
5. 218 CONTRACTOR TO INSTALL AND WIRE MOISTURE ELEMENTS.
6. LP6B-A-F CKT #11 IS ROUTED WITHIN THIS CONDUIT.
7. THE CONFIGURATION CONTROL DOCUMENTS FOR SCHEDULED CABLES AND RACEWAYS (EXTERNAL TO THE PGCC), ARE THE COMPUTERIZED DATA BASE CARPS AND THE FOLLOWING DRAWINGS:
E550 (POWER), E551 (CONTROL AND SIGNAL) FOR CABLES
E550-1 (POWER), E551-1 (CONTROL AND SIGNAL) FOR RACEWAYS
SEE CARPS USERS GUIDE; ENGINEERING STANDARD EES-1.
8. MOISTURE INTRUSION SEALS ARE REQUIRED TO MITIGATE THE SECONDARY EFFECTS OF RWCU HELB AND/OR LONGER TERM LOCA PROFILE 4/606' MOISTURE EFFECTS.
9. FOR MOUNTING AND INSTRUMENT TUBING CONNECTIONS, SEE CVI 220-01,2002,6.
EC11797
DESIGN DOCUMENT CHANGE
PLAN EL 606'-10 1/2"
SECTION "B-B"
SECTION "C-C"
PARTIAL PLAN
(G/4-5)
(NOT TO SCALE)
KEY PLAN
THIS DRAWING SUPERSEDES DWG E701 & F/L K701
(B&R FILE NO 218-00-8384, CVI 218-04,60,1)
ENERGY NORTHWEST
COLUMBIA GENERATING STATION
APPROVED RE MATTHEWS 1-10-86
ENGINEER RE MATTHEWS 1-10-86
DESIGN SUPV EBF 1-10-86
CHECKED JRM 1-10-86
DRAWN PH KING 12-11-85
REACTOR BUILDING
EL 606'-10 1/2"
INSTRUMENTATION AND CONTROL
CONDUIT AND TRAY PLAN
DRAWING # E701
SCALE 3/16"=1'-0"
REV. 8
SHEET 1 OF 2


EC 11797-563
ELEV 662'-10"
ELEV 606'-10 1/2"
ELEV 572'-0"
ELEV 548'-0"
ELEV 525'-0"
ELEV 522'-0"
ELEV 507'-0"
ELEV 501'-0"
ELEV 499'-10 3/4"
ELEV 484'-0"
FUEL POOL
SUSPENDED CEILING AREA
C304 CABLE SPREADING ROOM
RAD WASTE BUILDING
ELEC CHASE
REACTOR BUILDING
PENETRATION SEALS
FOR EC 11797
CONTROL ROOM/COMM
ROOM OPTION
ENERGY NORTHWEST
WNP 2
RELEASED PER EC 11797-563
B. STAFFORD 5-7-14
K. S. HALE 04-29-13
ELECTRICAL
CABLE/RACEWAY ROUTING FUNCTIONAL DIAG
REACTOR BUILDING
FUEL POOL LEVEL INDICATION
FSKE-11797-001

RAI No.3

Please provide the following:

a) The design criteria that will be used to estimate the total loading on the mounting device(s), including static weight loads and dynamic loads.

   Please describe the methodology that will be used to estimate the total loading, inclusive of design basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing or other effects that could accompany such seismic forces.

b) A description of the manner in which the level sensor (and stilling well, if appropriate) will be attached to the refueling roof and/or other support structures for each planned point of attachment of the probe assembly. Please indicate in a drawing the portions of the level sensor that will serve as points of attachment for mechanical/mounting and electrical connections.

c) A description of the manner by which the mechanical connections will attach the level instrument to permanent SFP structures so as to support the level sensor assembly.

<u>**Energy Northwest Response**</u>

a) The loading on the probe mount and probe body includes both seismic and hydrodynamic loading using seismic response spectra that bounds the site design basis maximum seismic loads applicable to the installation location(s). The static weight load is also accounted for in the modeling described below but is insignificant in comparison to seismic and hydrodynamic loads. Analytic modeling has been performed by the instrument vendor using Institute of Electrical and Electronic Engineers (IEEE)-344:2004, *Standard for Seismic Qualification of Equipment for Nuclear Power Plants*, methodology.

   The simple unibody structure of the probe assembly make it a candidate for analytic modeling and the dimensions of the probe and complex hydrodynamic loading terms in any case preclude meaningful physical testing.

   A detailed computational SFP hydrodynamic model has been developed for the instrument vendor by Numerical Applications, Inc., author of the GOTHIC computational fluid dynamics code. The computational model accounts for multi-dimensional fluid motion, pool sloshing, and loss of water from the pool.

   Seismic loading response of the probe and mount is separately modeled using finite element modeling software. The GOTHIC-derived fluid motion profile in the pool at the installation site and resultant distributed hydrodynamic loading terms are added to the calculated seismic loading

terms in the finite element model to provide a conservative estimate of the combined seismic and hydrodynamic loading terms for the probe and probe mount, specific to the chosen installation location for the probe.

b) The proximal portion of the level probe is designed to be attached near its upper end to a Seismic Category I mounting bracket configured to suit the requirements of the Columbia SFP. The bracket is welded to the SFP deck per Seismic Category I requirements.

c) See b above.

RAI No.4

Please provide the analyses used to verify the design criteria and methodology for seismic testing of the SFP instrumentation and the electronics units, including, design basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing or other effects that could accompany such seismic forces.

Energy Northwest Response

Signal processor (electronics) and Extended Batteries: Test was conducted to the Table Limits by the vendor to envelope Seismic Category 1 safe shutdown earthquake (SSE) conditions using IEEE-344:2004 methodology. The test specimen was monitored for structural integrity and loosening of fasteners; no loss of structural integrity or loose fasteners was noted. Seismic test results for the SFPI signal processing unit and the extended battery were documented, reviewed, and accepted by Energy Northwest personnel in accordance with Energy Northwest procedures and policies.

Probe assembly (level sensor): Seismic and hydrodynamic finite element analysis is performed by the vendor using relevant IEEE-344:2004 methodology (using enveloping seismic category 1 SSE conditions or site design basis maximum seismic loads relative to the location where the equipment is mounted). The sloshing analysis was based on GOTHIC, an industry-standard computer code for performing multiphase fluid flow. ANSYS, a finite element analysis computer code, was used to perform the hydrodynamic loading and structural analysis. A code-to-code verification was performed between ANSYS and GOTHIC with good results which were documented, reviewed, and accepted by Energy Northwest personnel in accordance with Energy Northwest procedures and policies.

Level Sensor Bracket: The probe is mounted as a cantilever onto the pool curb via a bracket in accordance with Energy Northwest calculation CE-02-13-13. The bracket itself and its mounting details are seismically qualified with the load combination of gravity plus the resulting of 3D hydrodynamic and internal loads from the vendor test

results plus the seismic loads of all 3 directions using the bounding seismic factor of 5.384g.

RAI No.5

For each of the mounting attachments required to attach SFP Level equipment to plant structures, please describe the design inputs, and the methodology that was used to qualify the structural integrity of the affected structures/equipment.

Energy Northwest Response

The SFP level probe is mounted as a cantilever onto the pool curb via a bracket. See the response to RAI No. 3.

RAI No.6

Please provide the following:

a) A description of the specific method or combination of methods you intend to apply to demonstrate the reliability of the permanently installed equipment under BDB ambient temperature, humidity, shock, vibration, and radiation conditions.

b) Further information indicating what will be the maximum expected ambient temperature in the room in which the sensor electronics will be located under BDB conditions in which there is no ac power available to run heating, ventilation and air conditioning (HVAC) systems.

c) Further information indicating the maximum expected relative humidity in the room in which the sensor electronics will be located under BDB conditions, in which there is no ac power available to run HVAC systems, and whether the sensor electronics is capable of continuously performing its required functions under this expected humidity condition.

d) An analysis of the maximum expected radiological conditions (dose rate and total integrated dose) to which the sensor and associated co-located electronic equipment will be exposed.

Energy Northwest Response

In the following responses, the results were documented, reviewed, and accepted by Energy Northwest personnel in accordance with Energy Northwest procedures and policies.

a) <u>Temperature</u>:

Signal processor and Extended Batteries: Installed in the mild environment (Main Control Room) and vendor testing/analysis qualify the signal processor and associated batteries from -10°C to 55°C.

Probe assembly: The SFP-1 probe assembly is constructed primarily of stainless steel (SS). The dielectric polyether-ether-ketone (PEEK) spacers in the probe body provide temperature, boric acid, and radiation resistance suitable for prolonged exposure to the SFP aqueous environment. Ethylene propylene diene terpolymer (EPDM) seals (0-ring and gasket) are used at the upper part of the repairable head. Qualification of the SFP-1 probe entails demonstrating the elastomers, metals and alloys used are resistant to degradation by the thermal, corrosion, and radiation conditions of the SFP environment.

Coaxial Transmission Cable: The Class 1 E wire and cable meet the requirements of IEEE 383-1974, *IEEE Standard for Type Test of Class 1 E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations*.

Repairable Head: The repairable head is constructed of material also resistant to temperature, corrosion, and radiation. The service life for the SFP-1 repairable head in the SFP environment is bounded by the conditions of the probe design.

<u>Humidity</u>:

Signal processor and Extended Batteries: Based on vendor testing/analysis signal processor and associated batteries are qualified for 5% to 95% relative humidity.

Probe assembly: The testing/analysis probe assembly is qualified up to 100% relative humidity.

<u>Shock & Vibration</u>:

Probe assembly, signal processor and associated batteries provide shock resistance appropriate for general robustness per International Electrotechnical Commission (IEC) 60068-2-27, *Basic Environmental Testing Procedures*, and for vibration resistance appropriate for equipment in large power plants and for general industrial use per IEC 60068-2-6, *Sine Vibration Test*.

Radiation:

The level sensor electronics outside of the spent fuel pool area are required to operate reliably in the mild environmental conditions radiation total integrated dose ≤1E03 rads.

The limiting critical component of the probe is the spacer. Testing/analysis show a cumulative radiation dose up to 2 gigarad (Grad) for EPDM or 10 Grad (100 MGy) for PEEK is assumed for the lowermost spacer located nominally 3 to 4 feet above the fuel rack.

b) GOTHIC temperature analysis for beyond-design-basis external events (BDBEE) was created for the main control room where the sensor electronics will be located.

   The results of this analysis show that the maximum expected temperature for main control room is 120° Fahrenheit (F). The processor continues to functions successfully in conditions up to 131°F at 100% humidity.

c) See the response above.

d) Columbia completed calculation CVI 1201-00,3. Four cases were performed for the northwest (NW) level indicator and for the southeast (SE) level indicator.

   Case 1 included dose rate calculations performed for the operating deck general area

   Case 2a included integrated dose calculations for the NW and SE Level Indicators from the fuel only (no control blade rollers), with the water level in the pool at L1 (normal), for 32 years.

   Case 2b included integrated dose calculations for the NW and SE Level Indicators from the fuel and the CB rollers, at the same water level (L 1 - Normal), for 8 years.

   Case 2c included integrated dose calculations for the NW and SE Level Indicators from the fuel and CB rollers, including skyshine from the scattered radiation in the reactor building, at a water level L3 (top of the fuel handles/fuel racks), for 7 days.

Using the highest dose rate from each case results in the following:

	Dose to NW Level Indicators in Rem	**Dose to SE Level Indicators in Rem**	**Dose to NW (SE) Deck in Rem**
Case 2a	6.50E+08	4.42E-01	1.03E-04 (8.84E-07)
Case 2b	1.63E+08	1.12E-01	2.57E-05 (7.11E-03
Case 2c	9.79E+06	5.86E+06	9.84E+06 (6.13E+06)
Total Integrated Dose	8.22E+08	5.86E+06	9.84E+06 (6.13E+06)

RAI No.7

Please provide information describing the evaluation of the comparative sensor design, the shock test method, test results, and forces applied to the sensor applicable to its successful tests demonstrating that the referenced testing provides an appropriate means to demonstrate reliability of the sensor under the effects of severe shock.

Energy Northwest Response

Probe assembly, signal processor electronics and the external battery enclosure provide shock resistance appropriate for general robustness per IEC 60068-2-27. See the response to RAI No. 4.

RAI No.8

Please provide information describing the evaluation of the comparative sensor design, the vibration test method, test results, and the forces and their frequency ranges and directions applied to the sensor applicable to its successful tests, demonstrating that the referenced testing provides an appropriate means to demonstrate reliability of the sensor under the effects of high vibration.

Energy Northwest Response

The probe assembly was tested separately from the signal processor electronics and the external battery enclosure. All were found to provide vibration resistance appropriate for equipment in large power plants and for general industrial use in accordance with IEC 60068-2-6. During testing a sample probe and a sample signal processor and external battery enclosure were exposed to (10) sine sweeps from 10-55Hz with a constant amplitude of 0.15mm at a rate of 1 octave/minute, repeated in all (3) axes.

RAI No.9

Please provide information describing the evaluation of the comparative display panel ratings against postulated plant conditions. Also, please provide results of the

manufacturer's shock and vibration test methods, test results, and the forces and their frequency ranges and directions applied to the display panel associated with its successful tests.

Energy Northwest Response

The EFP-IL signal processor water level is transmitted to a remote display. The display panel and signal processor are installed in the mild environment (main control room). The testing/analysis qualify the display panel from -10°C to 55°C and 5% to 95% relative humidity.

The display panel provides shock resistance appropriate for general robustness per IEC 60068-2-27. Per IEC 60068-2-27, a sample panel was exposed to (3) half-sine shock pulses of 15g and 11ms, repeated in all (6) directions. No deficiencies were identified. No instrument modification was required.

The display panel provides vibration resistance appropriate for equipment in large power plants and for general industrial use per IEC 60068-2-6. Per IEC 60068-2-6, sample panel was exposed to (10) sine sweeps from 10-55Hz with a constant amplitude of 0.15mm at a rate of 1 octave/minute, repeated in all (3) axes. No deficiencies were identified. No instrument modification was required.

RAI No. 10

Please provide analysis of the seismic testing results and show that the instrument performance reliability, following exposure to simulated seismic conditions representative of the environment anticipated for the SFP structures at Columbia, has been adequately demonstrated.

Energy Northwest Response

Signal processor (electronics) and extended batteries: Testing was conducted to the table limits to envelope Seismic Category 1 safe shutdown earthquake (SSE) conditions using IEEE-344:2004 methodology. The test specimen was monitored for structural integrity and loosening of fasteners; no loss of structural integrity or loose fasteners was noted.

Probe assembly (level sensor): Seismic and hydrodynamic finite element analysis was performed using relevant IEEE-344:2004 methodology (using enveloping Seismic Category 1 SSE conditions or site design basis maximum seismic loads relative to the location where the equipment is mounted). The sloshing analysis was based on GOTHIC, an industry-standard computer code for performing multiphase fluid flow. ANSYS, a finite element analysis computer code, was used to perform the hydrodynamic loading and structural analysis. A code-to-code verification was performed between ANSYS and GOTHIC with good results.

Based on this report, the level probe assembly meets IEEE 344:2004 requirements for adequacy of seismic design and installation with attention to seismic and hydrodynamic effects. The seismic qualification on the basis of this report is predicted on a seismic event bounded by the 5.384g.

RAI No. 11

Please provide the NRC staff with the final configuration of the power supply source for each channel so that the staff may conclude that the two channels are independent from a power supply assignment perspective.

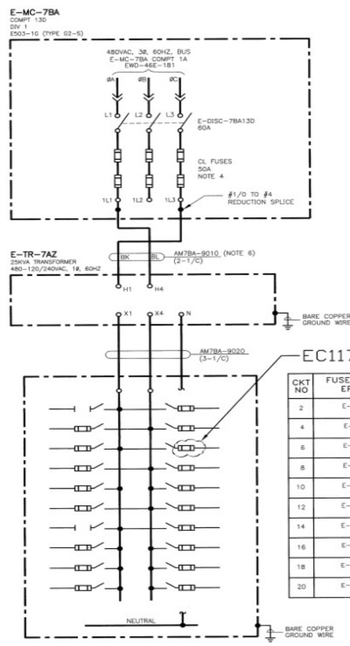
Energy Northwest Response

Each SFP instrument channel is normally powered from a 120/240-V ac 60 Hz plant distribution panel to support continuous monitoring of SFP level. The primary channel will receive power from a different 480V bus than the backup channel. Therefore, loss of any one 480-V ac bus does not result in loss of normal 120-V ac power for both instrument channels.

On loss of normal 120-V ac power, each channel's UPS automatically transfers to a dedicated backup battery. If normal power is restored, the channel will automatically transfer back to the normal AC power.

The drawings identifying the power supply source for each channel are attached below.

DRAWING	CABLE NO	SERVICE	FUSE DISCONNECT EPN / SIZE	CKT NO
		SPARE	E-DISC-PP7AZ/1	1
EWD-48I-004, EWD-46E-280	AP7AZ-9003	MULTI-SYSTEM CONTROL (E-CP-H13/P820 BOARD B)	E-DISC-PP7AZ/3 20A	3
EWD-78E-009, EWD-3E-125	AP7AZ-0005	CAS-DY-3A & 4A INSTRUMENTATION RRC-P-1A POWER MONITORING	E-DISC-PP7AZ/5 10A	5
EWD-46E-242C, EWD-46E-242D	AP7AZ-9007	SUPERVISORY SYSTEM (E-CP-CSA BAY EA1)	E-DISC-PP7AZ/7 20A	7
EWD-1E-011, EWD-1E-012, EWD-1E-013,	AP7AZ-9009	MULTI-SYSTEM CONTROL (E-CP-H13/P612)	E-DISC-PP7AZ/9 20A	9
EWD-46E-242J, EWD-46E-242K	AP7AZ-9011	SUPERVISORY SYSTEM (E-CP-CSA BAY CR1A)	E-DISC-PP7AZ/11 20A	11
		SPARE	E-DISC-PP7AZ/13	13
EWD-40E-052, EWD-40E-053, EWD-40E-054, EWD-40E-057, EWD-40E-059, EWD-66E-005, EWD-66E-006	AP7AZ-9015	OFF-GAS SYSTEM (E-CP-H13/P672)	E-DISC-PP7AZ/15 20A	15
EWD-46E-014, EWD-46E-015, EWD-46E-016, EWD-46E-024, EWD-46E-045, EWD-50E-028A, EWD-50E-024A	AP7AZ-9017	BATT E-B1-1 & E-B2-1 GND DETECTION GENERATOR INSTRUMENTATION & METERING (E-CP-H13/P800 BOARD C)	E-DISC-PP7AZ/17 20A	17
E536-3B	AP7AZ-9019	CABLE SPREAD ROOM BAROMETER PL-PT-1	E-DISC-PP7AZ/19 20A	19



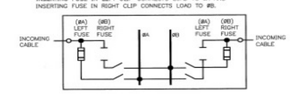
EC11797

CKT NO	FUSE DISCONNECT EPN / SIZE	SERVICE	CABLE NO	DRAWING
2	E-DISC-PP7AZ/2 20A	RRC DIV. 1 ADJUSTABLE SPEED DRIVE CONTROL (E-CP-H13/P634)	AP7AZ-9002	EWD-3E-140
4	E-DISC-PP7AZ/4 20A	MULTI-SYSTEM CONTROL (E-CP-H13/P601, P603, P621, P632)	AP7AZ-9004	EWD-46E-242B
6	E-DISC-PP7AZ/6 15A	SPENT FUEL POOL LIQUID LEVEL INDICATOR FPC-LIT-21A	AP7AZ-9006	EWD-38I-003A
8	E-DISC-PP7AZ/8 20A	MULTI-SYSTEM CONTROL POWER (E-CP-H13/P840 BOARD A)	AP7AZ-9008	EWD-58I-002, EWD-69I-013, EWD-72E-035, EWD-72E-041, EWD-69I-017
10	E-DISC-PP7AZ/10 20A	SUPERVISORY SYSTEM (E-CP-CSA BAY CWA)	AP7AZ-9010	EWD-46E-242G, EWD-46E-242H
12	E-DISC-PP7AZ/12 20A	SUPERVISORY SYSTEM (E-CP-CSA BAY EA2)	AP7AZ-9012	EWD-46E-242E, EWD-46E-242F
14	E-DISC-PP7AZ/14 20A	START-UP AUXILIARY XFMR E-TR-S INSTRUMENTATION (E-CP-H13/P842 BOARD F)	AP7AZ-9014	EWD-46E-021
16	E-DISC-PP7AZ/16 20A	OFF-GAS SYSTEM (E-CP-H13/P672)	AP7AZ-9016	EWD-40E-046, 02N64-05,1,3, EWD-40E-013, 038, 039, 040, 041, 043, 044
18	E-DISC-PP7AZ/18 20A	CW & TMU INDICATION CKTS (E-CP-H13/P821 & E-CP-H13/P824)	AP7AZ-9018	EWD-46E-242A
20	E-DISC-PP7AZ/20 20A	PROCESS RADIATION MONITORING SYSTEM (BD-RAD-24)	AP7AZ-9020	EWD-36E-019, EWD-36E-022, EWD-36E-023

EC11797

NOTES:

1. FOR GENERAL NOTES SEE EWD-46E-000.
2. THE INFORMATION CONTAINED ON THIS EWD SUPERSEDES THE INFORMATION OBTAINED FROM THE INDICATED REFERENCE DRAWING.
3. CONNECTION INSTALLATION DETAILS WITHIN THE ENCLOSURE(S) AS NOTED SHALL BE IN ACCORDANCE WITH PLANT SPECIFICATIONS 16A, 16B, 50A.3 AND THE FOLLOWING:
   a. THE QUALITY CLASS OF ELECTRICAL INSTALLATION WORK SHALL BE THE SAME AS IDENTIFIED IN THE MASTER EQUIPMENT LIST (MEL).
   b. ALL CABLES AND ASSOCIATED WIRING INTERNAL TO THE ENCLOSURES SHALL BE COMPATIBLE WITH THE DIVISION IDENTIFIED ON THE ENCLOSURE (ON THIS EWD) AND IN ACCORDANCE WITH THE DETAILS ON DRAWING E537-1A.
   c. ALL SPARE CONDUCTORS IN THE ENCLOSURE(S) SHALL BE CUT NO SHORTER THAN THE LONGEST TERMINATED CONDUCTOR IN THE CABLE, TAPED AND IDENTIFIED WITH THE SAME CABLE IDENTIFICATION TAG NUMBER AS APPEARS ON THE ORIGINAL JACKET IF VISUAL TRACEABILITY TO THE ORIGINAL CABLE JACKET IS IMPOSSIBLE OR OBSCURED.
4. FUSE DISCONNECT E-DISC-7BA13D WITH DOUBLE POWER FUSES PROVIDES ELECTRICAL SEPARATION BETWEEN SAFETY RELATED 480VAC SOURCE E-MC-7BA AND DOWNSTREAM NON-SAFETY RELATED CIRCUITS
5. EACH FUSE DISCONNECT CONTAINS A LEFT AND RIGHT FUSE CLIP. INSERTING FUSE IN LEFT CLIP CONNECTS LOAD TO ØA AND INSERTING FUSE IN RIGHT CLIP CONNECTS LOAD TO ØB.



6. E-PP-7AZ FEEDER CABLE AM7BA-9010 IS TERMINATED IN A NON-STANDARD CONFIGURATION. THIS WAS THE AS-FOUND TERMINATION. THE REPLACEMENT "SPECTRUM" DISCONNECT (TYPE G2-S) WAS INSTALLED AND TERMINATED WITH THE SAME NON-STANDARD CONFIGURATION IN R21 OUTAGE.

E-PP-7AZ (NOTE 3)
POWER PANEL

SERVICE: 120/240VAC, 1Ø, 3W, 60HZ
BUS RATING: 100A
DIVISION: A
PANEL TYPE: QMB

BOX DIMENSION		
HEIGHT	WIDTH	DEPTH
56	31	10 5/8

MOUNTING: SURFACE

GUTTERS			
TOP	BOTTOM	RIGHT	LEFT
5.5	5.5	6	6

EQUIPMENT GROUND BAR: X
MAINS: TOP
CONDUCTOR MAX SIZE: 300 MCM

EC11797

SPARED CABLES COILED IN PANEL		
CABLE NO	DESTINATION	REF DWG
~~AP7AZ-9013~~	~~E-CP-CSA (BAY MISC)~~	~~E542-7~~
AP7AZ-9001	E-CP-H13/P682 (BAY F)	E537-5C
AP7AZ-9024	E-CP-H13/P984 (BAY B)	E537-25C
AP7AZ-9023	E-CP-H13/P688 (BAY B)	E537-11C

	REF DWG	DESCRIPTION
	E503-7	ONE LINE DIAG (E-MC-7BA)
(NOTE 2)	E508-1	POWER PANEL SCHED (E-PP-7AZ)
	E528-35A	MCC OVERLOAD SUMMARY (E-MC-7BA)
	E535-1	CONN WIRING DIAG (MCC INDEX)
	E535-43A	CONN WIRING DIAG (E-MC-7BA)
(NOTE 2)	218-03,5299	PANELBOARD SCHEDULE (E-PP-7AZ)

DESIGN DOCUMENT CHANGE	
EC Number: 11797	Page No: 558
Doc Number: EWD-46E-242	Rev.: 25
Zone: E/3-6; C/1-2	
Prepared by: K.S. HALE	Date: 4-4-14
*Technical Eng. Checked by: [signature]	Date: 4/15/14
Physical Design Checked by: [signature]	Date: 4-15-14
* Required for Functional Design Documents (e.g. Flow Diag., One Line Diag's and Elementary Diag's, etc...) Ref: DES-5-5. If not required, indicate N/R	

26145H R2

SAFETY RELATED (NOTE 4)

REV	DATE	DESCRIPTION	DWN	CHK	APVD
21	4-1-08	REDRAWN AND REVISED PER 6733	JMF	GRP	GHM
22	5/22/08	REVISED PER 6735 (B/1-2, D/1, D/9)	AJE	GRP	GHM
23	5/23/13	REVISED PER 11486-336 (H/6).	HL	KB	DGW
24	7-8-13	REVISED PER 12056 (A/2; D/6; F-G/1-2; G-H/6-7).	HL	GHM	GHM
25	9-24-13	REVISED PER 12292 (E/6, F/1-2, F/5)	KWB	GHM	GHM

ENERGY NORTHWEST — COLUMBIA GENERATING STATION

SIGNATURE		DATE
APPROVED	GERRY MCHENRY	4-1-08
CHECKED	GERRY PRICE	4-1-08
DRAWN	JASON FITCH	12/12/07

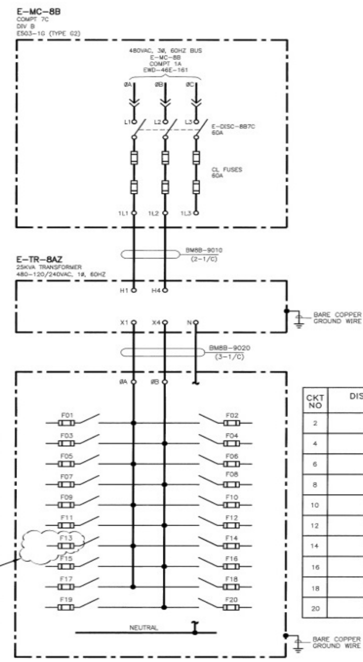
TITLE: ELECTRICAL WIRING DIAGRAM
AC ELECTRICAL DISTRIBUTION SYSTEM
POWER PANEL E-PP-7AZ

DWG NO. EWD-46E-242 REV. 25

SERVER84\ELECT\046\046E242.DWG

TOP TIER

ESD-2



DRAWING	CABLE NO	SERVICE	DISCONNECT SWITCH EPN / SIZE	CKT NO
EWD-46E-254B, EWD-46E-254C	BP8AZ-9001	MULTI-SYSTEM INSTRUMENTATION (E-CP-H13/P601, P602, P603, P613, P618, P619)	E-DISC-PP8AZ/1 20A	1
EWD-46E-254A	BP8AZ-9003	MULTI-SYSTEM INSTRUMENTATION (E-CP-H13/P602 & P603)	E-DISC-PP8AZ/3 20A	3
EWD-78E-009, EWD-3E-127	BP8AZ-5	CAS-DY-3B & 4B INSTRUMENTATION RRC-P-1B POWER MONITORING	E-DISC-PP8AZ/5 10A	5
EWD-50E-026A, EWD-50E-029A	BP8AZ-9027	125VDC BATTERIES E-B1-2 & E-B1-7 GROUND DETECTION (E-CP-H13/P800)	E-DISC-PP8AZ/7 20A	7
EWD-48E-033	BP8AZ-9009	JUNCTION BOX 'V'	E-DISC-PP8AZ/9 20A	9
EWD-46E-044	BP8AZ-9011	BACK-UP AUXILIARY XFMR E-TR-B INSTRUMENTATION (BOARD F)	E-DISC-PP8AZ/11 20A	11
EWD-38I-003A	BP8AZ-9014	SPENT FUEL POOL LIQUID LEVEL INDICATOR FPC-LIT-21B	E-DISC-PP8AZ/13 15A	13
EWD-46E-254D	BP8AZ-9015	MULTI-SYSTEM INSTRUMENTATION (E-CP-H13/P600, P614, P673)	E-DISC-PP8AZ/15 20A	15
E537-1VB-2	BP8AZ-9017	TG SYSTEM INSTRUMENTATION (E-CP-H13/P820 BOARD B)	E-DISC-PP8AZ/17 20A	17
EWD-66E-025	BP8AZ-9019	CONDENSER AIR REMOVAL SYSTEM CONTROL (E-CP-RC/B)	E-DISC-PP8AZ/19 20A	19

EC11797

CKT NO	DISCONNECT SWITCH EPN / SIZE	SERVICE	CABLE NO	DRAWING
2	E-DISC-PP8AZ/2 20A	TIP SYSTEM CONTROL (E-CP-H13/P607)	BP8AZ-9002	02C51-05,21,1 (EWD-17E-027)
4	E-DISC-PP8AZ/4 20A	LEAK DETECTION SYSTEM CONTROL (E-CP-H13/P642)	BP8AZ-9004	EWD-19E-001
6	E-DISC-PP8AZ/6 20A	SUPERVISORY SYSTEM E-CP-CSB BAY CR1B	BP8AZ-9006	EWD-46E-254J
8	E-DISC-PP8AZ/8 20A	SUPERVISORY SYSTEM E-CP-CSB BAY EB1	BP8AZ-9008	EWD-46E-254E
10	E-DISC-PP8AZ/10 20A	SUPERVISORY SYSTEM E-CP-CSB BAY CWB	BP8AZ-9010	EWD-46E-254G, EWD-46E-254A
12	E-DISC-PP8AZ/12 20A	JUNCTION BOX 'V'	BP8AZ-9112	EWD-48E-032
14	E-DISC-PP8AZ/14 20A	MET TWR SUPV DIV 2 TROUBLE CKT E-CP-CSB BAY MISC/MET	BP8AZ-9013	EWD-46E-248R
16	E-DISC-PP8AZ/16 20A	CW & TMU INDIC CKTS (E-CP-H13/P824)	BP8AZ-9016	EWD-46E-254A
18	E-DISC-PP8AZ/18 20A	COND AND RFW CONTROL SYSTEMS (E-CP-H13/P840 BOARD A)	BP8AZ-9018	EWD-72E-036, EWD-72E-042, EWD-69I-017
20	E-DISC-PP8AZ/20 30A	MULTI-SYSTEM INSTRUMENTATION (E-CP-H13/P602)	BP8AZ-9020	EWD-16E-004, EWD-16E-003, EWD-13E-010, EWD-3E-125A, EWD-3E-127A, EWD-3E-155,

NOTES:

1. FOR GENERAL NOTES SEE EWD-46E-000.
2. THE INFORMATION CONTAINED ON THIS EWD SUPERSEDES THE INFORMATION OBTAINED FROM THE INDICATED REFERENCE DRAWING.
3. CONNECTION INSTALLATION DETAILS WITHIN THE ENCLOSURE(S) AS NOTED SHALL BE IN ACCORDANCE WITH PLANT SPECIFICATIONS 16A, 16B, 50A.3 AND THE FOLLOWING:
   a. THE QUALITY CLASS OF ELECTRICAL INSTALLATION WORK SHALL BE THE SAME AS IDENTIFIED IN THE MASTER EQUIPMENT LIST (MEL).
   b. ALL CABLES AND ASSOCIATED WIRING INTERNAL TO THE ENCLOSURES SHALL BE COMPATIBLE WITH THE DIVISION IDENTIFIED ON THE ENCLOSURE (ON THIS EWD) AND IN ACCORDANCE WITH THE DETAILS ON DRAWING E537-1A.
   c. ALL SPARE CONDUCTORS IN THE ENCLOSURE(S) SHALL BE CUT NO SHORTER THAN THE LONGEST TERMINATED CONDUCTOR IN THE CABLE, TAPED AND IDENTIFIED WITH THE SAME CABLE IDENTIFICATION TAG NUMBER AS APPEARS ON THE ORIGINAL JACKET IF VISUAL TRACEABILITY TO THE ORIGINAL CABLE JACKET IS IMPOSSIBLE OR OBSCURED.

EC11797

E-PP-8AZ (NOTE 3)
POWER PANEL

SERVICE: 120/240VAC, 1Ø, 3W, 60HZ
BUS RATING: 100A
DIVISION: B
PANEL TYPE: QMB

BOX DIMENSION		
HEIGHT	WIDTH	DEPTH
56	31	10 5/8

MOUNTING: SURFACE

GUTTERS			
TOP	BOTTOM	RIGHT	LEFT
5 1/2	5 1/2	6	6

EQUIPMENT GROUND BAR: X
MAINS: TOP
CONDUCTOR MAX SIZE: 300MCM

EC11797

SPARED CABLES COILED IN PANEL		
CABLE NO	DESTINATION	REF DWG
BP8AZ-9012	E-CP-CSB	E542-5T
2P8A-9	E-PP-8A COILED IN	EWD-46E-317

	REF DWG	DESCRIPTION
	E503-8	ONE LINE DIAGRAM (E-MC-8B)
(NOTE 2)	E508-1	AC POWER PANEL DIAGRAM (E-PP-8AZ)
	E528-35	MCC OVERLOAD SUMMARY (E-MC-8B)
	E535-1	CONN WIRING DIAG (INDEX)
	E535-54A	CONN WIRING DIAG (E-MC-8B)
(NOTE 2)	218-03, 5299	PANEL BOARD SCHEDULE (E-PP-8AZ)

DESIGN DOCUMENT CHANGE	
EC Number: 11797	Page No: 559
Doc Number: EWD-46E-254	Rev.: 31
Zone: C/1-2; D/7-10; F-G/1-2	
Prepared by: K.S. HALE	Date: 4-4-14
*Technical Eng. Checked by: [signature]	Date: 4/15/14
Physical Design Checked by: [signature]	Date: 4-15-14
* Required for Functional Design Documents (e.g. Flow Diag., One Line Diag's and Elementary Diag's, etc...) Ref: DES-5-5. If not required, indicate N/R	

26145H R2

REV	DATE	DESCRIPTION	DWN	CHK	APVD
27	4-1-08	REDRAWN AND REVISED 6733	LRC	GRP	GHM
28	5/22/08	REVISED PER 6735 (B/1-2, D/1)	AJE	GRP	GHM
29	7/2/09	REV PER 7906-308 (D/4)	KB	HL	LJK
30	10-23-09	REV PER 8595 (D/8, D/10)	LJK	BLS	LJK
31	11-12-12	REV PER 8504 (E/3)	RBW	GHM	GHM

ENERGY NORTHWEST
COLUMBIA GENERATING STATION

	SIGNATURE	DATE
APPROVED	GERRY MCHENRY	4-1-08
CHECKED	GERRY PRICE	4-1-08
DRAWN	LEIGH R. CONLEY	12/17/07

TITLE: ELECTRICAL WIRING DIAGRAM
AC ELECTRICAL DISTRIBUTION SYSTEM
POWER PANEL E-PP-8AZ

DWG NO. EWD-46E-254 REV. 31

SERVER84\ELECT\046\046E254.DWG

ESD-2

TOP TIER

RAI No. 12

Please provide the following:

a) A description of the electrical ac power sources and capacities for the primary. Please provide the results of the calculation depicting the battery backup duty cycle requirements demonstrating that its capacity is sufficient to maintain the level indication function until offsite resource availability is reasonably assured.

Energy Northwest Response

a) As shown in the response to RAI No. 11, each instrument channel is supplied by separate 120-V ac power, through a UPS that automatically transfers to a dedicated backup battery.

b) The backup-power battery packs were tested to full discharge at several discharge rates to determine the battery capacity. The test data shows that when the system instrument was configured to operate in minimum power mode with sample rate of 15 samples per hour at room temperature, the battery capacity had 82 percent remaining after 17.8 days of operation. The backup-power source can provide at least 7-day battery life with minimum power mode using an average sample rate of 15 samples per hour. Based on test results, it was determined that the SFPI's replaceable batteries used for instrument channel power have sufficient capacity to maintain the level indication function for longer than 7 days.

RAI No. 13

Please provide the following:

a) An estimate of the final expected instrument channel accuracy performance (e.g., in percent of span) under both a) normal SFP level conditions (approximately Level 1 or higher) and b) at the BDB conditions (i.e., radiation, temperature, humidity, post-seismic and post-shock conditions) that would be present if the SFP level were at the Level 2 and Level 3 datum points.

b) A description of the methodology that will be used for determining the maximum allowed deviation from the instrument channel design accuracy that will be employed under normal operating conditions as an acceptance criterion for a calibration procedure to flag to operators and to technicians that the channel requires adjustment to within the normal condition design accuracy.

Energy Northwest Response

a) Accuracy: The absolute system accuracy exceeds the published vendor measurement accuracy of ±3 inches. This accuracy is applicable for normal conditions and also the temperature, humidity, chemistry, radiation levels, post-seismic and post-shock conditions expected for BDBE event conditions. This has been verified by testing.

   The instrument channel level accuracy is expected to be better than ±3.0 inches for all expected conditions. The expected instrument channel accuracy performance would be approximately ±1% of span.

b) In general relative to normal operating conditions, any applicable calibration procedure tolerances or acceptance criterion have been established based on manufacturer's stated/recommended accuracy. Both SFP primary and backup redundant sensor electronics require periodic calibration verification to check that the channel's measurement performance is within the specified tolerance (±3 inches). If the difference is larger than the allowable tolerance during the verification process, a calibration adjustment will be required.

   Instrument accuracy and performance are not affected by restoration of power or restarting the processor.

RAI No. 14

Please provide an analysis verifying that the proposed instrument performance is consistent with these estimated normal and BDB accuracy values. Please demonstrate that the channels will retain these accuracy performance values following a loss of power and subsequent restoration of power.

Energy Northwest Response

The level instrument automatically monitors the integrity of its level measurement system using in-situ capability. Deviation of measured test parameters from manufactured or as-installed configuration beyond a configurable threshold prompts operator intervention.

Each instrument electronically logs a record of measurement values over time in nonvolatile memory that is compared to demonstrate constancy, including any changes in pool level, such as that associated with the normal evaporative loss/refilling cycle. The channel level measurements can be directly compared to each other [i.e., regular cross channel comparisons]. The two displays are installed in close proximity to each other, thus simplifying channel comparison.

Diagnostics: The system performs and displays the results of real-time information related to the integrity of the cable, probe, and instrument channel.

Instrument accuracy and performance are not affected by restoration of power or restarting the processor. Test results indicate that no deficits were identified with respect to maintenance of reliable function, accuracy, or calibration as a result of power interruption. The SFPI system's accuracy was maintained without recalibration following the power interruption.

RAI No. 15

Please provide the following:

a) A description of the capability and provisions the proposed level sensing equipment will have to enable periodic testing and calibration, including how this capability enables the equipment to be tested in-situ.

b) A description of how such testing and calibration will enable the conduct of regular channel checks of each independent channel against the other, and against any other permanently-installed SFP level instrumentation.

c) A description of how functional checks will be performed, and the frequency at which they will be conducted. Describe how calibration tests will be performed, and the frequency at which they will be conducted. Provide a discussion as to how these surveillances will be incorporated into the plant surveillance program.

d) A description of what preventive maintenance tasks are required to be performed during normal operation, and the planned maximum surveillance interval that is necessary to ensure that the channels are fully conditioned to accurately and reliably perform their functions when needed.

Energy Northwest Response

a) The EFP-IL signal processor technical manual and the EFP-IL signal processor operator's manual provide the following information on the available calibration and diagnostic routines, and required maintenance.

   The calibration menu provides submenus to export and import the calibration files, correct liquid level measurement error, and turn liquid level railing on and off.

The diagnostics menu contains several submenus with diagnostic routines and system status for use upon receipt, installation, and for periodic maintenance and surveillance.

Six Month Maintenance Interval

- Memory Test
- Battery Test
- Temperature Compensation Test
- Scan Test
- Export Logs

Two Year Maintenance Interval

- EFP-BAT Battery Pack Replacement
- Memory Card Replacement
- Probe and Transmission Cable Health Checks
- Clock Battery Verification and Clock Calibration

b) A channel check is not a specified requirement in NEI 12-02. A channel check is specified in IEEE 338-1987, *Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems*. SFP level instrument channels are not safety related and are not subject to testing requirements of safety related instrumentation. If the plant staff determined a need to confirm that the two channels are performing as expected, the two channels may be read in the main control room. While the SFP is operating within design basis and at normal level, the indicators may be compared to fixed marks within the SFP by visual observation to confirm indicated level.

c) Functional checks are automated and/or semi-automated (requiring limited operator or technician interaction) and are performed through the instrument menu software and initiated by the operator or technician. There are a number of other internal system tests that are performed by system software on an essentially continuous basis without user intervention but can also be performed on an on-demand basis with diagnostic output to the display for the operator or technician to review. Other tests such as menu button tests, level alarm, and alarm relay tests are only initiated manually by the operator or technician.

Formal calibration checks are recommended by the vendor on a two-year interval to demonstrate calibration to external NIST-traceable standards. NEI 12-02

requires the periodic calibration verification will be performed within 60 days of a planned refueling outage considering normal testing scheduling allowances (e.g., 25%). Columbia is on a two-year refueling cycle, therefore, calibration will be scheduled to meet the NEI guidance without jeopardizing vendor recommendations.

Plant Procedures

Both the functional test procedure and the calibration procedure for the SFPLI system include precautions and limitations on the time the primary or backup instrumentation can be out of service for testing, maintenance and/or calibration. This time is restricted to 90 days. If the instrument channel is not expected to be restored compensatory actions are required. If both channels become non-functioning, then within 24 hours action is required to be initiated to restore at least one channel and to implement compensatory action within 72 hours. Compensatory measures can be the use of alternate suitable equipment or supplemental personnel.

The flowing tests are performed as part of the functional testing on a 6 month frequency:

1. Memory Test
2. Battery test
3. Temperature Compensation Test
4. Scan Test
5. Export Logs

The SFPLI calibration is performed on a 2 year frequency and within 60 days of a planned refueling outage which includes:

1. Export Logs
2. Battery Replacement
3. Probe and Transmission Cable Health Checks
4. Memory Card Replacement (if required) and Calibration
5. Time Check

d) See the response to item c above.

RAI No. 16

Please provide a list of the procedures addressing operation (both normal and abnormal response), calibration, test, maintenance, and inspection procedures that will be developed for use of the spent SFP instrumentation. The licensee is requested to include a brief description of the specific technical objectives to be achieved within each procedure.

Energy Northwest Response

The following procedures have been developed/revised to include information required for the operation (both normal and abnormal response), calibration, test, maintenance, and inspection of the new SFP instrumentation. The purpose/scope of each procedure has been provided.

4.626.FPC1, 626.FPC1 Annunciator Panel Alarms

Provides the actions to take to verify when alarms are received on annunciator panel 626.FPC1.

ABN-FPC-LOSS, Loss of Fuel Pool Cooling

Provides the actions to be taken on an unplanned loss of cooling to the Fuel Pool, an unplanned reduction in Fuel Pool level, or activation of the Skimmer Surge Tank-A(B) - Level Low/Low.

SOP-ELEC-AC-LU, AC Electrical Distribution System Breaker Lineup

Provides instructions for 6900, 4160, 480, and 120 Volt AC Electrical Distribution breaker lineup.

SOP-FPC-LEVEL-OPS, Spent Fuel Pool Level Monitor Operations

Provides instructions for operating the MOHR spent fuel pool level monitors.

SOP-FPC-START, Fuel Pool Cooling Start

Provides instructions for Fuel Pool Cooling and Cleanup System startup.

PPM 10.27.113, Spent Fuel Pool Level Indication Channel 1 - CFT

Provides channel functional test instructions for Channel 1 of the Spent Fuel Pool Level Indication System.

PPM 10.27.114, Spent Fuel Pool Level Indication Channel 1 - CC

Provides channel calibration instructions for Channel 1 of the Spent Fuel Pool Level Indication System.

PPM 10.27.116, Spent Fuel Pool Level Indication Channel 2 - CC

Provides channel functional test instructions for Channel 2 of the Spent Fuel Pool Level Indication System.

PPM 10.27.117, Spent Fuel Pool Level Indication Channel 2 - CFT

Provides channel calibration instructions for Channel 2 of the Spent Fuel Pool Level Indication System.

PPM 3.1.10, Operating Data and Logs

Provides instructions to assure that important events of plant operations are adequately recorded and the records are prepared, reviewed, and maintained in a meaningful manner.

RAI No. 17

Please provide the following:

a) Further information describing the maintenance and testing program the licensee will establish and implement to ensure that regular testing and calibration is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements. Please include a description of your plans for ensuring that necessary channel checks, functional tests, periodic calibration, and maintenance will be conducted for the level measurement system and its supporting equipment.

b) A description of how the guidance in NEI 12-02 section 4.3 regarding compensatory actions for one or both non-functioning channels will be addressed.

c) A description of what compensatory actions are planned in the event that one of the instrument channels cannot be restored to functional status within 90 days.

<u>Energy Northwest Response</u>

a) See the responses in RAI No.15

   SFPI maintenance and testing program requirements ensure design and system readiness. They are planned and are developed in accordance with plant processes and procedures and consider vendor recommendations to ensure that appropriate regular testing, functional tests, periodic calibration, and maintenance is performed.

Functional checks are automated and/or semi-automated and are performed through the instrument menu software and initiated by the operator. There are a number of other internal system tests that are performed by system software on an essentially continuous basis without user intervention, but can also be performed on an on-demand basis with diagnostic output to the display for the operator to review. Functional checks are described in detail in the vendor manual, and the applicable information is contained in plant procedures and preventive maintenance tasks. Functional checks are performed on the EFP-IL every 6 months as recommended by the vendor.

Channel calibration tests per maintenance procedures with limits established in consideration of vendor equipment specifications are performed at frequencies established in consideration of vendor recommendations.

b) See the response to RAI No. 15c above.

c) See the response to RAI No. 15c above.

For a single channel that is not expected to be restored, or is not restored within 90 days, the compensatory actions will include steps necessary to verify by administrative means the remaining channel is functional and include periodic direct visual monitoring of spent fuel pool level.

RAI No. 18

Please provide a description of the in-situ calibration process at the SFP location that will result in the channel calibration being maintained at its design accuracy.

Energy Northwest Response

The probe itself is a perforated tubular coaxial waveguide with defined geometry and is not calibrated. Calibration is performed using PPM 10.27.114 for Channel 1 and PPM 10.27.116 for Channel 2

The instrument automatically monitors the integrity of its level measurement system using in-situ capability. Deviation of measured test parameters from manufactured or as-installed configuration beyond a configurable threshold prompts operator intervention.

Vendor documentation has been used to develop provide the testing and calibration procedures for the SFPI. The SFPI can be calibrated in-situ without removal from its installed location. The system is calibrated using a CT-100 device and processing of vendor scanned files.

References:

1. CVI 1217-00,1,1 R0 MOHR Test and Measurement LLC EFP-IL SFPI System Manual
2. CVI 1217-00,1,2 R0 MOHR Test and Measurement, LLC EFP-IL SFPLI System Reports/FMEA
3. CVI 1201-00,1 R1 GOTHIC Analysis of CGS Radwaste Building Response to SBO
4. NE-02-13-04 R0 Cycle 22 Spent Fuel Pool Time-to-200°F
5. NE-02-17-02 R0 Cycle 24 SFP Time-to 200°F
6. CVI 1201-00, 3 Revision 0, EN-CGS Spent Fuel Pool Doses for New Level Instrument.
7. Letter from C. F. Lyon (NRC) to M. E. Reddemann (Energy Northwest), "Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan for Implementation of Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," dated November 7, 2013 (ML13302C136)

COLUMBIA GENERATING STATION

FINAL INTEGRATED PLAN FOR THE MITIGATION OF BEYOND-DESIGN-BASIS EXTERNAL EVENTS

Table of Contents

Final Integrated Plan (FIP)

1.0 BACKGROUND

1.1 Site Specific Location

Columbia Generating Station (Columbia) is located in the southeast area of the U.S. Department of Energy's (DOE) Hanford Site in Benton County, Washington. The site is approximately 3 miles west of the Columbia River at River Mile (RM) 352, approximately 10 miles north of North Richland, 18 miles northwest of Pasco, and 21 miles northwest of Kennewick. The reactor is located at 46° 28' 18" north latitude and 119° 19' 58" west longitude. The approximate Universal Transverse Mercator coordinates are 5,148,840 meters north and 320,930 meters east.

The reactor floor elevation of 441 ft. msl is 68 ft. above the water level estimated for the largest historical flood (approximately 373 ft. msl). There is no record of flooding in the immediate site area. The plant safety-related structures are located above high water elevations associated with Columbia River flooding, intense local precipitation, and upriver dam failures.

The design-basis groundwater elevation used for subsurface hydrostatic loadings is 420 ft. msl and was predicated on the possible future construction of Ben Franklin Dam at RM 348. Planning for the dam has been terminated. However, the water table beneath Columbia would rise to less than 405 ft. msl if the dam were to be completed. The actual water table beneath the project is about 385 ft. msl.

1.2 Response

In 2011, an earthquake-induced tsunami caused beyond-design-basis (BDB) flooding at the Fukushima Dai-ichi Nuclear Power Station in Japan. The flooding caused the emergency power supplies and electrical distribution systems to be inoperable, resulting in an extended loss of AC power (ELAP) in five of the six units on the site. The ELAP led to (1) the loss of core cooling, (2) the loss of spent fuel pool cooling capabilities, and (3) the inability to maintain containment integrity. All direct current (DC) power was lost early in the event on Units 1 and 2 and after some period of time at the other units. Core damage occurred in three of the units along with a loss of containment integrity resulting in a release of radioactive material to the surrounding environment.

The Nuclear Regulatory Commission (NRC) assembled a Near-Term Task Force (NTTF) to advise the Commission on actions the United States nuclear industry should take to preclude core damage and a release of radioactive material after a natural disaster such as that seen at Fukushima. The NTTF report contained

many recommendations to fulfill this charter, including assessing extreme external event hazards and strengthening station capabilities for responding to BDB external events (BDBEE).

Based on NTTF Recommendation 4.2 (Reference 1), the NRC issued Order EA-12-049 (Reference 2) on March 12, 2012, to implement mitigating strategies for BDBEEs. The order provided the following requirements for strategies to mitigate BDBEEs:

1) Licensees shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment integrity, and spent fuel pool (SFP) cooling capabilities following a BDBEE.

2) These strategies must be capable of mitigating a simultaneous ELAP and the loss of the ultimate heat sink (LUHS) and have adequate capacity to address challenges to core cooling, containment integrity and SFP cooling capabilities at all units on a site subject to the Order.

3) Licensees must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment integrity, and SFP cooling capabilities at all units on a site subject to the NRC Order.

4) Licensees must be capable of implementing the strategies in all modes.

5) Full compliance shall include procedures, guidance, training, and acquisition, staging or installing of equipment needed for the strategies.

   The order specifies a three-phase approach for strategies to mitigate BDBEEs:

   - Phase 1 Initially cope relying on installed equipment and on-site resources.
   - Phase 2 Transition from installed plant equipment to on-site diverse and flexible coping strategies (FLEX) equipment.
   - Phase 3 Obtain additional capability and redundancy from off-site equipment and resources until power, water, and coolant injection systems are restored or commissioned.

7) Submit an overall integrated plan (OIP), including a description of how compliance with the requirements would be achieved.

8) Complete implementation of the requirements no later than two refueling cycles after submittal of the OIP or December 31, 2016, whichever comes first.

The Nuclear Energy Institute (NEI) developed NEI 12-06, *Diverse and Flexible Coping Strategies (FLEX) Implementation Guide* (Reference 3) which provides guidelines for nuclear stations to assess extreme external hazards and implement the mitigation strategies specified in NRC Order EA-12-049. The NRC issued Japan Lessons-Learned Project Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, *Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events*, dated August 29, 2012, (Reference 4) which endorsed Revision 0 of NEI 12-06 with clarifications on determining baseline coping capability and equipment quality.

In December 2015, NEI issued Revision 2 of NEI 12-06. On January 22, 2016, the NRC issued Revision 1 to JLD-ISG-2012-01. The revised ISG endorses, with exceptions, additions, and clarifications, the methodologies described in the industry guidance document, NEI 12-06, Revision 2. Columbia's Final Integrated Plan (FIP) complies with Revision 2 of NEI 12-06 as it applies to Reference 2 with the exception of Appendix E which was finalized after the validation process was completed. Other aspects of NEI 12-06, Revision 2, while not applicable to this Order compliance, will be utilized for upcoming submittals (e.g., mitigating strategies assessment for the reevaluated hazards using Appendix G and Appendix H) and rulemaking (e.g., references to NEI 13-06 and NEI 14-01).

2.0 PERFORMANCE ATTRIBUTES

This baseline coping capability is built upon strategies that focus on a simultaneous ELAP and LUHS condition caused by unspecified events. The baseline assumptions have been established on the presumption that other than the loss of the alternating current (AC) power sources and normal access to the UHS, plant equipment that is designed to be robust with respect to design basis external events is assumed to be fully available. Plant equipment that is not robust is assumed to be unavailable. The baseline assumptions are provided below.

2.1 General Elements – Assumptions

The assumptions used for the evaluations of a Columbia ELAP/LUHS event and the development of FLEX strategies are stated below. These assumptions do not represent the numerous less severe conditions that may exist after the initial external hazard occurs, or the plant equipment that may actually be available for

response. In an actual event, all available equipment and resources will be utilized to mitigate the event.

2.1.1 Boundary Conditions

General Criteria and Baseline Assumptions are established to support development of FLEX strategies, as follows:

- The reactor is initially operating at full power, unless there are procedural requirements to shut down due to an impending event. The reactor has been operating at 100% power for the past 100 days.
- The reactor is successfully shut down when required (i.e., all rods inserted, no anticipated transient without scram (ATWS)). Steam release to maintain decay heat removal upon shutdown functions normally, and reactor coolant system (RCS) overpressure protection valves respond normally, if required by plant conditions, and reseat.
- Onsite staff is at site administrative minimum shift staffing levels.
- No independent, concurrent events, e.g., no active security threat.
- All personnel onsite are available to support site response.
- The reactor and supporting plant equipment are either operating within normal ranges for pressure, temperature, and water level, or available to operate, at the time of the event consistent with the design and licensing basis.

2.1.2 Initial Conditions

The following initial conditions are applied:

- No specific initiating event is used. The initial condition is assumed to be a loss of offsite power (LOOP) at a plant site resulting from an external event that affects the offsite power system either throughout the grid or at the plant with no prospect for recovery of offsite power for an extended period.
- All installed sources of emergency onsite AC power and station blackout (SBO) alternate AC power sources are assumed to be not available and not imminently recoverable. Station batteries and associated DC buses along with AC power from buses fed by station batteries through inverters remain available.
- Cooling and makeup water inventories contained in systems or structures with designs that are robust with respect to applicable hazards are available.

- Normal access to the ultimate heat sink (UHS) is lost, but the water inventory in the UHS remains available and robust piping connecting the UHS to plant systems remains intact. The motive force for UHS flow, i.e., service water or circulating water pumps, are assumed to be lost with no prospect for recovery.
- Fuel for FLEX equipment stored in structures with designs that are robust with respect to applicable hazards, remains available.
- Permanent plant equipment that is contained in structures with designs that are robust with respect to applicable hazards, are available.
- Other equipment, such as portable AC power sources, portable back up DC power supplies, spare batteries, and equipment for 10 CFR 50.54(hh)(2) Loss of Large Areas (LOLA), may be used provided it is reasonably protected from the applicable external hazards in accordance with NEI 12-06 (Reference 3), and has predetermined hookup strategies with appropriate procedures/guidance and the equipment is stored in a relative close vicinity of the site.
- Installed electrical distribution system, including inverters and battery chargers, remain available provided they are protected consistent with current station design.
- No additional events or failures are assumed to occur immediately prior to or during the event, including security events.

2.1.3. Reactor Transient

The following additional boundary conditions are applied for the reactor transient:

- Following the loss of all AC power, the reactor automatically trips and all rods are inserted.
- The main steam system valves (such as main steam isolation valves, turbine stops, atmospheric dumps, etc.), necessary to maintain decay heat removal functions operate as designed.
- Safety/relief valves (SRVs) initially operate in a normal manner if conditions in the RCS so require. Normal valve reseating is also assumed.
- No independent failures, other than those causing the ELAP/LUHS event, are assumed to occur in the course of the transient

2.1.4. Reactor Coolant Inventory Loss:

Sources of expected boiling water reactor (BWR) coolant inventory loss include:

- Normal system leakage
- Losses due to BWR recirculation pump seal leakage
- BWR inventory loss due to operation of steam-driven systems, SRV cycling, and RPV depressurization.

2.1.5. Spent Fuel Pool (SFP) Conditions

The initial SFP conditions are:

- All boundaries of the SFP are intact, including the liner, gates, transfer canals, etc.
- Although sloshing may occur during a seismic event, the initial loss of SFP inventory does not preclude access to the refueling deck around the SFP.
- SFP cooling system is intact, including attached piping.
- SFP heat load assumes the maximum design basis heat load for the site.
- The SFP level instrumentation installed in accordance with NRC Order EA-12-051 is functioning normally.

2.1.6. Containment Isolation Valves

It is assumed that the containment isolation actions delineated in current SBO coping capabilities is sufficient.

2.2. Columbia Generating Station Site Specific Elements

These additional assumptions associated with implementation of FLEX Strategies include:

- Off-site deployment resources are assumed to begin arriving at hour 6 and fully staffed by 24 hours.

This plan defines strategies capable of mitigating a simultaneous loss of all AC power and loss of normal access to the UHS resulting from a BDB event by providing adequate capability to maintain or restore core cooling, containment, and SFP cooling capabilities at the site. Specific strategies have been developed. However, due to the inability to anticipate all possible scenarios, the strategies are also diverse and flexible to encompass a wide range of possible conditions. These pre-planned strategies, developed to protect the public health and safety, have been incorporated into the station's emergency operating procedures (EOP) in accordance with the established EOP change processes, and their impact to the design basis capabilities of the station have been evaluated under 10 CFR 50.59, *Changes, Tests and Experiments*.

The plant technical specifications (TS) contain the limiting conditions for normal unit operations to ensure that design safety features are available to respond to a design-basis accident and direct the required actions to be taken when the limiting conditions are not met. The result of the BDB event may place the plant in a condition where it cannot comply with certain TS and/or with its Security Plan, and, as such, may warrant invocation of 10 CFR 50.54(x) and/or 10 CFR 73.55(p)), *Suspension of Security Measures*. This position is consistent with the previously documented Task Interface Agreement (TIA) 2004-04, *Acceptability of Proceduralized Departures from Technical Specifications (TSs) Requirements at the Surry Power Station, (TAC Nos. MC4331 and MC4332)*, dated September 12, 2006 (Accession No. ML060590273).

2.3. Initial Site Access following an event

The event impedes site access as follows as defined by NEI 12-01, Revision 0 (Reference 5):

- Post event time: 0-6 hours - No site access. This duration reflects the time necessary to clear roadway obstructions, use different travel routes, mobilize alternate transportation capabilities (e.g., private resource providers or public sector support), etc.
- Post event time: 6-24 hours—Limited site access: Individuals may access the site by walking, personal vehicle, or via alternate transportation resources that are available to deliver equipment, supplies, and large numbers of personnel.
- Post event time: 24+ hours – Improved site access. Site access is restored to a near-normal status and/or augmented transportation resources are available to deliver equipment, supplies, and large numbers of personnel.

2.4. Staffing assumptions

To support time-sensitive FLEX actions, staffing is assumed to be consistent with NEI 12-06 (Reference 3) guidance.

- No independent, concurrent events, e.g., no active security threat, and
- All personnel on-site are available to support site response.

3.0 MITIGATION STRATEGIES

The Columbia event response actions follow the command and control of the existing site procedures and guidance based on the underlying symptoms that result from the event. The priority for the plant response is to utilize systems or equipment that provides the highest probability for success. Other site impacts

as a result of the event would be addressed according to plant priorities and resource availability.

The Objective of the FLEX strategies is to establish a long term plant coping capability in order to:

- Prevent damage to the fuel in the reactor and the spent fuel pool.
- Maintain containment integrity.

These strategies address station coping capability as a result of a BDBEE that would result in an ELAP and LUHS.

Columbia's coping capability is attained through the implementation of pre-determined FLEX strategies that are focused on maintaining or restoring key reactor core, containment and spent fuel pool safety functions. The FLEX strategies are not tied to any specific damage state or mechanistic assessment of events. Rather, the FLEX strategies are developed to maintain key plant safety functions based on the evaluation of plant response to a coincident ELAP event. A safety function-based approach provides consistency and allows coordination with existing plant EOPs. FLEX strategies are implemented using FLEX Support Guidelines (FSGs).

The strategies for coping with the plant conditions that result from an ELAP/LUHS event involve a three-phase approach as described below:

- Phase 1 Initially cope relying on installed equipment and on-site resources.
- Phase 2 Transition from installed plant equipment to on-site FLEX equipment.
- Phase 3 Obtain additional capability and redundancy from off-site equipment and resources until power, water, and coolant injection systems are restored or commissioned.

The FLEX strategies described below are capable of mitigating an ELAP/LUHS resulting from a BDBEE by providing adequate capability to maintain or restore core cooling, containment, and SFP cooling capabilities. Though specific strategies have been developed, due to the inability to anticipate all possible scenarios the FLEX strategies are also diverse and flexible to encompass a wide range of possible conditions.

These pre-planned strategies which have been developed to protect the public health and safety are incorporated into the Columbia EOPs in accordance with established EOP change processes, and their impact to the design basis capabilities of the Unit evaluated under 10 CFR 50.59.

3.1 Maintain Core Cooling and Heat Removal Strategy

The strategies discussed in Section 3.1 apply in Modes 1, 2 and 3.

The strategy for decay heat removal from the reactor core is to release steam from the reactor pressure vessel (RPV) to the suppression pool using the SRVs. The installed reactor core isolation cooling (RCIC) system is used to maintain RPV water inventory with water from the suppression pool. Decay heat is removed from the suppression pool by venting the wetwell to atmosphere.

Load shedding of the DC bus is initiated upon determination that a loss of all normal or emergency AC power has occurred. This shedding is performed in accordance with plant procedure PPM 5.6.2, *Station Blackout (SBO) and Extended Loss of AC Power ELAP Attachments*, will be completed within 1 hour and is used to extend the station battery availability to at least 8 hours and provides time to deploy one of two available on-site FLEX diesel generators (DG). The FLEX DG will provide power to the battery charger to continue to power key instrumentation.

3.1.1 Phase 1 Strategy

Following the loss of AC power, the main steam system valves and safety relief valves are assumed to operate normally following the automatic trip of the reactor and insertion of control rods. No independent failures, other than those causing the ELAP/LUHS event, are assumed to occur in the course of the transient.

The RPV water level will drop due to continued steam generation by decay heat. The control room operators initially enter the EOPs and plant procedure PPM 5.6.1, *Station Blackout (SBO) and extended Loss of AC Power (ELAP)*. The RCIC system automatically starts on reaching low-low RPV water level (Level 2). The RCIC turbine is driven with a portion of the decay heat steam from the reactor. Using RCIC under ELAP conditions has been evaluated. See calculation ME-02-12-07 in Section 3.1.8 "Mechanical Analysis."

It is assumed that the condensate storage tanks (CST) are unavailable since they are not seismically robust and that the RCIC suction will be realigned to the suppression pool which is seismically robust.

Note: Columbia responded to NRC audit questions 35 (original) and 29-B (February 2015 On-site audit) by stating that:

> The CSTs are no longer credited. Upon reaching CST low level, automatic switch-over of RCIC suction to the suppression pool occurs. The redundant level switches for this function are located in the reactor building (elevation 441). The redundant switch over instrumentation and

components are protected from external events effects and are Seismic Class I. Switch-over is effectuated using two DC-powered motor operated valves (MOV). Both MOVs are located in the reactor building and are Seismic Category I and protected against all environmental challenges. Both valves have hand wheels for manual actuation.

Should CST water become unavailable and automatic switch-over fails to occur due to instrument failure (multiple sensor failing as-is or failing high only), the operators have redundant and diverse indications in the control room to trigger a manual swap-over.

In this alignment, the RCIC system is able to maintain adequate core cooling by providing the RPV with make-up water. Steam flow through the SRVs and the RCIC turbine exhaust is discharged to the suppression pool, removing decay heat from the reactor. The RPV pressure is reduced and maintained within a pressure range of 175 to 300 pounds per square inch gauge (psig).

In Phase 1, the required vital instrumentation is supplied by the station batteries for at least 8 hours, as a result of load shedding.

In Phase 1, operators take the following actions in accordance with the SBO/ELAP procedure:

- Initiate compensatory measures to limit control room temperatures and promote cooling.
- Bypass the RCIC trips for high area temperature, high area differential temperature, and high exhaust pressure to ensure continued RCIC operation.
- Shed loads on Division 1 station batteries E-B1-1 and E-B2-1. Division 2 batteries and non-1E batteries are removed from service to conserve power and reduce heat loads. The station batteries are discussed in Section 3.1.4.4, "Batteries."
- Vent the main generator
- Remove the pin on support RCIC-967N located on the RCIC suction before suppression pool temperature exceeds 170°F. See calculation ME-02-14-09 in Section 3.1.8, "Mechanical Analysis."
- Vent the containment at 6 hours (Anticipatory wetwell venting was approved by the NRC as stated in Reference 6.) A reliable hardened containment vent (HCV) capable of operation under severe accident conditions has been installed at Columbia under EC 13094.
- Maximize RCIC room cooling.

With normal RCIC room cooling unavailable, it is necessary to provide alternate ventilation for the RCIC pump room. A reactor building analyses using a generation of thermal-hydraulic information for containments (GOTHIC) analysis produced CVI-1201-00,2, *GOTHIC Temperature Analysis of the CGS Reactor Building Response to SBO*, which indicates that one of the three plugs in the ceiling of the RCIC pump room must be removed within 12 hours to keep the room within the Licensee Controlled Specification equipment qualification limit of 150°F.

- Implement additional compensatory measures to promote cooling in required areas of the reactor building, control room, and vital island.

3.1.2 Phase 2 Strategy

Electrical Power

As stated in Phase 1, the station batteries are available for at least 8 hours. Prior to depletion, safety related batteries will be charged and associated circuits will be repowered using a FLEX generator. The connection of a 480-V ac FLEX DG has been validated to be accomplished within 160 minutes. Either of the FLEX DGs, discussed in Section 3.1.4.6, can be connected to either of the installed connection points, discussed in Section 3.1.5.1. This connection repowers the station battery charger to charge the batteries which continue to supply DC power to the key parameters identified in Section 3.1.7 used in responding to the event.

Core Cooling and Heat Removal

In Phase 2, RCIC will continue to operate and provide RPV makeup and core cooling. The HCV will continue to transfer heat from the suppression pool to the environment. The RPV pressure is maintained within a pressure range (175 to 300 psig) to facilitate long-term RCIC operation. It should be noted that during FLEX mitigation activities there has been no fuel damage in the core. Therefore, no radioactivity associated with fuel damage is released to the environment.

As evidenced in calculation CVI 1201-00,2 (See Section 3.1.8), maintaining the 300 gpm of makeup to the SFP cascading to the suppression pool starting at 12 hours with the following lineup will maintain the suppression pool temperature less than or equal to 240°F. Operators will initiate connecting the makeup water supply as shown in Attachment A, Figure 1, *Phase 2 Water Makeup Flow Diagram*, additional core cooling and heat removal can be provided. This figure shows both the SFP connection and the alternative method of RPV makeup.

Implementation of these makeup strategies involves connecting hoses from the on-site FLEX pump, which is deployed near the service water (SW) spray ponds. Hoses are laid across the yard area and up the reactor building northeast stairwell or southwest stairwell. In the reactor building, the hose will be connected to a tee and valves to control flow. One discharge path will supply the SFP and the other to the residual heat removal (RHR) system piping at valve RHR-V-63A/B/C. Once connected, the makeup strategy can be used to supply makeup water to the SFP with overflow cascading to the suppression pool or directly to the RPV as an alternative method of cooling the core.

In order to limit the maximum suppression pool temperature to 240°F for long-term RCIC operation, the SFP is cascaded by gravity flow to the suppression pool through fuel pool cooling (FPC) and RHR system piping. To maintain level in the suppression pool below the HCV penetration, water vapor is reduced through the HCV during the venting process. The feasibility of this flow path has been confirmed. See calculations ME-02-14-12 and ME-02-14-13 in Section 3.1.8, "Mechanical Analysis."

ABN-FSG-002, *Water Makeup Strategies for RPV, SFP, DW, WW, CSTS during an Extended Loss of AC Power or other Beyond Design Basis Event*, contains the procedures and diagrams used to deploy the Phase 2 makeup water strategies. Valves will be manually operated as necessary to control the flow to the SFP and RPV.

<u>Alternative RPV Makeup</u>

As shown in Attachment A Figure 1, an alternate means of makeup water is available to supply the RPV through a connection to the RHR system at RHR-V-63A/B/C. This connection to the RHR system provides a backup to the RPV in case RCIC were to fail. The ability of the FLEX pumps to provide the required flow has been verified. See Calculation ME-02-12-06 in Section 3.1.8, "Mechanical Analysis."

The Columbia Phase 2 response uses one pump and one generator out of the following on-site FLEX equipment shown below.

Phase 2 FLEX Equipment			
Plant EPN No. or FLEX Tag No.	**Component Description**	**Location**	**FLEX Strategy Use**
B5B-P-1	On-site High-Head Pump, B5b Pumper Truck	FLEX Storage Building 600	Motive force for reactor coolant water makeup

Phase 2 FLEX Equipment			
Plant EPN No. or FLEX Tag No.	**Component Description**	**Location**	**FLEX Strategy Use**
FLEX-P-1	On-site Hi-Head Godwin Pump HL130M	FLEX Storage Building 82	Motive force for reactor coolant water makeup
E-GEN-DG4	DG-4 480-V ac Diesel Generator Set, 400 kW Cummins	Outside Near the south side of the DG Building	AC source to the station battery chargers to batteries for vital instrumentation
FLEX-GEN-DG5	DG-5 480-V ac trailer-mounted Diesel Generator Set, 400 kW, Caterpillar C15	FLEX Storage Building 600	AC source to the station battery charges to batteries for vital instrumentation

3.1.3 Phase 3 Strategy

Management of SFP and containment conditions using Phase 2 actions can be continued indefinitely.

The Phase 3 strategy begins with the arrival of the off-site equipment from the National SAFER Response Center (NSRC). Included in this equipment are a generator and pump shown below provide additional capability and redundancy for the on-site Phase 2 FLEX equipment and use the same connections as the on-site FLEX pumps and generators. Columbia will receive the following NSRC equipment designed to provide the same function and use the same connection points as the on-site Phase 2 FLEX equipment:

NSRC Component	Performance Description
480 VAC Diesel-powered Generator	480 Volts AC/1000 kW
SG/RPV Water Makeup Pump	500 PSI/500 GPM

The NSRC equipment above is to arrive within 24 hours of informing the NSRC of the event at either the Portland or Seattle airports. It will then be transported by truck to Staging Area B which is the parking area located by the on-site training facility (Coordinates 46-28-16N/119-20-02W) shown in Attachment A Figure 2A. Staging Area A is the final in-place location of the equipment as identified in Figure 2. Additional NSRC equipment is discussed in Section 3.1.4.8.

3.1.4 Systems, Structures, and Components

3.1.4.1 Ventilation

Following the onset of an ELAP event, ventilation to occupied areas and areas where FLEX equipment may be relied upon to implement FLEX strategies will be lost. Analyses have been performed to determine the actions required to preserve the radwaste and reactor buildings access during the first 72 hours of the ELAP event.

Radwaste Building

In the radwaste building, the key areas identified for the execution of the FLEX strategies include the vital island and the control room. These areas have been evaluated in calculation CVI 1201-00,1, *GOTHIC Analysis of CGS Radwaste Building Response to SBO*, to determine the temperature profiles following the onset of an ELAP event during Phases 1 and 2. Based on the GOTHIC analyses, actions were identified in TM-2187, *Actions, Limitations, and Notes Associated with an Extended Loss of AC Power (ELAP)*, that need to be taken to support cooling of the vital island rooms and the control room. These actions are detailed in the attachments to plant procedure 5.6.2, *Station Blackout (SBO) and Extended Loss of AC Power ELAP Attachments*. The table below identifies the equipment located in the radwaste building that support the key safety functions of core cooling and containment integrity as identified in NEI 12-06. The table shows that no room temperature will reach a License Controlled Specification (LCS) limit within the first 72 hours of the event by which time the NSRC equipment identified in Section 3.1.4.8, "Additional Off-site Equipment" will be on site. The table also shows the expected temperature at seven days if no further action is taken. Procedures to use the equipment in Section 3.1.4.8 are discussed in Section 3.1.10, "Procedures."

Equipment Room Temperature Table

Room	**Description**	**LCS Equipment Temperature Limit, °F**		**Time to Reach LCS Limit, days**	**Temperature @ 7 days, °F**
C206	Div 2 Critical Switchgear Room 2	E-SM-8 E-SL-81, 83	120 120	4.33 4.33	129.2
C208	Div 1 Critical Switchgear Room 1	E-SM-7 E-SL-71,73 E-DP-S11F	120 120 129	3.67 3.67 5.92	132.5
C210	Div 1 Battery Room 1	E-B2-1, E-B1-1 E-B0-1A, E-B0-1B	110 110	3.54 3.54	121.7
C211	RPS Room 1	E-MC-7A E-MC-S1/1D E-IN-3A, 3B	129 129 131	3.63 3.63 4.08	141.6

Room	Description	LCS Equipment Temperature Limit, °F		Time to Reach LCS Limit, days	Temperature @ 7 days, °F
C213	RPS Room 2	E-MC-8A E-MC-S1/2D RPS-EPA- 3A,3C,3E,3F E-IN-2A, 2B E-C0-3	129 129 129 131 140	4.39 4.39 4.39 4.88 NA NA	138.2
C215	Div 2 Battery Room 2	E-B2-2 E-B0-2A, E-B0-2B E-B0-3	110 110 110	3.42 3.42 3.42	122.7
C216	Div 1 Battery Charging Room 1	E-C2-1, E-C0-1A,1B E-MC-S2/1A,1B E-C1-1A,1B	122 129 131	3.08 4.88 5.13	137.2
C224	Div. 2 Battery Charging Room 2	E-C0-2A,2B E-C1-2A,2B	122 131	3.79 6.04	134.1
C414	Main Control Room	All Safety Related	120*	3.92	129.0
C507	HVAC Room 1	WMA-AH-51A WMA-AH-52A WMA-AH-53A E-MC-7F	120 120 120 129	NA	116.2
C508	HVAC Room 2	WMA-AH-51B WMA-AH-52B WMA-AH-53B E-MC-8F	120 120 120 129	NA	110.6

* Although LCS limits the main control room to 104°F, the station blackout analysis in FSAR Chapter 8A defines a limit of 120°F for this area.

As shown in the Equipment Room Temperature Table, the most limiting equipment relied on for the Columbia response is E-C2-1 (250V BATTERY) which provides RCIC functional support and reaches its temperature limit in approximately 77 hours. Validation Plan 10 shows the FLEX pump being connected starting at 2 hours into the event with a time constraint of 12 hours. The FLEX pump will be in operation able to supply SFP and RPV cooling. Additionally, the NSRC equipment is expected begin arriving in 24 hours from notification.

Therefore, the continued use of RCIC in Phase 2 is not relied upon once the on-site FLEX pump and generator are placed in service.

<u>Reactor Building</u>

Section 2.2.3.1 Determination of Design Basis Events, of the FSAR contains the following on the design of the reactor building.

The reactor building is a reinforced-concrete structure up to the refueling floor and is designed to withstand the worst probable combination of wind velocity and associated pressure drop due to a design basis tornado. A differential pressure

of 3 psi between the exterior and interior of the building is also considered in the design.

FSAR Table 3.2-1 provides the following information on the reactor building:

Table 3.2-1 Equipment Classification

46. Buildings	Safety Class	Quality Class	Seismic Category
Reactor building	2	I	I

In the reactor building, the key areas identified for implementation of the FLEX strategies include the refueling floor, RHR-V-63A/B/C valve rooms, and the RCIC pump room. These areas have been evaluated in calculation CVI-1201-00,2, *Reactor Building GOTHIC Temperature Analysis during PSBO/ELAP*, to determine the temperature profiles during an ELAP event and include the heat load introduced by operation of the RCIC turbine-driven pump and passage of steam through the HCV. When SFP cooling is lost, a significant amount of energy is released at the refueling floor.

Following the onset of an ELAP, reactor building ventilation is established by natural convection taking advantage of the building's height. Actions required within 12 hours are identified in plant procedure PPM 5.6.2. These actions preserve reactor building accessibility.

Also following the onset of an ELAP, normal cooling of the SFP is lost. As a result SFP temperature starts to rise. The results from the GOTHIC analysis demonstrate the maximum temperature reached is 135°F at 12 hours, after which temperature drops because of the cooling effect of the 300 gpm of makeup/cooling water that is assumed to start at 12 hours.

During the 72-hour ELAP transient that is analyzed, the only LCS limit exceeded in the reactor building was the general area of the refueling floor (606'), which reached 111°F in comparison to the 104°F LCS limit. The peak value only occurs temporarily due to the diurnal cycle in ambient temperature. The daily average temperature is approximately 103°F.

Actions to mitigate the temperatures in the identified areas of the reactor building are identified in plant procedures. See discussion of PPM 5.6.2 in Section 3.1.10, "Plant Procedures."

An additional ventilation concern is the potential buildup of hydrogen in the battery rooms. Calculation ME-02-13-14 addresses the off-gassing of hydrogen from the station batteries when the batteries are charging after power is restored in Phase 2 and confirms that there is no hydrogen accumulation approaching

flammability limits during battery charging. See Section 3.1.9, "Electrical Analysis" for additional discussion.

3.1.4.2 Reactor Core Isolation Cooling (RCIC) Pump

The RCIC system is not an emergency core cooling system (ECCS) or an engineered safety feature (ESF) system. However, portions of the system are safety-related. RCIC is designed to initiate during plant transients that result in low reactor water level. The RCIC system is located in the reactor building providing protection from the hazards discussed in Section 4.0, "Characterization of External Events."

FSAR Table 3.2-1 provides the following information on RCIC establishing its robustness.

Table 3.2-1 Equipment Classification

	Safety Class	Quality Class	Seismic Category
13. RCIC*	1 & 2	I	I

* All components with the exception of:

- Piping, drip pot discharge valve to the condenser
- Piping, Condenser to vacuum tank and to the condensate pump discharge and vacuum pump discharge to the outboard check valve break flange

Additional room cooling is required to maintain RCIC operation as discussed in Section 3.1.1.

The RCIC system provides make-up water to the reactor vessel when the vessel is isolated. The RCIC system consists of a steam turbine-driven pump which operates automatically to provide sufficient coolant flow to maintain adequate water level in the reactor vessel. The RCIC pump has a recommended minimum flow rate of 300 gpm under normal system operating conditions. Low flow rates at elevated pumped fluid temperatures can potentially reduce pump reliability. The minimum flow line associated with the pump was designed for 100 gpm, and that value has been used in assessments as a reasonable short term operation limit for minimum RCIC flow. A total RCIC flow rate of 100 gpm is below the recommended normal minimum of 300 gpm. However, 100 gpm is the design flow rate of the RCIC (minimum flow) line. Therefore, the RCIC system is expected to be able to support continued coping of an ELAP as shown in ME-02-12-07.

The industry and Energy Northwest have evaluated RCIC performance and concluded that the RCIC system can be operated with pumped water

temperature as high as 240°F without risking failure over the duration. RCIC can take suction from the CSTs or suppression pool and is normally aligned to the CSTs. However, in an ELAP event the CSTs are assumed to be unavailable and the RCIC suction will be realigned to the suppression pool. Calculation ME-02-12-18 shows the response of the suppression pool (temperature, pressure, level) and that RCIC net positive suction head available (NPSHa) under this scenario remains adequate for at least 40 hours after ELAP event initiation with no water makeup to the suppression pool. Analyses of RCIC and the suppression pool are discussed in Section 3.1.8, "Mechanical Analysis."

3.1.4.3 Reactor Pressure Relief System

A pressure relief system, consisting of safety/relief valves (SRV) on the main steam lines is provided to prevent excessive pressure inside the nuclear system following an abnormal operational transient or accident.

Eighteen SRVs are mounted on the four main steam lines. When SRVs are actuated, steam from the RPV flows through the SRV discharge lines into the suppression pool where the steam is condensed.

The SRVs can be opened by energizing a solenoid pilot valve (SPV) and are used by operators to manually control RPV pressure. Seven of the SRVs are used for automatic depressurization. The automatic depressurization system (ADS) SRVs are equipped with an air accumulator and backup air source to ensure that the valves can be held open following failure of the normal air supply.

Three of the seven ADS SRVs and the SPVs which actuate the SRVs are environmentally qualified for the full post loss of cooling accident (LOCA) time frame. All other SRVs and their SPVs are qualified for 24 hour post-LOCA. During the February 2015 audit (Question 14-E), Columbia was requested to provide a summary evaluation to confirm that the temperature and pressures within containment will not exceed the environmental qualification of electrical equipment that is being relied upon as part of the FLEX strategies. The summary provided consisted of calculation ME-02-14-13 Appendix A, pages A-5, A-80 and A-81; calculation ME-02-14-12 page 5.100; calculation ME-02-12-18 Appendix A, pages 1 and 5 of 74; Appendix A9, pages 1 and 5 and of 70; and FSAR Figure 3A.2.1-7. The identified calculations are discussed in Section 3.1.8, "Mechanical Analysis."

3.1.4.4 Batteries

The safety related batteries and associated DC distribution systems are located within safety related structures providing protection from the hazards discussed in Section 4.0, "Characterization of External Hazards." These batteries will be

used to initially power required key instrumentation and applicable DC components. The batteries for the HCV have been installed in the same location. As shown if Section 3.1.4.1, the battery room temperatures do not reach LCS limits during the first 72 hours of the event providing ample time to establish an additional room cooling if required.

FSAR Table 3.2-1 provides the following information on the radwaste/control building and the station batteries establishing their robustness.

Table 3.2-1 Equipment Classification

46. Buildings	Safety Class	Quality Class	Seismic Category	Notes
Radwaste/control building	3/G	I, II	I/II	33

Note 33. Those portions of the radwaste and control building that house systems or components necessary for safe shutdown of the reactor are designed to Quality Class I and Seismic Category I requirements. Those portions of the radwaste building housing equipment containing significant quantities of radioactive material are designed to Seismic Category I requirements.

Table 3.2-1 Equipment Classification

39. Aux 125/250-V dc power	Safety Class	Quality Class	Seismic Category
Batteries	2	I	I
Chargers	3	I	I

The 125 volt station batteries supply power to the SPVs that actuate the SRVs. The 125 volt batteries can support SRV actuations and other required loads for at least 8 hours without recharging. The station battery capacity was calculated in accordance with IEEE-485, *Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications*, methodology using manufacturer discharge test data applicable to the Columbia FLEX strategy as outlined in the NEI white paper on "Battery Life Issues" and endorsed by the NRC in Reference 7. See calculation 2.05.01 in Section 3.1.9, "Electrical Analysis."

As shown in the hydrogen generation analysis, the battery rooms were found to result in a maximum concentration less than 0.5 percent 7 days after AC power loss and below the 4 percent flammability limit between 5 and 6 days. Even with the delay in opening doors in the Vital Island, from 2 hours to 8 hours, the hydrogen concentration in the battery rooms remained below 0.5%. See CVI 1201-00,1 in Section 3.1.9, "Electrical Analysis."

3.1.4.5 On-Site FLEX Pumps

Two FLEX pumps (N and +1) are stored in FLEX buildings B600 and B82. These buildings are discussed in Section 5.1. They are used for supplying cooling water during Phase 2. One FLEX pump is a high-head diesel-powered

(Godwin) pump rated at 600 gpm at 274 psig. No equipment lighting was supplied with this equipment. Lighting is discussed in Section 8.3, "Lighting."

The other FLEX pump is a truck-mounted pump used to meet the B.5.b requirements and is rated at 500 gpm at 270 psig. Each pump is supplied with non-collapsible suction hose, a wye-connection and isolation valve assembly, and two floating suction strainers. The wye-connection and isolation assembly allows the cleaning of one suction strainer without interruption of FLEX pump flow. During pump operation, pump strainers will be routinely monitored and cleared of debris as required.

Calculation ME-02-12-06 evaluated the performance of both FLEX pumps which is summarized in Section 3.1.8, "Mechanical Analysis." This analysis shows that each pump can meet the flow requirements for both the core cooling and SFP cooling strategies. Therefore, the B.5.b pumper truck pump can be used to meet the N+1 requirement. The B.5.b pumper truck is stored in FLEX Building 600 and the Godwin pump is stored in FLEX Building 82.

Both FLEX pumps are stored in locations that protect them from the hazards identified in Section 4.0. The pumps must be protected from ashfall when positioned in their response locations. Procedure ABN-ASH provides the steps required to install and maintain oil-bath intake filters which are also stored in the FLEX buildings.

3.1.4.6 On-Site FLEX Generators

The FLEX strategy to re-power the station's battery chargers requires the use of a 480-V ac diesel-powered FLEX generator. Two FLEX generators are available. One FLEX generator, DG4, is a previously existing generator used to support of Technical Specification 3.8.1 AC Sources - Operating Action B.4 Completion Time by extending the Completion Time up to 14 days. It is located outside near the DG building; the other, DG5, is located in FLEX building 600. Both of the FLEX generators are 400 kW standby rating generators and are capable of being connected to either of the connection points described in Section 3.1.5.1 below. Each on-site FLEX DG is stored with the cabling required to connect the generator to the FLEX connection points. No equipment lighting was supplied with this equipment. Lighting is discussed in Section 8.3, "Lighting."

Calculation E/I-02-91-03 documents that any load increases as a result of its use during a BDBEE do not exceed the load rating of DG4. This calculation shows the expected load on DG4 is 230 KW. The loading assessment performed showed that the loading does not exceed the DG4 nameplate. DG5 is

acceptable as an equivalent to DG4. See Section 3.1.9, "Electrical Analysis." Both FLEX generators have adequate overcurrent protection to be connected to Class 1E equipment.

DG5 is stored in FLEX building B600 which protects it from the hazards identified in Section 4.0. DG5 must be protected from ashfall when positioned in its response locations. Procedure ABN-ASH provides the steps required to install and maintain oil-bath intake filters which are also stored in the FLEX buildings. DG4 is located outside near the diesel generator building. This area has been evaluated as an acceptable storage area for the hazards identified in Section 4.0 with the exception of ashfall. Procedure ABN-ASH provides the steps required to install and maintain oil-bath intake filters which are also stored in the FLEX buildings.

The fueling/refueling of the FLEX DGs is discussed in Section 6.5, "Fueling of FLEX Equipment."

3.1.4.7 Standby Service Water Spray Ponds

As stated in the FSAR, the ultimate heat sink consists of two man-made Seismic Category I spray ponds and is designed to withstand extreme natural phenomena. The two adjacent spray ponds are Seismic Category I structures that are not part of the building complex. Each is provided with an integrally constructed standby service water pump house.

Section 4.0 discusses impact of the various external hazards on the spray ponds.

Makeup water for the Columbia mitigation strategies comes from spray ponds, A and B. Each pond is approximately 250 ft. by 250 ft. and 15 ft. deep. A siphon between the ponds allows for water flow from one pond to the other. In the event that the spray pond level falls below the minimum level required for 30 days of cooling, an alarm is sounded and makeup to the spray ponds is provided using the tower makeup system (TMU). The UHS is capable of accomplishing its safety function for a normal cooldown or an emergency cooldown following a LOCA without the availability of offsite power. The UHS provides cooling capability for a period of 30 days without outside makeup (except following a tornado). Provisions are made for replenishment of the UHS to allow continued cooling capability beyond the initial 30-day period.

Section 3.1.8, "Mechanical Analysis," discusses the results of questions concerning recirculation pump seal leakage (ME-02-12-06), spray pond water losses (ME-02-14-02), and ashfall (ME-02-15-04) raised during the February 2015, NRC audit. Surveillance Requirement (SR) 3.7.1.4 is performed in accordance with the Surveillance Frequency Program and verifies the average

sediment depth in each UHS spray pond is less than 0.5 feet. Plant procedure PPM 12.14.1, *Chemical Treatment of Standby Service Water*, provides controls and documentation for chemical treatment of the standby service water system. The chemical treatment consists of the addition of corrosion and biological inhibitors. Blowdown and makeup are performed as necessary to maintain the control limits as defined in SWP-CHE-02, "Chemical Process Management and Control."

In addition to the spray ponds, two other non-credited water sources may be available.

1. Two interconnected condensate storage tanks each containing 400,000 gallons.
2. One fire protection embankment supported bladder tank containing 400,000 gallons.

3.1.4.8 Additional Off-Site NSRC Equipment

In addition to the generator and pump discussed in Section 3.1.3, Columbia will receive two 4160 volt generators and a large diesel driven pump shown below.

<table>
<tr><th>Component</th><th>Description</th><th>Unit</th></tr>
<tr><td rowspan="6">Medium Voltage Generator</td><td rowspan="2">Performance</td><td>4160 VAC</td></tr>
<tr><td>1 MW</td></tr>
<tr><td>Quantity</td><td>2</td></tr>
<tr><td rowspan="2">Fuel Consumption</td><td>103 GPH</td></tr>
<tr><td>12 HRS</td></tr>
<tr><td>Fuel Tank Capacity</td><td>1247 GAL</td></tr>
<tr><td rowspan="8">Low Pressure /High Flow (Dewatering) Pump (Generic)</td><td rowspan="4">Performance</td><td>150 PSI</td></tr>
<tr><td>5000 GPM</td></tr>
<tr><td>12 FEET</td></tr>
<tr><td>140 °F</td></tr>
<tr><td>Quantity</td><td>1</td></tr>
<tr><td rowspan="2">Fuel Consumption</td><td>26 GPH</td></tr>
<tr><td>12 HRS</td></tr>
<tr><td>Fuel Tank Capacity</td><td>400 GAL</td></tr>
</table>

This equipment can be used to support longer term recovery efforts by repowering certain equipment and providing SW flow directly through the SW

system. The equipment can be used to restart shutdown cooling and supply vital area room coolers when and if needed.

A special adapter is used to connect the discharge of the NSRC pump to the bonnet of the SW pump discharge check valve.

3.1.5 FLEX Connection Points

3.1.5.1 480 Volt AC Electrical Connections (Two)

One 480-V ac connection point is located on the outside wall of the DG building, Figure 3. A second 480-V ac connection point is located in the radwaste building at Elevation 437, Figure 4. Both buildings are Seismic Category 1. Either connection point can be used to supply Division 1 or Division 2 480 volt electrical loads. Both on-site FLEX 480-V AC DGs have sufficient cabling and connectors to access either connection point.

FSAR Table 3.2-1 provides the following information on the diesel generator building establishing its robustness. The radwaste building is discussed in Section 3.1.4.1 above.

Table 3.2-1 Equipment Classification

46. Buildings	Safety Class	Quality Class	Seismic Category
Diesel generator building	3	I	I

Cabling for the connections is color coded.

3.1.5.2 4160 Volt AC Connections

The NSRC supplied 4160-V AC generators can be connected to the 4.16-kV buses via two diverse lug-and-connect connection points. One connection pathway is via the turbine generator (TG) building, the other, via the DG building. Two storage lockers containing cabling for the connection through the TG building are located in the TG building at different locations. The cabling for the connection through the DG building is supplied with the NSRC generators. Cabling for the connections is color coded. See Appendix A Figures 5 and 6 for connection points.

3.1.5.3 Water Supply Connections

As shown in Figure 1, a FLEX pump is staged at the SW spray pond and used to establish a make-up water supply to the core and SFP. A hose is connected from the FLEX pump to a tee connection. From the tee, a hose is run to the SFP and to one of the three RHR flanges at RHR-V-63A/B/C using a special adaptor with a hose fitting. If for some reason the refueling floor in not accessible, water can be supplied to the SFP without accessing the refueling floor by using RHR-V-

63B. Water can also be supplied directly to the RPV using any of the three connections. A validation was used to verify making these connections as the restriction is maintaining accessibility to the reactor building. This action could start as early as 2 hours into the event. The actual set-up time is dependent upon the number of off-site responder available. It is expected that the water supply connections will be made in less than 12 hours from the event initiation.

Operator habitability has been evaluated based on the reactor building room temperature results. The results show that the connection areas have unlimited access times based on temperature and humidity.

3.1.6 Reactor Recirculation Pump Seals

The reactor recirculation pumps are described in FSAR Section 5.4.1 Reactor Recirculation Pumps. During normal operation, reactor coolant system leakage is limited by TS 3.4.5 to less than 25 gpm and reactor recirculation pump seal leakage is limited to 25 gpm (per pump) by the seal breakdown bushing. Section 3.1.8, "Mechanical Analysis," discusses the results of questions raised during the February 2015, NRC audit (ME-02-12-06). Analysis shows that reactor recirculation pump seal leakage is not a concern considering the low expected value (less than 2 gpm total for two pumps) and the large flow capability margin (175 gpm) at the significantly reduced reactor pressure during an ELAP event when providing make-up to the RPV using either FLEX pump. With the low expected seal leakage, no significant effect is expected on drywell temperature.

3.1.7 Key Parameters

Instrumentation providing the following key parameters is credited for all phases of the FLEX strategies.

Reactor Vessel Essential Instrumentation:

- RPV level – Wide Range (MS-LR/PR-623A)
- RPV pressure (MS-LR/PR-623A)

Containment Essential Instrumentation:

- Drywell pressure (CMS-PR-1)
- Suppression pool (Wetwell) pressure (CMS-PR-3)
- Suppression pool level (CMS-LR-3)
- Drywell temperature (CMS-TI-5)
- Suppression pool temperature (SPTM-TI-5)

The following additional instruments will remain powered throughout the event to assist with the mitigation of a loss of core cooling:

- RCIC flow and control (RCIC-FIC-600)
- Containment radiation monitor (CMS-RIS-27E)

For all instruments listed above, the normal power source and long-term power source are the 125 VDC vital batteries.

- In responding to an event during a full-core offload, the SFP level instrumentation installed in accordance with NRC Order EA-12-051 is available to monitor level.

  This instrumentation is powered by a dedicated power system.

In the unlikely event that 125 volt DC vital bus infrastructure is damaged, procedure ABN-FSG-001, *Accessing Essential Instrumentation during Extended Loss of AC Power with no Power Available*, is in place for obtaining the critical parameters locally. Key parameters can be obtained in the control room, at the remote shutdown panel, alternate remote shutdown panel, or locally at the instrument racks. These readings are obtained using self-powered FLUKE meters. In addition, a portable tachometer is available for operating RCIC with no AC or DC power. Spent fuel pool level instrumentation (SFPLI) is not included as its power is not supplied from the 125 volt DC vital bus (station batteries).

On-site FLEX equipment is supplied with the local instrumentation required for operation. The use of this instrumentation is detailed in the associated procedures for use of the equipment. These procedures are based on inputs from the equipment suppliers, operating experience, and the expected equipment function during an ELAP event.

3.1.8 Mechanical Analysis

3.1.8.1 Containment Response

The containment consists of primary and secondary containment systems. The primary containment structure is a free-standing steel pressure vessel which contains both a drywell and a suppression chamber (wetwell). The secondary containment structure is composed of the reactor building, which completely encloses primary containment.

FSAR Table 6.2-1 provides the following design parameters:

Parameter	Drywell	Suppression Chamber
Internal design pressure, psig	45	45
External design pressure, psig	2	2
Drywell deck design differential pressure, psid	25 (downward) 6.4 (upward)	
Design temperature, °F	340	275
Net free volume, $ft.^3$ (drywell includes vents)	200,540	144,184 maximum
Maximum allowable leak rate, %/day	0.5	0.5
Suppression chamber free volume, minimum, $ft.^3$		142,500
Suppression chamber water volume minimum[a] $ft.^3$		112,197
Pool cross section area, $ft.^2$		5,770
Pool free surface cross section area, $ft.^2$		4,520
Pool depth (normal), ft.		31

Calculations ME-02-12-18, *Containment Response during Extended Loss of AC Power (ELAP) – A Beyond Design Basis Assessment* and ME-02-14-13, *Containment Response During an Extended Loss of AC Power (ELAP) with Suppression Pool Makeup*, determine the conditions in the Columbia containment and the reactor pressure vessel (RPV) for 72 hours during a BDBEE resulting from an ELAP. The Modular Accident Analysis Program (MAAP), Version 4.0.4, BWR (Boiling Water Reactor) code was used for these analyses. Additionally, these calculations were developed consistent with the guidelines contained in the 2013 EPRI Technical Report 3002001785, with regards to the use of MAAP4 in support of post-Fukushima applications. The primary outputs of interest are wetwell and drywell pressure, wetwell, drywell, and suppression pool temperature, and the maintenance of the RPV water level over the top of the active fuel. Reference 35 contains the Columbia analysis of the letter of October 3, 2013 from Jack Davis (NRR) to Joe Pollock (NEI) (ADAMS Accession Number ML13275A318) regarding use of MAAP4 in simulating ELAP events for BWRs, addressing each one of the limitations stated in the NRC endorsement letter.

These calculations determine containment conditions following a BDBEE, including drywell pressure and temperature, suppression pool temperature, containment pressure and temperature, and reactor water level. The analysis time of 72 hours is used to verify that the FLEX strategies are effective in controlling conditions inside containment within acceptable parameter values prior to the arrival and use of the NSRC equipment.

During the development of the mitigation strategies, these two calculations (ME-02-12-18 and ME-02-14-13) included several cases. In ME-02-12-18 numerous combinations of actions and timing, such as the time the HCV is opened, were

considered that could be taken to mitigate the potential effects of the loss of AC power. The actions considered were focused on preventing core damage. Therefore, accidents involving nuclear fuel damage were not addressed and focused on the ability to maintain conditions which support RCIC operability since it is the primary, installed system capable of cooling the core during an ELAP. The use of the portable FLEX equipment was also considered, especially as it supports RCIC operability.

It was determined that makeup to the SFP was beneficial in maintaining accessibility to the reactor building, and that the makeup could be cascaded to the suppression pool, benefiting both the SFP and suppression pool by lowering temperatures while maintaining SFP level. Calculation ME-02-14-13 was prepared to quantify the effect of the cascading of SFP makeup on suppression pool temperature response, and expected water level increase in the suppression pool. It was shown that 300 gpm makeup to the SFP, cascaded to the suppression pool, gave acceptable conditions for long term RCIC operability without flooding the wetwell vent within 72 hours after loss of power. With the 300 gpm cascaded makeup it was also found that a vent flow resistance coefficient, K, of 4.6 or less would be needed to maintain suppression pool temperature below 240°F, which is consistent with long term, reliable RCIC operation. Calculation ME-02-13-03, *Pipe Sizing and Pressure Drop Calculation for the Hardened Containment Vent System,* modeled the HCV piping using the RELAP code. The objective of that calculation was to size the piping system for passage of steam equivalent to 1% of reactor rated thermal power at containment design pressure as required by EA-13-109 for severe accidents, and to meet the requirement for a flow resistance coefficient less than 4.6 to support long term RCIC operation. The final design of the HCV meets those requirements with significant margin. The final flow resistance value of the installed HCV is less than 3.5.

3.1.8.2 Thermal Hydraulic Response Analysis

The analysis reported in ME-02-14-13 represents the culmination of the efforts to obtain the optimal set of actions and timing for mitigation of the ELAP at Columbia. Actions and associated assumptions related to the thermal hydraulic performance in containment are summarized below. Other required actions, such as those needed to mitigate temperatures in the radwaste and reactor buildings are discussed in Sections 3.1.4.1 and 3.1.10. The results of the analyses are given in Section 3.1.8.2.2. All times are referenced to the loss of power, and pressure results in MAAP are in psia.

3.1.8.2.1 Action, Timing, and Assumptions

a. Reactor pressure is cycled between 175 and 300 psig.
The RCIC system requires sufficient steam within the proper pressure range to fulfill its water injection function. Columbia has established the reactor pressure range as 175 to 300 psig. With controlled makeup from the RCIC system and steam release through an SRV to the suppression pool, the reactor is cooled down at approximately 80°F per hour, but not faster than the Technical Specification limit of 100°F. The cooldown continues until reactor pressure is 175 psig, at which point the SRV is closed to prevent further pressure reduction that could eventually inhibit RCIC's ability to supply sufficient RPV makeup. With the SRV closed the pressure rises; when it reaches 300 psig the SRV is opened and the cycle is established. This cycling has been simulated in the MAAP programing.

b. Anticipatory venting of the containment is done at 6 hours or less using the hardened containment vent.

   In calculation ME-02-12-18 vent opening times of 2, 4, 6, and 8 hours were investigated in MAAP analysis to determine the effect on maximum suppression pool temperature. Suppression pool temperature must be below 240°F to assure long term reliability of the RCIC system when taking suction from the suppression pool, as it does in the ELAP scenarios. Early venting is a factor affecting suppression pool temperature. In this calculation it was found that the temperature remained below 240°F when the vent was opened at 6 hours.

   While it was found in later scenarios that the suppression pool temperature rise could be reduced by cascading SFP makeup to the suppression pool, the 6 hour venting recommendation was retained because the SFP makeup was only needed within 12 hours, and should there be a delay in providing that makeup, venting at 6 hours assures that the suppression pool temperature remains below 240°F even without cascading SFP makeup to the suppression pool.

c. Makeup water from a spray pond is provided at a rate of approximately 300 gpm to the spent fuel pool within 12 hours.

   As discussed in Section 3.1.2, it was found that accessibility of the reactor building was maintained if sufficient SFP makeup could be provided to limit boiling. GOTHIC analyses (CVI 1201-00,2, See Section 3.1.8) of the reactor building confirmed that 300 gpm makeup to the SFP provided within 12 hours could maintain accessibility in the reactor building.

Suppression pool makeup water is also needed to preserve NPSH for the RCIC pump within about 40 hrs. Providing makeup within 12 hours assures that net positive suction head (NPSH) will be preserved.

d. Prior to the ELAP, the plant is operating at full rated power including the measurement uncertainty recapture power uprate. The assumed initial power level bounds the current licensed power level.

e. Decay heat in the MAAP code was increased to meet the Auxiliary Systems Branch Technical Position decay heat formulation, ASB BTP 9.2.

   Many analyses at Columbia used the BTP 9.2 decay heat formulation, which is somewhat higher than that of the MAAP code. The decay heat values in the MAAP code were increased in the site specific programming to include the conservatism inherent in the BTP 9.2 formulation.

f. Maximum spray pond temperature during normal operation (77°F) was used for SFP makeup water.

   This parameter affects the effectiveness of the SFP makeup water in cooling the SFP and the suppression pool. The 77°F temperature is the maximum value allowed in Technical Specifications.

g. A conservative temperature of 130°F was used for the water cascading to the suppression pool.

   Calculation ME-02-14-13 determined that with makeup water at 81°F, the water cascading to the suppression pool would be approximately 125°F. To add conservatism the cascading water is assumed to be 130°F in the MAAP analyses.

3.1.8.2.2 Results of Thermal Hydraulic Analyses

NEI 12-06 Revision 2 stated in Section 3.3, *Considerations in Utilizing Off-Site Resources*, that site access is considered to be restored to near-normal within 24 hours, by 72 hours from the event initiation, outside resources should be able to be mobilized by that time such that a continuous supply of needed resources will be able to be provided to the site. Within these first 72 hours a site will have deployed its FLEX strategies which should result in a stable plant condition on the FLEX equipment and plans will have been established to maintain the key safety functions for the long term. Therefore, FLEX strategies and/or resources are not required to be explicitly planned in advance for the period beyond 72 hours.

Using the MAAP 4.0.4 computer code, analyses of the first 72 hours was performed. These results were used to identify the mitigating actions and timing discussed in the Columbia response.

a. Summary of Key Input

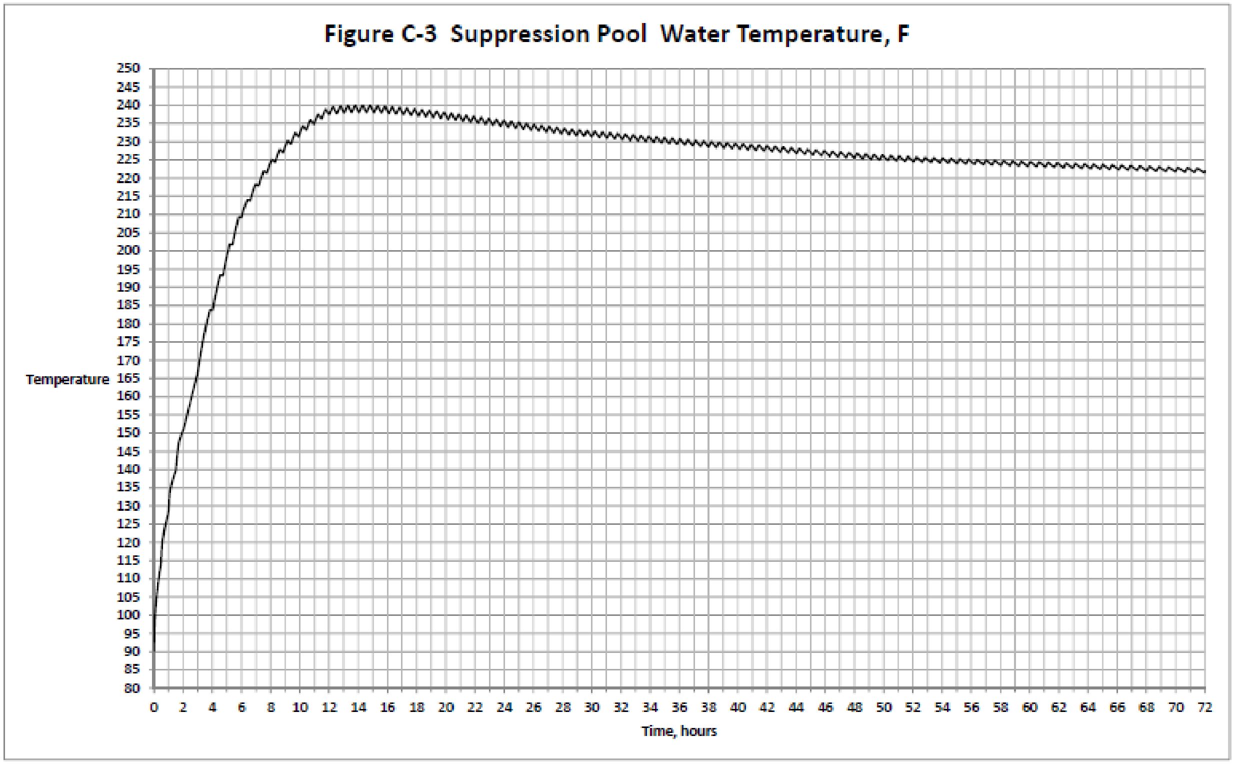
1. Reactor Cooldown rate approximately 80°F/hr.
2. RCIC suction from the suppression pool
3. Makeup to the suppression pool at 12 hours, 300 gpm, 130°F water (cascaded from the SFP)
4. Wetwell vent flow resistance coefficient K=4.6
5. Wetwell vent opened at 6 hours

b. RPV Pressure Cycling Dynamics

Operators will manually control the RPV pressure in the range 175 to 300 psig. The MAAP results indicate that after closing the SRV the pressure rises from 175 to 300 psig in about 30 minutes. When the SRV is opened, pressure decreases from 300 to 175 psig in about 12 minutes. One open/close cycle takes 42 minutes, giving about 35 cycles per day. There are 7 automatic depressurization system (ADS) valves that can be used, and with use properly rotated among them, each valve would only be cycled 5 times/day.

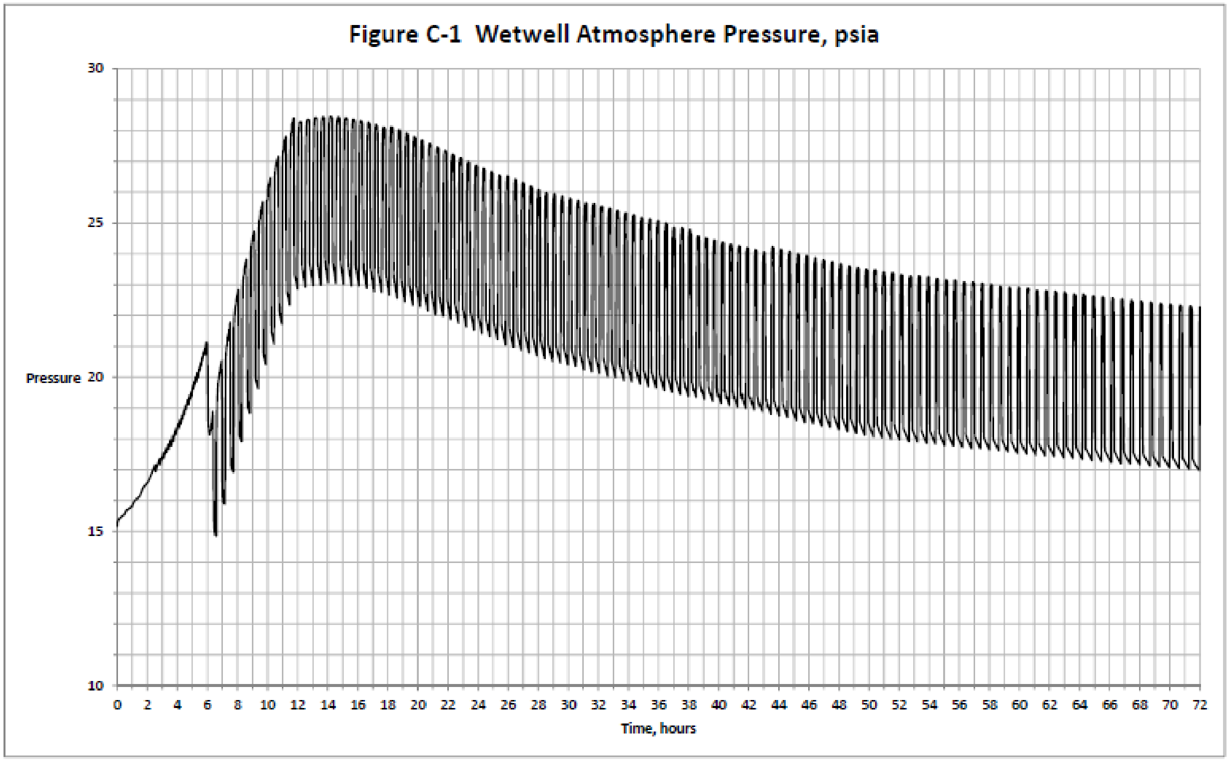
c. Wetwell Temperature

As shown in the graph below, the suppression pool temperature peaks at 240°F at approximately 12 hours after loss of power. The timing coincides with the commencement of cascaded makeup from the spent fuel pool. Temperature then gradually decreases to 222°F at 72 hours.



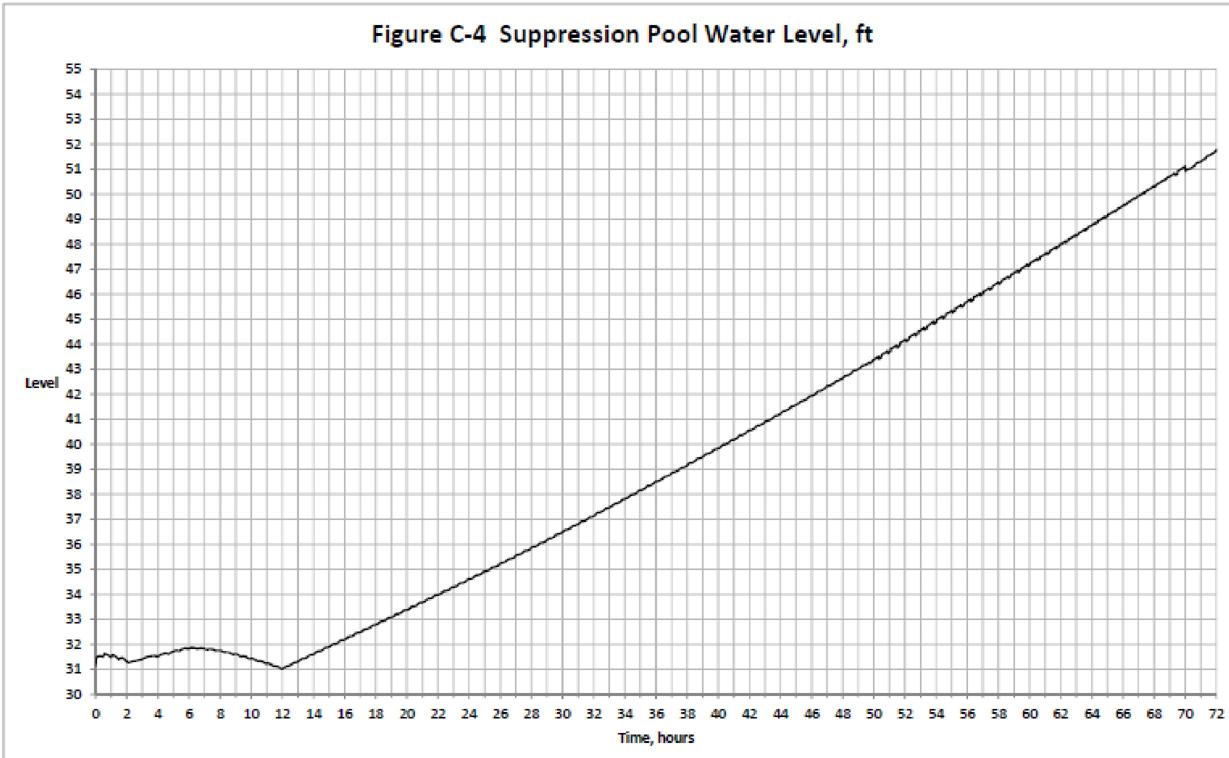
d. Wetwell Pressure

As shown in the following graph, wetwell pressure rises quickly to approximately 21 psia, then drops abruptly to near atmospheric pressure when the wetwell vent is opened at 6 hours. With the hardened containment vent (HCV) open, the cycling of the reactor pressure between 175 and 300 psig cycles wetwell pressure as steam is blown down to the suppression pool through the quenchers. The wetwell pressure peaks at about 28.4 psia (13.7 psig) at 14 hours. After that pressure decreases as decay heat generation decreases.



e. With 300 gpm makeup starting at 12 hours into the event, suppression pool level does not reach the wetwell vent penetration within the first 72 hours of the event.

   The suppression pool floor is at elevation 435' 3" and the vent penetration centerline is at 491 ft. The vent containment penetration is a nominal 12 inches in diameter. The bottom of the vent penetration is at elevation 490' 6". Suppression pool water level in MAAP is measured relative to the bottom of the wetwell making the bottom of the vent penetration 55' 3". In the graph below, beginning at 12 hours from the start of the event when makeup commences, the level in the wetwell rises at a relatively steady rate. After 72 hours the analysis shows that level in the wetwell has risen to approximately 51.8 ft. or about 3.7 ft. below the vent.



f. Drywell Pressure and Temperature

As shown in the graphs below, the drywell pressure rises quickly to a peak of 28.2 psia at approximately 17.5 hours, then stays between 27 and 28 psia for the remainder of the analyzed transient. Drywell temperature rises steadily, reaching 300°F at 72 hours.

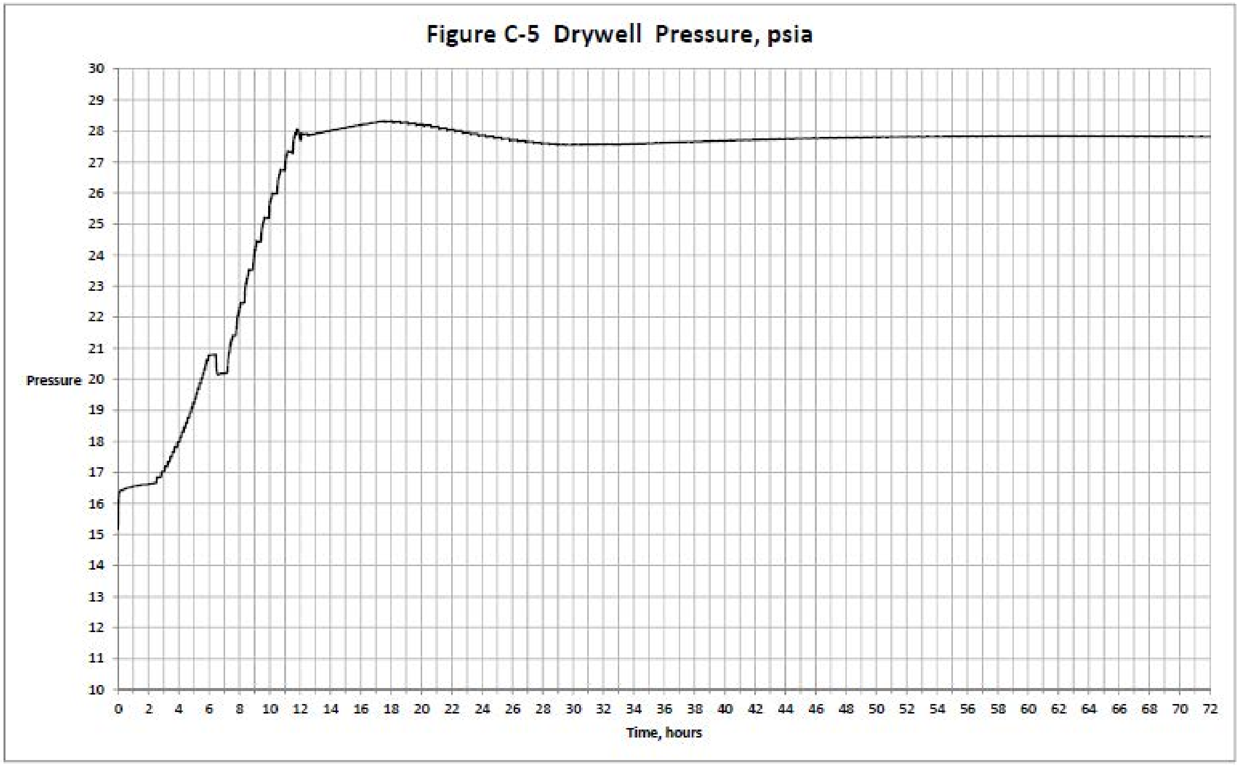


Figure C-5 Drywell Pressure, psia
Pressure
Time, hours

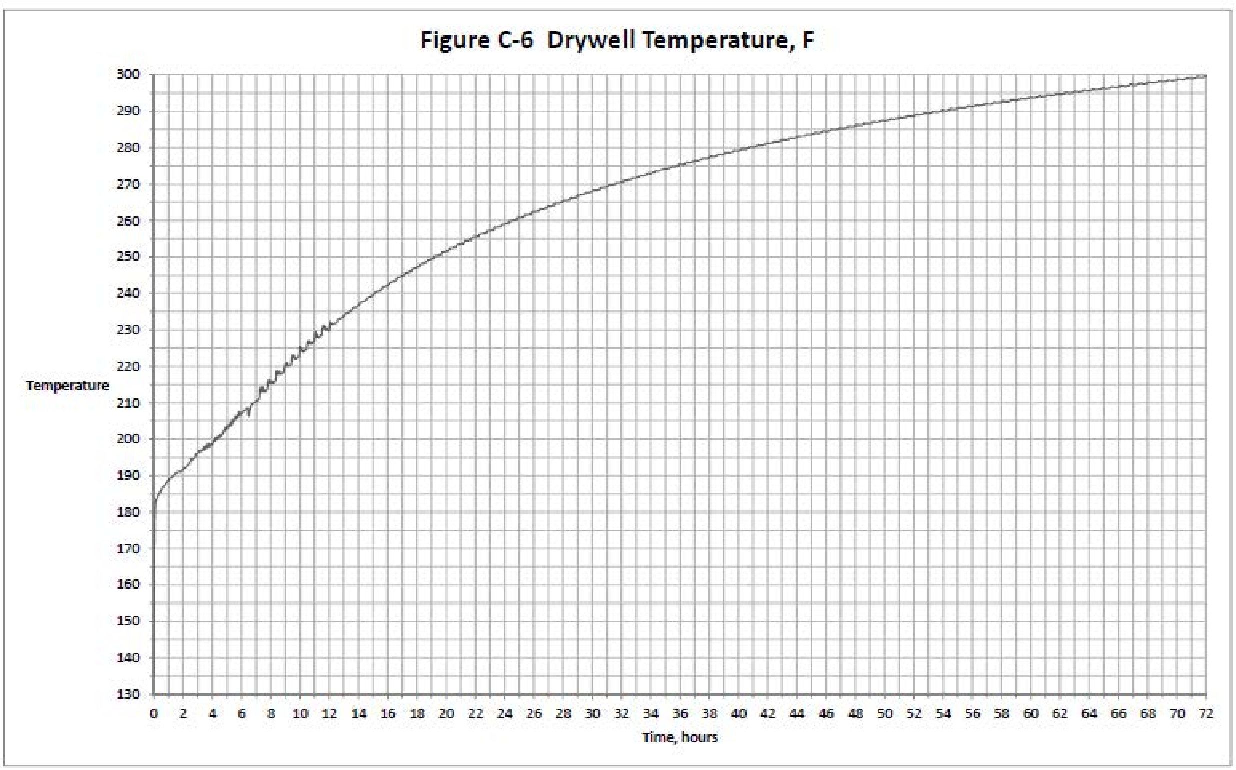


Figure C-6 Drywell Temperature, F
Temperature
Time, hours

3.1.8.2 Supporting Calculations

ME-02-12-06, *Evaluation of the Use of Portable Equipment during an Extended Station Blackout*

Provides flow requirements that assure acceptable temperatures in critical areas of the Reactor Building.

Scenarios 10 and 11 address the full-core offload to the SFP and Scenarios 12 and 13 address supplying cooling/makeup water to the SFP and RPV if the event occurs during normal operations.

This calculation demonstrates that the Godwin HL130M pump or the B5b pumper truck can provide the flow rates necessary for core cooling while the core is in the SFP or the RPV, including the reactor recirculation pump seal leakage.

Appendix D of the calculation discusses the reactor recirculation pump seal leakage which is shown not to be a concern considering the low expected value (5.4 gpm total for two pumps) and the large flow capability margin (135 gpm). RRC pump seal leakage is discussed in Section 3.1.6

ME-02-12-07, *Evaluation of RCIC Operation during an Extended Station Blackout*

This calculation confirms that the RCIC system can be operated for 72 hours with RPV pressure controlled between 175 and 300 psig with suppression pool temperature maximum 240°F. Concerns addressed in this calculation are (1) high Suppression Pool temperature, (2) compliance with piping design temperatures, (3) low RCIC pump flow, (4) maintenance of minimum RCIC turbine speed, and (5) maintenance of acceptable turbine lube oil temperature.

ME-02-12-18, *Containment Response During Extended Loss of AC Power (ELAP) - A Beyond Design Basis Assessment*

This calculation investigated the expected pressure-temperature response of primary containment to an ELAP. Numerous combinations of actions are considered to mitigate the potential effects of the loss of AC power. The actions considered are focused on preventing core damage, so accidents involving nuclear fuel damage are not addressed.

The results show that containment parameters support RCIC operation during an ELAP. The analysis concluded that RCIC operability is assured for the duration of the ELAP (72 hours) as long as suppression pool makeup is initiated within 40 hours to maintain RCIC pump NPSH. As discussed in ME-02-12-06 makeup to

the SFP begins in at least 12 hours. As discussed in the Phase 2 strategy in Section 3.1.2, this makeup cascades to the suppression pool.

<u>ME-02-14-02, *General Technical Support for Fukushima Related Licensing Documents, Appendix B*</u>

This calculation was completed in response to a February 2015 NRC audit question concerning the loss of spray pond volume during high winds. Appendix B states that the spray ponds contain over 12.5E06 gallons of water and that less than 6 percent of the available volume would be required to remove the decay heat from the reactor core and spent fuel pool during the first 72 hours of an ELAP event. The calculation shows that high winds are not expected to remove enough water to jeopardize the spray ponds' ability to serve as a source of water for the plant during an ELAP event.

<u>ME-02-14-09, *RCIC Suction Piping Hanger Analysis for Beyond Design Basis External Event*</u>

This calculation evaluates the impact of the thermal mode (250°F) on the associated piping supports. To avoid overloading of the RCIC-P-1 suction nozzle during a BDBEE, the struts for Hanger RCIC-967N will be disconnected prior to the suppression pool temperature reaching 170°F. The evaluation determined that hanger RCIC-967N is not required for BDBEE conditions.

<u>ME-02-14-12, *Cascading of Fuel Pool Overflow to the Suppression Pool during an ELAP Event*</u>

This calculation validates that the flow rate that can be established between the SFP and the suppression pool during an ELAP event with the only driving force being the elevation head is sufficient to facilitate SFP and suppression pool cooling.

<u>ME-02-14-13, *Containment Response during an Extended Loss of AC Power (ELAP) with Suppression Pool Makeup*</u>

This calculation determines the temperature and pressure response of the primary containment to an ELAP. It has been found necessary to provide 300 gpm cooling water to the SFP within 12 hours after loss of power in order to maintain accessibility in the reactor building. This calculation modeled the effectiveness of cascading flow to cool and replenish suppression pool inventory. A design criterion for the HCV system was established such that the flow resistance coefficient K, must be equal to or less than 4.6 (at 11.374" ID). In addition, primary containment must be vented within 6 hours after initiation of an

ELAP event in order to limit suppression pool temperature to less than 240°F for RCIC long term operation.

ME-02-15-04, *Potential Effects of Volcanic Ash in Spray Pond Water*

The analysis addressed the following issues raised during the February 2015 NRC audit. (1) the potential effect on components contacted by the water being pumped (hose, pipe, fittings, pumps, etc.), (2) the cooling of fuel in the RPV and the SFP, including the potential for plugging of coolant flow passages in the fuel assemblies, (3) accumulation of ash in the RPV and SFP as a result of addition of makeup from the spray ponds, and (4) potential for plugging of the FLEX pump suction strainers. The analysis showed that because of the very small size of the partials, there is no plugging or flow path restriction expected in the flow passages in the fuel assemblies in the SFP or RPV, nor in the pump suction strainer. As a result, no special actions are required for the protection of the spray ponds during or following ashfall to support the cooling of the fuel.

CVI 1201-00,1, *GOTHIC Analysis of CGS Radwaste Building Response to SBO*

This GOTHIC model provides the evaluation of SBO in the control room and vital island when repowering with a FLEX DG.

CVI 1201-00,2, *Reactor Building GOTHIC Temperature Analysis of during PSBO/ELAP*

This analysis addresses the temperature and humidity response throughout the reactor building due to the heat loads during an ELAP event. Mitigating actions analyzed included various makeup flows to the SFP and building ventilation actions.

3.1.9 Electrical Analysis

2.05.01, *Battery Sizing, Voltage Drop, and Charger Studies for Div. 1 & 2 Systems*

The scope of this calculation is for Divisions 1 and 2 battery systems and their associated DC buses which are simulated using ETAP 12.6.0N software. The ETAP software is used as an aid for the following studies/evaluations:

1. Battery sizing and margin determination
2. Battery charger sizing
3. Voltage analysis
4. Determination of the DC bus voltage during the design basis accident (DBA)

5. Determination of the duration for battery voltage to reach a predetermined value to support other calculations
6. Determination of the duration in which the battery voltage reaches its pre-determined minimum value during a Prolonged Station Blackout (PSBO)

CMR 13425 – *Incorporate ELA/Prolonged SBO Analysis into 2.05.01*

This change to the calculation defines load profiles for batteries E-B1-1 (125V Division 1) and E-B2-1 (250V Division 1 battery) when RCIC is aligned to either the condensate storage tanks (CSTs) or the suppression pool for injection water and confirms the battery run time capabilities for these profiles during the ELAP scenario in accordance with IEEE-485 as identified in the NEI whitepaper in ML13241A186 (Reference 36) and endorsed by the NRC in ML13241A188 (Reference 7).

ME-02-13-14, *Hydrogen Generation in Battery Rooms during ELAP (SBO)*

This analysis objective is to determine the hydrogen generation rate as a function of battery temperature in the battery rooms in the radwaste building during a beyond design basis ELAP. The analysis included both battery divisions and the batteries planned for the HCV system. The analysis confirmed that there is no hydrogen accumulation approaching flammability limits during battery charging.

CMR 13278 – *Calculation ME-02-13-14 REV 1 CALC Modification Record (CMR) in Support of EC 13094 - HCV Wetwell Addition*

This change is to update calculation ME-02-13-14 with the HCV design details (per Engineering Change (EC) 13094) which were previously included in Revision 1 of this calculation as preliminary.

CVI-1201-00,1 R1 Appendix D, *Repowering with DG4*

This portion of the analysis also looked at hydrogen concentrations and determined that the hydrogen concentrations are an acceptable 0.25% (concentration = 0.0025) at the end of the 72-hr transient.

E/I-02-91-03, *Calculation for Division 1 and 2 and 3 Diesel Generator Loading*, R19

This analysis shows that the 480-V ac FLEX generators have ample capacity to supply the key instrumentation identified in Section 3.1.7 while supplying the station battery chargers.

3.1.10 Procedures

PPM 5.6.1 *Station Blackout (SBO) and extended Loss of AC Power (ELAP)*

This procedure provides a flow chart of the actions required in an SBO or ELAP.

PPM 5.6.2, *Station Blackout (SBO) and Extended Loss of AC Power ELAP Attachments*

This procedure supports the performance of the PPM 5.6.1, "Station Blackout" flowchart.

PPM 12.14.1, *Chemical Treatment of Standby Service Water*

This procedure provides controls and documentation for chemical treatment of the Standby Service Water System (SSW). Chemical treatment consists of the addition of a corrosion inhibitor and biological inhibitors. Blowdown and makeup are performed as necessary to maintain the control limits.

PPM 1.20.3, *Outage Risk Management*

This procedure describes the process used to ensure shutdown safety while planning, assessing and implementing plant outages. This procedure establishes expectations that shutdown nuclear safety will incorporate the concepts of defense-in-depth and decay heat removal hardening.

PPM 10.2.222, *Seismic Storage Requirements for Transient Equipment*

This procedure provides instructions for proper installation/storage of transient equipment to prevent their damaging safety-related equipment during a seismic event.

PPM 1.5.18, *Managing B.5.B and FLEX Equipment Unavailability*

This procedure defines the method of documentation when Beyond Design Basis (BDB) event mitigation equipment is required to be removed from its designated location for maintenance, or is reported missing/degraded/non-functional.

OI-18, *Equipment Operator Rounds*

This procedure details the expectation of proper Operator tours.

OMI-3.2, *Shutdown Safety Plan Development and Approval Process*

This instruction provides the details surrounding the process for development, approval, revisions, and independent review of the shutdown safety plan while adhering to the defense-in-depth and safe shutdown philosophy described in PPM 1.20.3, *Outage Risk Management*.

ABN-ASH, *Ash Fall*

This procedure provides actions to respond to an ash fall event.

ABN-FSG-001, *Accessing Essential Instrumentation during Extended Loss of AC Power with no Power Available*

This procedure provides instructions on obtaining data from Essential Instrumentation during an ELAP with no DC power available.

ABN-FSG-002, *Water Makeup Strategies for RPV, SFP, DW, WW, CSTS during an Extended Loss of AC Power or other Beyond Design Basis Event*

This procedure provides instructions for various makeup water strategies.

ABN-FSG-003, *DG4 Crosstie to E-MC-7A and E-MC-8A*

This procedure provides instructions for connecting DG4 to either FLEX 480 volt connection point.

ABN-FSG-004, *DG5 Crosstie to E-MC-7A and E-MC-8A*

This procedure provides instructions for connecting DG5 to either FLEX 480 volt connection point.

ABN-FSG-NSRC-001, *NSRC 4160V DG Crosstie via DG-1, DG-3, or SM-3*

This procedure provides instructions for the use of the NSRC 4160-V DG if alignment is required.

ABN-FSG-NSRC-002, *NSRC Portable SW Pump Alignment to SW Loop A or SW Loop B*

This procedure provides instruction for the use of the NSRC pump if needed to provide SW flow to support plant needs.

ABN-FSG-NSRC-003, *NSRC 480V DG Crosstie to E-MC-7A and E-MC-8A*

This procedure provides instructions for connecting the 480 volt DG from the NSRC to either FLECX 480 volt connection point.

SOP-FLEX-EQUIPMENT-STORAGE, *FLEX Equipment Storage*,

This procedure provides storage locations and inventory details for FLEX equipment that is stored in storage locations inside or adjacent to Building 82 and Building 600, in the FLEX cabinet located in the clean tool crib on GSB 441', in the FLEX rigging cabinet on RB 471', and in the FLEX electrical storage cabinets on RW 467', TG 471' and TG 441'. This procedure also contains requirements for verifying wheeled FLEX equipment wheels are chocked to prevent interactions.

<u>SOP-FLEX-EQUIPMENT-REFUEL, *FLEX Equipment Refueling*</u>

This procedure provides details for the refueling of FLEX equipment during a beyond design bases external event that results in an ELAP.

<u>SOP-FLEX-FULL-CORE-OFFLOAD, *FLEX Activities to Support a Full Core Offload*</u>

This procedure identifies the required FLEX activities that must be performed to support a full-core offload. Additionally, the procedure describes the required actions that must be performed to support a BDBEE that results in an ELAP with full-core offload.

3.2 Spent Fuel Pool (SFP) Cooling/Inventory Strategy

The SFP strategy assumptions are discussed in Section 2.1.5.

The FPC system normally provides forced cooling of the SFP water. FSAR Table 9.1-6, *Bounding Fuel Pool Cooling Events*, describes a number of fuel pool heat load scenarios including normal refueling and full core off-load refueling. FSAR Table 9.2-5, *Heat Load Rates Used in Ultimate Heat Sink Analysis*, identifies the design heat load of the SFP with fuel in the RPV as 8.2E+06 Btu per hour. The maximum heat load of the SFP with a full core off-load is 44.3E+06 Btu per hour.

3.2.1 Phase 1 Strategy

No action is required to maintain SFP level or temperature during Phase 1. The heat capacity of the SFP water will absorb the heat from the stored fuel during Phase 1. The SFP time to 200°F is recalculated for each refueling. The current calculation, NE-02-17-02, R0, *Cycle 24 SFP Time-to-200°F*, shows that with no SFP cooling and maximum heat load, the time-to-200°F is more than 30 hours with a maximum starting temperature of 125°F.

3.2.2 Phase 2 Strategy

The reactor building area/room temperature/accessibility analyses shows that during an ELAP, in order to maintain good accessibility in the reactor building, 300 gpm cooling water to the SFP is needed within 12 hours after the loss of AC power.

As discussed in Section 3.1.2 and shown in Figure 1, make-up water will be supplied from the SW spray ponds to the SFP using one of the FLEX pumps and fire hoses. This lineup provides water directly into the SFP. If required, an oscillating spray nozzle can be used with the hose to spray water into the SFP.

If the refueling floor is inaccessible, an alternate path is available by connecting the supply hose to RHR-V-63B as discussed in Section 3.1.5.3.

3.2.3 Phase 3 Strategy

As stated in Section 3.1.3, equipment from the NSRC will be available to provide additional capability and redundancy to the on-site FLEX equipment. The Phase 2 strategy for adding water to the SFP can continue to be used during Phase 3.

3.2.4 Normal Refueling

The above SFP strategies apply during normal operations.

Procedure 1.20.3, *Outage Risk Management*, requires development of shutdown safety plan and procedure OMI-3.2, *Shutdown Safety Plan Development and Approval Process*, provides a section on FLEX equipment contingency planning which states that a contingency plan should be prepared each outage for pre-staging of FLEX equipment during times when time to boil is less than or close to the time to deploy FLEX equipment (i.e. during transition from Mode 4 to Mode 5 when the RPV head is not tensioned and RPV level is low).

3.2.5 Full-Core Offload

With a full-core offload, the increased heat load in the SFP significantly reduces the time to reach 200°F following an ELAP, and the refueling floor (reactor building 606 foot elevation) will eventually become inaccessible due to high temperatures and humidity. ME-02-14-07 indicates that there is at least 30 minutes available to complete actions on the refueling floor before wet bulb temperatures are too high for extended access.

Calculation ME-02-14-02 R1 determined that the water loss from the SFP during a full-core offload due to evaporation is initially approximately 100 gal/min. This makeup flow rate is well within the capacity of either FLEX pump as shown in Section 3.1.4.5. Without any makeup, the time before uncovering the fuel in the SFP is 34.6 hours.

If it is desired to maintain reactor building accessibility, GOTHIC analysis shows that with additional SFP makeup flow (up to 600 gpm) and certain ventilation enhancing actions, the floors below the refueling floor would remain moderate (< 104°F).

3.2.6 Systems, Structures, and Components

The discussions in Sections 3.1.4.4 through 3.1.4.7 also apply to the SFP makeup strategies.

3.2.7 Key SFP Parameters

Although reliable SFP level instrumentation is required by NRC Order EA-12-051, the Phase 1, 2, and 3 SFP strategies are time dependent and actions to implement the above strategies are not based on SFP level indication except during a full-core offload. During a full-core offload, if supplying less than 600 gpm to the SFP, makeup to the SFP is dependent on level. The key parameters are listed in Section 3.1.7, Key Parameters.

3.2.8 Mechanical Analysis

The mechanical analysis is discussed in Section 3.1.8,"Mechanical Analysis."

3.2.9 Electrical Analysis

The electrical analysis is discussed in Section 3.1.9, "Electrical Analysis."

3.3 Containment Integrity Strategy

It is assumed that the containment isolation actions delineated in current station blackout coping capabilities is sufficient. The strategies discussed in Section 3.3 apply in Modes 1, 2 and 3.

3.3.1 Phase 1 Strategy

As discussed above in Section 3.1.1 for Phase 1 core cooling and heat removal, the operator will reduce RPV pressure to between 175 psig and 300 psig using the SRVs at a rate not to exceed 100°F per hour.

If necessary, the operator performs anticipatory venting of the containment. Anticipatory venting of containment will relieve pressure to control suppression pool water temperature below 240°F and enable continued RCIC operation to provide reactor cooling. The venting strategy is consistent with the NRC endorsement of boiling water reactor (BWR) containment venting in Reference 6 via implementation of Boiling Water Reactor Owners Group (BWROG) Emergency Procedure Guideline (EPG)/Severe Accident Guideline (SAG), Revision 3.

3.3.2 Phase 2 Strategy

As discussed above in Section 3.1.2 for Phase 2 core cooling, a FLEX pump is used to provide a source of make-up water to maintain suppression pool level in order to provide sufficient NPSH for RCIC operation. Also, a FLEX DG will provide power to station battery chargers to ensure key instrumentation remains available to monitor containment parameters.

3.3.3 Phase 3 Strategy

This is the same as discussed above in Section 3.1.3.

3.3.4 Systems, Structures, and Components

The discussions in Sections 3.1.4.1 through 3.1.4.7 remain the same for the containment integrity strategies.

3.3.4.1 Hardened Containment Vent System

NRC Orders EA-12-049 and EA-12-050 required full implementation of the Order requirements no later than two refueling cycles after submittal of the overall integrated plan or December 31, 2016, whichever came first. However, the rescindment of Order EA-12-050 by Order EA-13-109 revised the schedule timelines for implementation of the containment venting system. Therefore, Energy Northwest requested (Reference 8) and received (Reference 9) relaxation of the initial due date for complete implementation of Order EA-12-049 as it applied to the reliable hardened containment vent. The overall integrated plan for Phase 1 of the hardened containment vent was submitted on June 30, 2014 (Reference 10). The overall integrated plan for the HCV was resubmitted on December 16, 2015, and included both Phase 1 and Phase 2 (Reference 11).

3.3.4.2 Containment

The containment consists of primary and secondary containment. The primary containment structure is a free-standing steel pressure vessel that contains both a drywell and a suppression chamber. The secondary containment structure is composed of the reactor building, which completely encloses primary containment. The primary containment employs the pressure suppression concept. The pressure suppression system consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting submerged vent system between the drywell and water pool, isolation valves, containment cooling system, and other service equipment.

Design Pressures and Temperatures

a. Pressure Suppression Chamber

Internal Design Pressure (LOCA pressure)	45 psig
Design Temperature	275°F

b. Drywell

Internal Design Pressure (LOCA pressure)	45 psig
Design Temperature	340°F

c. Pressure Suppression Chamber and Drywell

Pneumatic Over Pressure	51.8 psig at ambient

Test (115% of 45 psig)	temperature

Calculation ME-02-14-13 provides the containment response during an ELAP event (Section 3.1.8,"Mechanical Analysis."

3.3.5 Key Containment Parameters

Key containment parameters are discussed in Section 3.1.7.

3.3.6 Mechanical Analysis

Analysis in Section 3.1.8 confirms that:

- Wetwell pressure, suppression pool temperature, drywell pressure, and wetwell vent flow rate remain acceptable if containment venting is initiated within 6 hours.
- The effects of increasing suppression pool level due to cascading of SFP make-up for cooling are acceptable.
- Suppression pool temperature is maintained at or below 240°F, which assures RCIC functionality from the perspective of NPSH and bearing lube oil temperature.

3.3.7 Electrical Analysis

The electrical analysis is discussed in Section 3.1.9,"Electrical Analysis."

4.0 CHARACTERIZATION OF EXTERNAL HAZARDS

4.1 Seismic

Section 5.2 of NEI 12-06 (Reference 3), requires that all plants address seismic considerations in the implementation of FLEX strategies.

TM-2143, *Geology, Seismology, and Geotechnical Engineering Report*, which is incorporated by reference into FSAR Chapter 2.5, *Geology, Seismology, and Geotechnical Engineering*, describes the extensive geology, seismology, and foundation investigations conducted to establish a site-specific seismic design for Columbia.

During the operating license review stage, four earthquakes were evaluated in a deterministic fashion to confirm the original seismic design basis. The results of this review are documented by the NRC staff in the safety evaluation report for Columbia (NUREG-0892, Supplement No. 1). The Columbia design basis safe shutdown earthquake (SSE) is based on an approved response spectrum anchored at 0.25g (FSAR Section 3.7.1.1, *Design Response Spectra*). The operating basis earthquake (OBE) was assumed to be half of the SSE or 0.125g (FSAR Chapter 3.7, *Seismic Design*). Structures, systems, and components

related to plant safety are designed to withstand the effects of the safe shutdown and operating basis earthquakes. The methodology used to incorporate these values into the design of the site structures is described in FSAR Chapter 3.7.

As documented in FSAR Section 3.4.1.4.2, *Groundwater Protection Requirements*, soil liquefaction is not postulated at the Columbia site due to soil type and unsaturated conditions.

Further discussion is provided in Section 2.5.4.8 of TM-2143 which states, "The Columbia site is underlain by 45 to 60 feet of loose to medium dense, medium to fine grained sand. Below this stratum is a 200-ft-thick layer of very dense gravel (Ringold formation) having 'rock-like' engineering properties, which is underlain by interbedded layers of hard silt, clay, and gravel, extending to a conglomerated zone and basalt bedrock, the top of which is at a depth on the order of 525 feet. Based on the existing groundwater conditions at the site, no possibility of liquefaction of the soils underlying the site could result from motions associated with the SSE."

However, it was decided at the time of construction to remove the glaciofluvial sand down to the underlying very dense Ringold gravel and replace it in a denser condition by compaction.

Accordingly, the Seismic Category I structure foundations, including the service water spray ponds, are supported by the replacement backfill well above the stable groundwater elevation of about 378±4 feet msl, which is about 60 feet below grade (FSAR Section 2.4.13.1, *Description and Onsite Use*). As discussed in Section 4.2, *External Flooding*, liquefaction is not a concern at the Columbia site and the effects of a potential failure of a CW pipe, coincident with the ELAP, was considered to ensure that the FLEX storage areas are located such that deployment of at least one set of on-site equipment can be accomplished. It was determined that a CW pipe failure would not impede FLEX equipment deployment.

Section 5.3 of NEI 12-06 (Reference 3), contains additional requirements for plants relying on downstream dams. As documented in FSAR Section 2.4.11, *Low Water Considerations*, water levels at the Columbia River intake are not influenced by backwater from the downstream McNary Dam. Although the river intake provides make-up water to the UHS spray ponds, the combined water volume of the spray ponds is over 12 million gallons, which is adequate to provide cooling water for 30 days without make-up (FSAR Section 9.2.5.2). Therefore, the Columbia UHS water supply is not impacted by failure of a downstream dam.

The *Fukushima SAFER Response Plan for Columbia Generating Station* (Reference 13) contains information on routes and contingencies to move the equipment from the NSRC to Columbia. Within the Columbia owner controlled area there are two paved access roadways (i.e., one normal and one secured alternate access). There are no bridges or dams on the site that could compromise site access following an earthquake.

Specifically, the following considerations have been applied to the FLEX equipment with respect to seismic:

- Structures used for FLEX equipment storage meet the plant's seismic design basis or in structures designed or evaluated equivalent to the American Society of Civil Engineers (ASCE) standard ASCE 7-10, *Minimum Design Loads for Buildings and Other Structures*, with a Building Risk Category of IV. For storage in the latter, the building(s) are equipped with a means to provide backup power.
- Equipment normally stored in structures has been evaluated to protect it from seismic interactions between components.

  Procedure SOP-FLEX-EQUIPMENT-STORAGE contains requirements for verifying wheeled FLEX equipment is chocked to prevent interactions. Procedure 10.2.222, *Seismic Storage Requirements for Transient Equipment*, provides guidance on maximum allowed sliding distances for stored items.
- Equipment normally stored outside has been evaluated for seismic interactions to ensure equipment is not damaged by non-seismically robust components or structures.
- At least one connection point for the FLEX equipment requires access only through seismically robust structures including both the connection point and any areas that plant personnel will have to access.
- The means to move FLEX equipment is reasonably protected.
- The procedural interface described in NEI 12-06 (Reference 3), Item 1 (alternate instrument readouts) is addressed in Section 3.1.7 above.
- The effects of a potential failure of a CW pipe, coincident with the ELAP, have been addressed as described above.

Columbia screens in for a seismic probabilistic risk evaluation (PRA) and submitted the Expedited Seismic Evaluation Process (ESEP) Report in Reference 21. In Reference 22, Energy Northwest provided additional information. In Reference 23, the NRC grouped Columbia into Group 1 for

completion of the seismic PRA. Any impact resulting from the completion of the ESEP or seismic PRA that affects the adequacy or feasibility of FLEX strategies will be included in the Program Document when completed. The Probabilistic Seismic Hazard Analysis (PSHA) and a Ground Motion Response Spectra (GMRS) were finalized and the seismic re-evaluation was submitted to the NRC on March 12, 2015 (Reference 24). In References 25 and 26, the NRC requested additional information to support Columbia's seismic reevaluation. Energy Northwest provided the response to Information Requests 1 through 4 in Reference 27. Information Request 5 was responded to in Reference 28. In Reference 29, the NRC concluded that the Columbia reevaluated seismic hazard is suitable for other actions associated with Near-Term Task Force Recommendation 2.1, "Seismic" and provided an acceptable response to Requested Information Items (1) - (3) and (5-7), and the comparison portion to Item (4), identified in Enclosure 1 of the 50.54(f) letter. In reaching this conclusion, the NRC staff confirmed that Columbia's GMRS exceeds the SSE at the Columbia site. As such, Energy Northwest will perform a seismic risk evaluation, SFP evaluation, and high frequency confirmation.

4.2 External Flooding

Section 6.2.1 of NEI 12-06 (Reference 3), requires plants that are not dry sites to perform a flood-induced challenge evaluation.

A dry site is one that is built above design basis flood level. Columbia is built above the design basis flood level of 433.3 ft. msl. As documented in FSAR Section 2.4.10, *Flooding Protection Requirements*, the approximate finished grade of all Seismic Category I structures, except the spray ponds, is at elevation 440 feet msl. The finished grade of the spray ponds is 434 feet msl.

As documented in FSAR Section 2.4.3, *Probable Maximum Flood on Streams and Rivers*, the Probable Maximum Flood (PMF) elevation of the Columbia River at the site is estimated to be 390 feet msl. As documented in FSAR Section 2.4.3.6, *Coincident Wind Wave Activity*, the flood elevation due to the Probable Maximum Precipitation (PMP) (including surge and wave run-up) is 433.3 feet msl.

Therefore, Columbia is built above the design basis flood level and is not required to evaluate flood-induced challenges for the protection of FLEX equipment. FLEX equipment storage is above elevation 433.3 feet msl. FLEX equipment deployment route(s) from one FLEX building (Building 600) may be affected by the current licensing basis PMP and is discussed in Section 6.0, *Planned Deployment of FLEX Equipment*. However, the ability to deploy

equipment from the other FLEX building (Building 82) would not be affected. A revised flood hazard analysis was completed and submitted on October 6, 2016 (Reference 12) and did not identify any impact to the current FLEX mitigation strategies.

TM-2184 *Evaluate Circulating Water (CW) Pipe Break with Respect to Deployment of FLEX Strategies*, evaluated the usability of the deployment route locations for FLEX equipment following a seismic event and from the effects of a CW line break following that seismic event. This evaluation is based on FSAR Section 3.4.1.2 where liquefaction was not postulated at the site based on its soil types, the backfill for the CW piping following Quality Class 1 and 2 requirements, the ground water table at approximately 380 ft. and the centerline elevation of the CW piping at 421 ft.

The deployment route locations as well as the underground portion of the CW piping system have been installed/constructed using well graded compacted granular material above the top of the water table. The deployment route locations, as well as the underground CW piping, reside in unsaturated soil conditions. The conditions in which the deployment route locations as well as the CW piping have been installed do not meet the requirements to be susceptible to liquefaction. Liquefaction issues are not a concern to the deployment of FLEX equipment from the initiating seismic event that signifies the start of an ELAP.

In the event the underground CW piping experiences a localized failure where water leakage occurs, there exists enough water volume and hydrostatic pressure head in the CW piping system to saturate the surrounding soil. Due to the well graded nature of the backfill and without a vibratory load input, assuming no seismic aftershocks, liquefaction will not be a concern and soil stability will be maintained. However, depending on the size of the postulated failure and where the failure location is, a small degree of erosion at the ground surface near the failure location could be expected. This level of erosion can be easily and quickly repaired by the dedicated debris removal equipment and the FLEX deployment route restored for usability to support Phase 2 of the mitigation strategies.

Therefore, the usability of the deployment route locations for FLEX equipment from the effects of a CW line break following a seismic event will be maintained.

The NRC requested licensees to re-evaluate all appropriate external flooding sources, including the effects from local intense precipitation on the site, probable maximum flood (PMF) on streams and rivers, storm surges, seiches, tsunami, and dam failures (Reference 30). The NRC requested that the re-evaluation apply present-day regulatory guidance and methodologies.

With the information transmitted in Reference 31 that provided U.S. Army Corps of Engineering information, Energy Northwest completed Columbia's Flooding Hazard Reevaluation Report (HRR) (Reference 12) which reported the results are either bounded by the current design basis or available physical margin exists. For flood causing mechanisms that are not described in the FSAR and the water surface elevation exceed the critical elevation of 441 foot-mean sea level, the results are inconsequential and do not compromise safety-related equipment.

In Reference 32, the NRC provided a summary of the staff's assessment of the reevaluated flood-causing mechanisms and concluded that Columbia's reevaluated flood hazard information is suitable for the assessment of mitigating strategies developed in response to Order EA-12-049. Further, the NRC staff concluded that Columbia's reevaluated flood hazard information is a suitable input for other assessments associated with Near-Term Task Force Recommendation 2.1, "Flooding." The mitigating strategies assessment (MSA) is scheduled to be completed by January 31, 2018.

4.3 Severe Storms with High Winds

Section 7.2.1 of NEI 12-06 (Reference 3), contains a screening process to identify whether sites should address high wind hazards as a result of hurricanes and tornadoes.

FSAR Section 2.1.1.1, *Specification of Location*, states that the reactor is located at 46° 28' 18" North latitude and 119° 19' 58" West longitude. Using NEI 12-06, Figures 7-1 and 7-2, Columbia screens out for both hurricanes and tornados for the protection and deployment of FLEX equipment. Storage of FLEX equipment is either in structures meeting Columbia's design basis for wind, or structures designed or evaluated to be equivalent to ASCE 7-10. Equipment stored outside has been evaluated for severe storms and high winds.

4.4 Ice, Snow, and Extreme Cold

Section 8.2.1 of NEI 12-06 (Reference 3), requires all plants consider the temperature ranges and weather conditions for the site in storing and deploying the FLEX equipment. As documented in FSAR Chapter 9.4, *Heating, Ventilation, and Air Conditioning Systems*, the winter outdoor design temperature is 0 °F with an extreme outdoor winter condition of -27°F. FSAR Section 2.3.1.2.2, *Design Snow Load*, describes that a value of 20 pounds per square foot was used as the design snow and ice loading for Columbia structures.

Columbia is located above the 35th parallel; thus, snow removal equipment is required. FSAR Section 2.3.1.2.1*, Heavy Rain, Snow, and Ice*, documents the

record snowfalls for the site as follows: (1) the greatest 24-hour snowfall is 10.2 inches, and (2) the highest number of days with greater than 12 inches of snow on the ground is 9 days. As identified in Figure 8-2 of Reference 3, *Record 3-Day Snowfalls*, Columbia is located in the yellow region (Level 3) and must consider ice storm impacts (i.e., low to medium damage to power lines and/or existence of considerable amount of ice).

Specifically, the following considerations were applied to the FLEX equipment with respect to cold temperatures, snow, and ice:

- Storage of FLEX equipment is either in structures meeting Columbia's design basis for snow, ice, and cold conditions or in structures designed or evaluated as equivalent to ASCE 7-10, Building Risk Category IV.
- Equipment was procured to function in the cold weather conditions applicable to Columbia and is maintained within a temperature range to ensure it will function when called upon.
- Snow removal equipment is available onsite including a large front wheel loader which is part of the FLEX support equipment available for debris removal and clearing of snow from deployment pathways.
- Under extreme cold conditions coincident with the ELAP, the surface of the SW spray ponds could freeze. Actions have been developed to ensure the continued availability of the water inventory from these sources.

4.5 High Temperature

Section 9.2 of NEI 12-06 (Reference 3), requires all plants to consider the high temperature conditions for the site in storing and deploying the FLEX equipment.

As documented in FSAR Chapter 9.4, *Heating, Ventilating, and Air Conditioning Systems*, the summer outdoor design temperature for Columbia is 105 °F (dry-bulb) with an extreme outdoor summer condition of 115°F (dry-bulb). Specifically, the following considerations were applied to the FLEX equipment with respect to high temperatures:

- Equipment was procured to function in the hot weather conditions applicable to Columbia, and
- Equipment is maintained within a temperature range to ensure it is likely to function when called upon.

4.6 Site-Specific Hazards

4.6.1 Volcanic Ash

There are several major volcanoes in the Cascade Range west of the Columbia site. The closest is Mount Adams approximately 165 km distant; the most active is Mount St. Helens approximately 220 km west-southwest of the site. The guidance in Reference 3 acknowledges that forest fires, grass fires, lightning, sandstorms, and volcanic hazards are considered to be enveloped by baseline coping strategies of an ELAP. However, for Columbia, the design basis ash fall criterion was applied during the process of developing the site-specific FLEX capabilities.

Because most volcanic activity is confined to the immediate area of the volcano, mud flows, avalanches, pyroclastic rock flows, lava flows, and shock waves that may be associated with such activity do not pose a hazard to the site. The only potential hazard to the site is ash fall resulting from a major eruption of one of these volcanoes.

Appendix B of NEI 12-06 (Reference 3), notes that some hazards may contribute to the potential for a simultaneous ELAP and LUHS but environmental conditions do not significantly challenge the structures and internal plant equipment. The NEI guidance acknowledges that forest fires, grass fires, lightning, sandstorms, and volcanic hazards are considered to be enveloped by baseline coping strategies of the ELAP. For Columbia, the design basis ash fall criteria have been applied as appropriate during the process of developing the site-specific FLEX capabilities.

Specifically, the following considerations have been applied to the FLEX equipment with respect to ash fall:

- FLEX mitigation equipment is stored in locations capable of withstanding the ash fall hazards applicable to the Columbia site such that no one external event can reasonably fail the site FLEX capability and
- Deployment of equipment has been evaluated to ensure manual actions required by plant personnel can be accomplished under ash fall conditions.
- Oil bath filters are available in the FLEX buildings for deployment with the equipment. Plant procedure ABN-ASH, Attachment 7.4, provides instructions for installing the oil bath filters on:
  - DG4
  - DG5

- FLEX-P-1
- B5b Pumper Truck
- Building 600 House Generator
- Building 82 House Generator

During the ELAP, the SW spray ponds are relied upon for cooling/makeup water. The ashfall does not inhibit the access or use of the spray pond water.

Additionally, the various potential effects of ash fall are evaluated and discussed in Section 3.1.8, "Mechanical Analysis."

5.0 PROTECTION OF FLEX EQUIPMENT

In accordance with NEI 12-06 (Reference 3), if on-site FLEX equipment is pre-staged such that it minimizes the time delay and burden of hook-up following an external event, then the equipment will be evaluated to not have an adverse effect on existing SSCs. Otherwise, FLEX equipment will be stored in one or more of following three configurations such that no one external event can reasonably fail the site FLEX capability (N):

1. In a structure that meets the plant's design basis for the Safe Shutdown Earthquake (SSE) (e.g., existing safety-related structure).
2. In a structure designed to or evaluated equivalent to ASCE 7-10, Minimum Design Loads for Buildings and Other Structures.
3. Outside a structure and evaluated for seismic interactions to ensure equipment is not damaged by non-seismically robust components or structures. See Section 4.1.

Large FLEX equipment such as pumps and power supplies are secured as appropriate to protect them during a seismic event. See Section 4.1 for additional information.

5.1 FLEX Buildings

Two dedicated FLEX buildings are used to provide diverse storage locations. Building 82 is an approximately 4,700 square foot structure located in the protected area south of the DG building. Building 600 is an approximately 9,400 square foot structure located outside the protected area east of the plant. Both FLEX buildings are fully covered by sprinklers and are non-combustible structures with occupancy Type S-1 and S-2 (low to moderate hazard storage space).

Each building has an air conditioned area for storage of temperature-sensitive equipment. The remaining building areas are unconditioned storage space.

Both Building 82 and Building 600 include a permanently installed diesel-powered generator for the primary purpose of carrying building house loads during a power outage.

The house loads are those components, such as lighting, and battery chargers necessary to support various activities in an emergency. Other than the above identified loads, the FLEX building house generators are not used to support mitigation actions. Power is not required to deploy equipment, as all the doors can be manually operated. The list of major Phase 2 on-site FLEX equipment, its expected storage location, and intended use is provided in FLEX-01 Appendix G, Major On-Site FLEX Equipment.

The technical basis for design and acceptability of the equipment storage locations, FLEX Buildings 82 and 600, has been documented to meet the requirements of ASCE 7-10 and are designed to be structurally capable of withstanding wind loading and ash fall deposit.

The location of the FLEX buildings is shown on Figure 2*, FLEX Deployment Routes*.

Considerations have been given to the transport from the storage area following the external event recognizing that external events can result in obstacles restricting normal pathways for movement.

As stated in Section 4.1, soil liquefaction is not postulated at the Columbia site due to soil type and unsaturated conditions.

6.0 PLANNED DEPLOYMENT OF FLEX EQUIPMENT

Deployment of the FLEX equipment or debris removal equipment from storage locations does not depend on off-site power or on-site emergency AC power (e.g., to operate roll up doors, lifts, elevators, etc.).

Both FLEX buildings 82 and 600 are above flood levels and designed to withstand all site specific external hazards.

The normal access road to Building 600 is located below the plant design basis flood plain (elevation 433.3 feet) and the PMP flood plain elevation of 433.3 feet msl and may be unavailable during a site flood event. The equipment stored in Building 600 can be deployed during flood conditions by removing a portion of the vehicle barrier system to create an alternate route to the plant area that is above the design basis flood level. When the mitigating strategies assessment (MSA) is completed, a determination will be made whether it will be necessary to

pre-deploy FLEX equipment in the event of forecast of extreme precipitation which might cause local flooding that would complicate deployment during the precipitation event.

6.1 Deployment Routes

As stated in Section 4.2, FLEX equipment storage will be at or above the design bases flood level of elevation 433.3 feet msl.

As stated in Section 4.1, a CW pipe failure will not impede FLEX equipment deployment.

As Columbia occasionally has snow, plans already are in place to remove ice and snow and can be used to remove snow and ice from the FLEX equipment deployment routes shown in Figure 2.

Debris removal equipment includes a wheel loader and an excavator. Towing equipment includes a flatbed truck and a 5-yard dump truck. The towing and debris removal equipment were purchased with block heaters.

The flatbed truck was procured with a mobile 50 gallon gas/50 gallon diesel emergency fuel dispensing system. This tow vehicle can also be used for fuel delivery.

As stated in Section 4.1, soil liquefaction is not postulated at the Columbia site due to soil type and unsaturated conditions.

6.2 Debris/Snow Removal

See the discussion in 6.1 above.

6.3 Deployment of a FLEX Pump

An on-site FLEX pump will be deployed to one of the SW spray ponds (A or B) which are located southeast of the reactor building.

The deployment of either FLEX pump includes non-collapsible suction hose allowing the pump to be placed near either SW pond.

The suction hose includes two floating suction strainers with a Y-connection and isolation valves in the suction path. This will allow cleaning of one strainer while pumping through the second strainer, preventing the loss of the FLEX pump suction.

The discharge hose routing from the pump is as described in Section 3.1.2. The excavator can be used to break ice on the spray pond if it occurs.

6.4 Deployment of a FLEX Generator

Prior to the depletion of the station batteries at approximately 8 hours, one of the on-site FLEX generators will be made available to power the station battery chargers.

DG4 is normally located at its deployment site. DG5 will be deployed to one of the two locations identified in Attachment A, Figure 2. Both on-site FLEX generators have ample cabling so either FLEX generator can be connected to either connection point discussed in Section 3.1.4.6 above.

6.5 Fueling of FLEX Equipment

Columbia has developed two Flex equipment refueling strategies. Both strategies rely on the fuel oil transfer equipment stored in each FLEX building. Each FLEX building contains a small fuel supply stored in the flammable storage cabinets. The 30-gallon gasoline drum is reserved for fueling one fuel transfer pump for 72-hours. Procedure SOP-FLEX-EQIPMENT-REFUEL contains instruction for extracting fuel and determining the delivery times to the credited FLEX equipment. The procedure also identifies the location and capacities of the various fuel oil sources.

Refueling is estimated to begin after supplemental staffing arrives on site (estimated to be 6 hours from declaring an event) as FLEX equipment is deployed with sufficient fuel to operate for 10-hours. Credited equipment is refueled beginning 8 hours after being placed in service and eight hours thereafter. Vehicles to deploy and refuel the FLEX equipment are available on-site.

TM-2185, *Equipment Refueling Strategy of Phase 1 and 2 FLEX Components During the First 72 Hours Following an ELAP*, shows the FLEX building locations, fuel storage tanks distribution locations, and site access roads to be cleared and used for fuel vehicle distribution routes. These are shown in Figure 2. In the event that roads are inaccessible, mobile fuel carts and cans will be used to manually transport fuel to FLEX equipment based on priority of mission. Attachment E of TM-2185 provides fuel consumption estimates for credited equipment of 325 gallons every 8 hours. This equates to approximately 6,825 gallons of diesel fuel needed for a 7-day coping period.

7.0 ON-SITE AND OFF-SITE RESOURCES

7.1 On-site Resources

The table in Section 3.1.2 identifies the major on-site FLEX equipment used in the Phase 2 strategies. The FLEX equipment stored in the FLEX building is inventoried in accordance with SOP-FLEX-EQUIPMENT-STORAGE.

7.2 Off-site Resources and the National SAFER Response Center (NSRC)

To meet the requirements of Section 12.2 of the NEI guidance, a SAFER team, an alliance between AREVA and Pooled Equipment Inventory Corporation (PEICo), was established. The SAFER team is contracted by the nuclear industry through PEICo to establish NSRCs operated by Pooled Inventory Management (PIM) and in collaboration with AREVA to purchase, store, and deliver emergency response equipment in the case of a BDBEE in the U.S.

CVI 1228-00,10 (Reference 13), is the SAFER Response Plan for Columbia Generation Station and sets forth the overall plan to establish the means to ensure necessary resources will be available from off-site.

This document is intended to define the SAFER team and Columbia actions to ensure successful activation, delivery, and operational status of the equipment required by Columbia to ensure indefinite coping capability in the event of a BDBEE as described in NEI 12-06 (Reference 3) and EA-12-049 (Reference 2).

To ensure a comprehensive approach to the off-site response, requirements for six functional areas have been established within Columbia's SAFER response plan (Reference 13):

- SAFER Control Center
- National SAFER Response Center
- Logistics and Transportation
- Staging Area
- Site Interface Procedure
- Equipment Requirements

In the event of a BDB event, equipment will be moved from an NSRC to a local assembly area identified in Reference 13 which is the parking area along side of the Columbia Training Center (see Figure 2A). The plan contains information on routes and contingencies to move the equipment to Columbia. The NSRC equipment will be flown to either Seattle or Portland airports and then moved by truck to the Columbia site. Contingencies are also in place for helicopter

transport from a staging area at Connell City Airport to the Columbia site. The Columbia SAFER Response Plan contains an abnormal conditions checklist to be used if the following were to occur:

- Routes become blocked and/or detour/road construction.
- Vehicle malfunction/wreck.
- Inclement weather

8.0 HABITABILITY AND OPERATIONS

The guidance in NEI 12-06 (Reference 3), states that the mitigation strategies must be capable of execution under the adverse conditions (unavailability of installed plant lighting, ventilation, etc.) expected following an ELAP event. Following an ELAP event, ventilation providing cooling to occupied areas and areas where FLEX equipment may be located will be lost. The primary concern regarding the loss of ventilation is the heat buildup which occurs in areas that continue to have heat loads.

The GOTHIC analyses discussed in Sections 3.1.8 and 3.1.4.1 were performed to determine the temperatures expected in specific areas related to FLEX implementation during the first 72 hours of the ELAP event to ensure the environmental conditions remain acceptable for personnel habitability and within equipment qualification limits. If it is expected that temperatures in these areas may become limiting after 72 hours, the opposite division of the equipment can be used, or available NSRC equipment can be used to provide additional cooling. The use of the NSRC equipment for this purpose is contained in procedures ABN-FSG-NSRC-001, *NSRC 4160V DG Crosstie via DG-1, DG-3, or SM-3* and ABN-FSG-NSRC-002, *NSRC Portable SW Pump Alignment to SW Loop A or SW Loop B*.

8.1 Equipment Operating Conditions

The key areas identified for all phases of execution of the FLEX strategy activities are in the radwaste building, which includes the vital island and the control room, and in the reactor building, which includes the refueling floor and RCIC room. These areas have been evaluated to determine the temperature profiles following an ELAP event. Actions for maintaining acceptable temperatures in these areas are identified in procedures.

With the exception of the control room, none of the key areas exceed the Licensee Controlled Specification (LCS) temperature limits during the 72-hour SBO transient. Compensatory actions for the control room are prescribed in Plant Procedure Manual (PPI) 5.6.2, *Station Blackout and Extended Loss of AC Power ELAP Attachments*.

As stated in Section 3.2.2 of NEI 12-06 (Reference 3), Guideline 12, the effects of a loss of heat tracing used to ensure cold weather conditions must not result in freezing important piping and instrumentation systems should be evaluated. The primary source of water for Phase 1 FLEX strategy is the suppression pool. Therefore, no specific action is required to compensate for a loss of heat trace during ELAP. Equipment to break SW pond ice is available, if needed, to implement the Phase 2 strategy.

On-site FLEX equipment has been procured to function in the conditions applicable to Columbia and includes block heaters. The towing and debris removal equipment were purchased with block heaters.

8.2 Personnel Habitability

As discussed in Section 3.1.2, *Phase 2 Strategy*, analysis shows that good accessibility in the reactor building can be maintained with a combination of SFP makeup and opening a natural convection pathway. However, if access to the refueling floor is not possible, makeup to the SFP can be supplied using RHR-V-63B.

Habitability of the control room is maintained by implementing the actions described in PPM 5.6.2.

8.3 Lighting

The standard equipment carried by operators with duties in the plant (i.e., outside the control room) includes flashlights. The requirement to carry flashlights is currently specified in procedures OI-18, *Equipment Operator Rounds*, and PPM 1.3.1, *Operating Policies, Programs and Practices*. Lighting for the control room will be maintained throughout the ELAP event by the DC-powered control room emergency lighting system. Although not credited for the FLEX strategies, lighting that meets the10 CFR 50 Appendix R requirements also provides emergency lighting initially in select areas of the plant where operators or maintenance personnel may need to perform actions, during ELAP conditions. This lighting has batteries that last for a minimum of 8 hours. The FLEX buildings contain portable lights and a stock of flashlights and head lights to further assist the staff responding to an ELAP event during low light conditions.

An evaluation of the time-critical tasks was performed and included the available lighting in the designated task areas. Tasks evaluated included traveling to/from the various areas necessary to implement the FLEX strategies, making required mechanical and electrical connections, performing instrumentation monitoring, and component manipulations. The available lighting was found to be adequate.

Additional portable lighting fixtures (Portable Light 2000 watts) are available in each FLEX Building

8.4 Communications

The primary means of onsite and offsite communication to be used during an ELAP event at Columbia are the installed sound powered telephone and radio systems. It has been determined that the indoor and outdoor locations where plant equipment or on-site FLEX equipment may be used, can be communicated with by using sound-powered phone headsets, satellite phones, or hand-held radios in radio-to-radio mode. The sound powered phone system can still be used if and when other communication methods become unavailable.

Sound Powered Phones

In addition to the installed sound powered phone system, five portable sound powered phone kits are available in each FLEX building. The kits are available to provide point-to-point communication to areas that have lost the other forms of communication. They can also be used to extend the current sound powered phone system. Each kit contains 800 feet of cable, two headsets, and junction boxes. The junction boxes allow multiple kits to be attached together to provide longer cable runs or to allow additional headsets to be connected.

Satellite Phone System

Each portable satellite phone is battery powered. Three batteries have been allocated to each phone with each battery providing 4 hours of talk time. The batteries are rechargeable and require a 4 hour charging time for a drained battery. This allows one battery to be used in the phone, one battery to be carried with the phone, and one battery to be charging. The battery chargers for the satellite phones are stored in the same locations as the phones. These chargers can be powered by the FLEX building standby generators.

Base stations are located in the control room, technical support center (TSC), emergency operations facility (EOF), alternate EOF and joint information center (JIC).

The base stations are powered from automobile type batteries located in the FLEX buildings for the control room and TSC, in the telecom room for the EOF and in the APEL/TEC facility for the alternate EOF and JIC. These batteries are good for approximately 24 hours on a loss of power.

The chargers for the base station batteries are powered by normal site power with the respective facility's back-up generators providing the emergency power.

Hand-held Radios

It has been determined that the mounting of some radio system components do not meet the seismic requirements necessary to assure radio system availability following an earthquake. However, hand-held portable radio-to-radio capability will exist for use.

The radios are powered by rechargeable batteries. Spare batteries and the charges are located in FLEX building 82. Radio battery life is heavily dependent upon the amount of talk time. Three batteries have been allocated to each radio. Battery charging can be accomplished in FLEX Building 82.

8.5 Staffing

In December 2011, the NRC added paragraph A.9 to Section IV of 10 CFR 50 Appendix E that required licensees to perform a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned function as specified in the emergency plan. NEI 10-05 (Reference 15) contains the guidance for completing the analysis.

In NEI 12-01, Section 1.3.1, *Staffing*, NEI provided requirements for performing staffing assessments. This guidance stated that for single-unit plants like Columbia, the staffing assessment required by the recent emergency plan rule satisfied the Phase 1 staffing assessment. In letter GO2-12-069 (Reference 16), Energy Northwest stated that it would provide an onsite and augmented staffing assessment considering functions related to NTTF Recommendation 4.2 by December 28, 2014. The results of the Phase 2 staffing study were provided to the NRC in Reference 17 and summarized below.

- The analysis was conducted in accordance with the guidance in NEI 12-01.
- The on-shift organization, as defined by the Emergency Plan, Fire Protection, and Operations shift staffing procedures, was assessed.

The personnel that are assumed to be on-site during the BDBEE are part of the minimum complement required by the Columbia Emergency Plan (Reference 14).

- The draft FLEX Support Guidelines (FSGs) in place as of November, 2014 and draft revisions to Operations procedures were used for the assessment.
- The assessment considered required actions performed during the initial and transition phases of an ELAP (first 24 hours).

- The assessment concluded that sufficient on-shift resources are available at all times to implement the strategies developed to maintain or restore core cooling, containment integrity, and spent fuel pool cooling during a BDBEE that results in an ELAP.
- The difference between the draft FSGs and final FSGs was evaluated and determined not to adversely affect the staffing assessment.

9.0 SHUTDOWN AND REFUELING ANALYSIS

Cold Shutdown and Refueling:

Strategies for mitigating an ELAP and LUHS event during Modes 4 and 5 incorporate the supplemental guidance provided in the NEI position paper entitled *Shutdown/Refueling Modes* (Reference 18). Plant procedures have been revised to incorporate this guidance to enhance the shutdown risk process. In Reference 19 the NRC endorsed the NEI position paper.

During Modes 4 or 5, plant procedure 1.20.3, *Outage Risk Management*, states that FLEX equipment should remain available during outages to mitigate beyond design basis external events, including an ELAP. Due to the unique conditions of each refueling outage, the specific FLEX strategies to be employed and equipment to be pre-staged will be addressed in the FLEX equipment contingency plan of the shutdown safety plan (SDSP).

- During plant conditions where deployment of FLEX equipment would take longer than the time for plant conditions to degrade to an unacceptable level (such as times of low vessel inventory with high decay heat during the transition from Mode 4 to Mode 5), contingency plans should be developed to pre-stage FLEX equipment.
- In cases where FLEX equipment would need to be deployed in locations that would quickly become inaccessible as a result of a loss of decay heat removal from an ELAP event, prestaging of that equipment is required.

As discussed in plant procedure OMI-3.2, *Shutdown Safety Plan Development and Approval Process*, a contingency plan should be prepared each outage for pre-staging of FLEX equipment during times when time to boil is less than or close to the time to deploy FLEX equipment (i.e. during transition from Mode 4 to Mode 5 when the RPV head is not tensioned and RPV level is low). FLEX pumping deployment is nominally 4 hours.

Calculation ME-02-14-02 has determined that with a normal fuel inventory in the SFP and without recovery actions that the SFP water level would be 2 feet above

the top of the fuel after 7 days. With a full-core offload, boil-off occurs at a rate of 97.1 gal/min.

10.0 SEQUENCE OF EVENTS

Columbia's FLEX strategy encompasses the three key safety functions: Reactor Core Cooling, Containment Integrity, and Spent Fuel Pool Cooling. A timeline for actions taken during power operations that starts with Phase 1 and ends with the implementation of Phase 3 is provided in Attachment B, Table B-1: Integrated FLEX Strategy Timeline.

Attachment B, Table B-2, *Full Core Off-Load FLEX Strategy Timeline*, identifies the actions required during a full core off-load.

11.0 PROGRAMMATIC ELEMENTS

Columbia's overall program document, FLEX-01, *FLEX Program*, contains the programmatic controls for the diverse and flexible coping strategies program for Columbia required by Section 11 of NEI 12-06 (Reference 3). This section summarizes the programmatic controls that were considered in the implementation of the plant-specific FLEX strategies developed for Columbia. A short description of the key elements of the program as defined in NEI 12-06, are presented below. Each description provides a reference to either a section in this plan or to FLEX-01, the FLEX program document.

11.1 Quality Attributes

Quality attributes are the non-functional requirements of a structure, system, or component. NEI 12-06 (Reference 3) states that FLEX equipment associated with these strategies will be procured as commercial equipment with design, storage, maintenance, testing, and configuration control as outlined in this section. If the equipment is credited for other functions (e.g., fire protection), then the quality attributes of the other functions apply. See Section 6.2 of FLEX-01.

11.2 Equipment Design

Design requirements and supporting analysis have been developed for the FLEX equipment that directly performs a FLEX mitigation strategy for core, containment, and SFP that provides the inputs, assumptions, and documented analysis that the mitigation strategy and support equipment will perform as intended. See Section 6.3 of FLEX-01.

11.3 Equipment Storage

FLEX equipment is stored in locations that provide reasonable protection during specific external events. When FLEX equipment is moved from its designated

storage or stored outside, the location is evaluated to assure that the equipment is protected so that no one external event can reasonably fail the site's FLEX capability (N). When selecting the storage locations, consideration was given to the transport routes to be used following the event, recognizing that external vents can result in obstacles restricting normal pathways for movement. Section 5.0, *Protection of FLEX Equipment*, provides discussion on the FLEX buildings, deployment paths, and refueling of FLEX equipment.

11.4 Procedure Guidance

Section 7.0 of FLEX-01 describes the procedural approach for the implementation of the FLEX strategies. This approach includes appropriate interfaces between the various accident mitigation procedures so that overall strategies are coherent and comprehensive.

The FSGs provide pre-planned strategies for accomplishing specific tasks. The FSGs support the EOPs, Extensive Damage Mitigation Guidelines (EDMG), and Severe Accident Management Guidelines (SAMG) strategies. They provide clear criteria for entry to ensure that the FLEX strategies are used only as directed, and are not used inappropriately in lieu of existing procedures. The FSGs are controlled under the site procedure control program SWP-PRO-02.

11.5 Maintenance and Testing

FLEX equipment was either initially tested or other reasonable means were used to verify that the performance conforms to the limiting FLEX requirements. Validation of the source manufacturer quality was not performed.

FLEX equipment that directly performs a FLEX mitigation strategy for the core, containment, or SFP was developed using the maintenance and testing guidance provided in INPO AP 913, *Equipment Reliability Process*, and is detailed in Appendix E of FLEX-01.

The maintenance program ensures that FLEX equipment reliability is being achieved. Specifically, the following was considered when developing the maintenance and testing program:

1. Periodic testing and frequency was determined based on equipment type and expected use,
2. Preventive maintenance was determined based on equipment type and expected use, and
3. The existing work control processes will be used to control the maintenance and testing of the FLEX equipment.

The maintenance and testing for plant equipment used in the FLEX strategies is conducted in accordance with existing plant procedures and processes.

The unavailability of FLEX equipment and applicable connections that directly performs a FLEX mitigation strategy for core, containment, and SFP is managed such that risk to mitigating strategy capability is minimized in accordance FLEX-01 Section 6.6, "Unavailability of Equipment and Connections," and Station procedure 1.5.18, *Managing B.5.B and FLEX Equipment Unavailability*.

B.5.b Pumper Flow Verification

The B5b pumper truck had service performed on its pump in November 2014 by Hughes Fire Equipment in Tacoma, WA. At the completion of this service, a flow test was performed by Hughes to verify performance of the pump. This test data was produced using draft (unpressurized) suction and shows that the pump is performing according to its design parameters.

A separate test just to verify draft capability with the FLEX suction strainer configuration was performed on March 17, 2014. It showed that the pump successfully primes from a pond source of water with the floating strainer. Flow rates of greater than 600 gpm through a 4 inch fire hose were generated in this test.

B5b Pumper Truck			
PM Task Name	**EPRI Baseline**	**Columbia**	**PMID**
Standby Walkdown	1 M	1 M	23638-01
Component Operational Inspection	1 Y	1 Y	PPI 15.1.29, AR 317715
Functional Test and Inspection	3 M	1 M	23638-01
In-Service Walkdown	NA	NA	NA
Fluid Analysis	1 Y	1 Y	23579-02
Return to Standby	AR	AR	23638-01
Component Operational Inspection	1 Y	1 Y	AR 317715, 23638-01
Fluid Filter Replacement	2 Y	3 M	23579-02
Performance Test	3 Y	1 Y PPI 15.1.29	AR 317715

GODWIN Pump Testing

The pump was subjected to a performance test performed by the manufacturer on June 1, 2012, showing that the pump met the published performance characteristics. This pump underwent an on-site flow and pressure verification test in April of 2015. The acceptance criteria is that at 2100 rpm the pump head

must not be less than 85% of the rated capability reflected in the vendor pump curve at a flow rate between 550 and 600 gpm.

While these values are slightly lower than the pump manufacturer design curves, they are within the 15% limit specified by calculation ME-02-12-06, Appendix H to satisfy the required flow characteristics to perform its FLEX function. The referenced calculation states that the flow and head required to meet Spent Fuel Pool and RPV flows are 465 gpm at a head of 523 feet. As shown on the pump curve plots, FLEX-P-1 exceeds this in both flow and pressure output.

FLEX-P-1			
PM Task Name	**EPRI Baseline**	**Columbia**	**PMID**
Standby Walkdown	1 M	1 M	PPI 3.1.10 (no PMID)
Component Operational Inspection and Performance Test	AR	AR	26233-02
Functional Test and Inspection	3 M	6 M	26233-03
In-Service Walkdown	NA	NA	NA
Fluid Analysis	1 Y	1 Y	26233-07
Return to Standby	AR	AR	26233-01, 02,03
Component Operational Inspection	1 Y	1 Y	26233-02
Fluid Filter Replacement	2 Y	2 Y	26233-06
Performance Test	3 Y	4 Y	26233-01

Diesel Generator 5 Testing

Sigma Control performed load bank testing of DG5 between August and September of 2014 with acceptable results.

DG5			
PM Task Name	**EPRI Baseline**	**Columbia**	**PMID**
Standby Walkdown	1 M	1 W	PPI 3.1.10 (no PMID)
Component Operational Inspection	1 Y	1 Y	26234-01
Functional Test and Inspection	3 M	3 M	26234-01,02
In-Service Walkdown	NA	NA	NA
Fluid Analysis	1 Y	1 Y	26234-03
Return to Standby	AR	AR	26234-01,02
Fluid Filter Replacement	2 Y	2 Y	26234-05
Generator Full Load Test	3 Y	1 Y	26234-02
Generator Offline Testing	3 Y	3 Y	26234-04

Diesel Generator 4 Testing

This FLEX generator is the existing DG4 which is normally in standby condition stored outdoors. The preventive maintenance program for this generator was already in existence to support its function in establishing the alternate AC sources to division 1 or division 2 to support TS 3.8.1 *AC Sources – Operating* and was compared to the EPRI template frequencies for FLEX generators. This comparison shows that the existing PM program frequencies are equal to or more frequent than required by the EPRI PM templates for FLEX generators.

DG4			
PM Task Name	**EPRI Baseline**	**Columbia**	**PMID**
Standby Walkdown	1 M	1 W	PPI 3.1.10
Component Operational Inspection	1 Y	1 M	22773-08
Functional Test and Inspection	3 M	3 M	22773-06
In-Service Walkdown	NA	NA	NA
Fluid Analysis	1 Y	1 Y	22773-01
Return to Standby	AR	AR	22773-06, 08
Fluid Filter Replacement	2 Y	1 Y	22773-01
Generator Full Load Test	3 Y	1 M	22773-08
Generator Offline Testing	3Y	3 Y	22773-07
Replace Engine Coolant	NA	2 Y	22773-04

11.6 Training

FLEX-01, Section 8.0, *Personnel*, discusses the training provided to key personnel relied upon to implement the procedures and guidelines for responding to a beyond design basis event using the "Systems Approach to Training" (SAT) elements listed in 10 CFR 55.4.

11 7 Staffing

Staffing is discussed in Section 8.5 of this plan.

11.8 Configuration Control

The FLEX strategies and bases are maintained in the overall program document FLEX-01, FLEX Program.

The program document contains a historical record of any previous strategies and the basis for changes. The program document also contains the basis for the ongoing maintenance and testing programs in Appendix E.

11.9 Additional Program Elements

The overall program document, FLEX-01, also contains the following program elements:

- Section 2.0, FLEX Program Roles and Responsibilities
- Section 3.0, Associated Regulatory Requirements
- Appendix D, Time Critical Validation Documentation
- Appendix G, Major On-Site FLEX Equipment
- Appendix H, Listing of Technical Documents
- Appendix I, Plant Systems Requiring Interface to Implement FLEX Strategies
- Appendix M, Columbia's N+1 Strategy

Attachments:

Attachment A – Figures

Attachment B - FLEX Strategy Timelines

Attachment C - References

Attachment A:
Figures

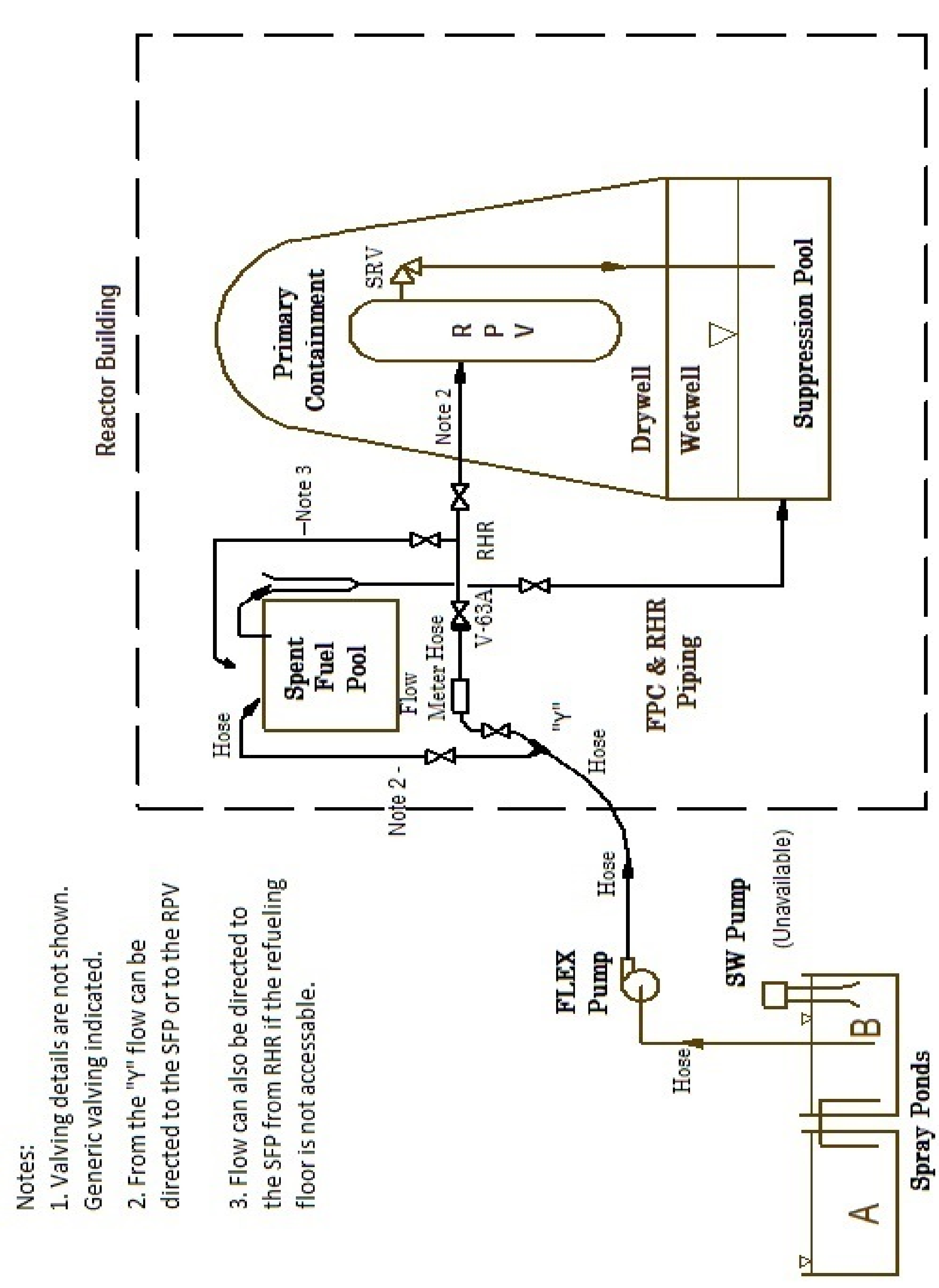


Figure 1: Phase 2 Water Makeup Flow Diagram

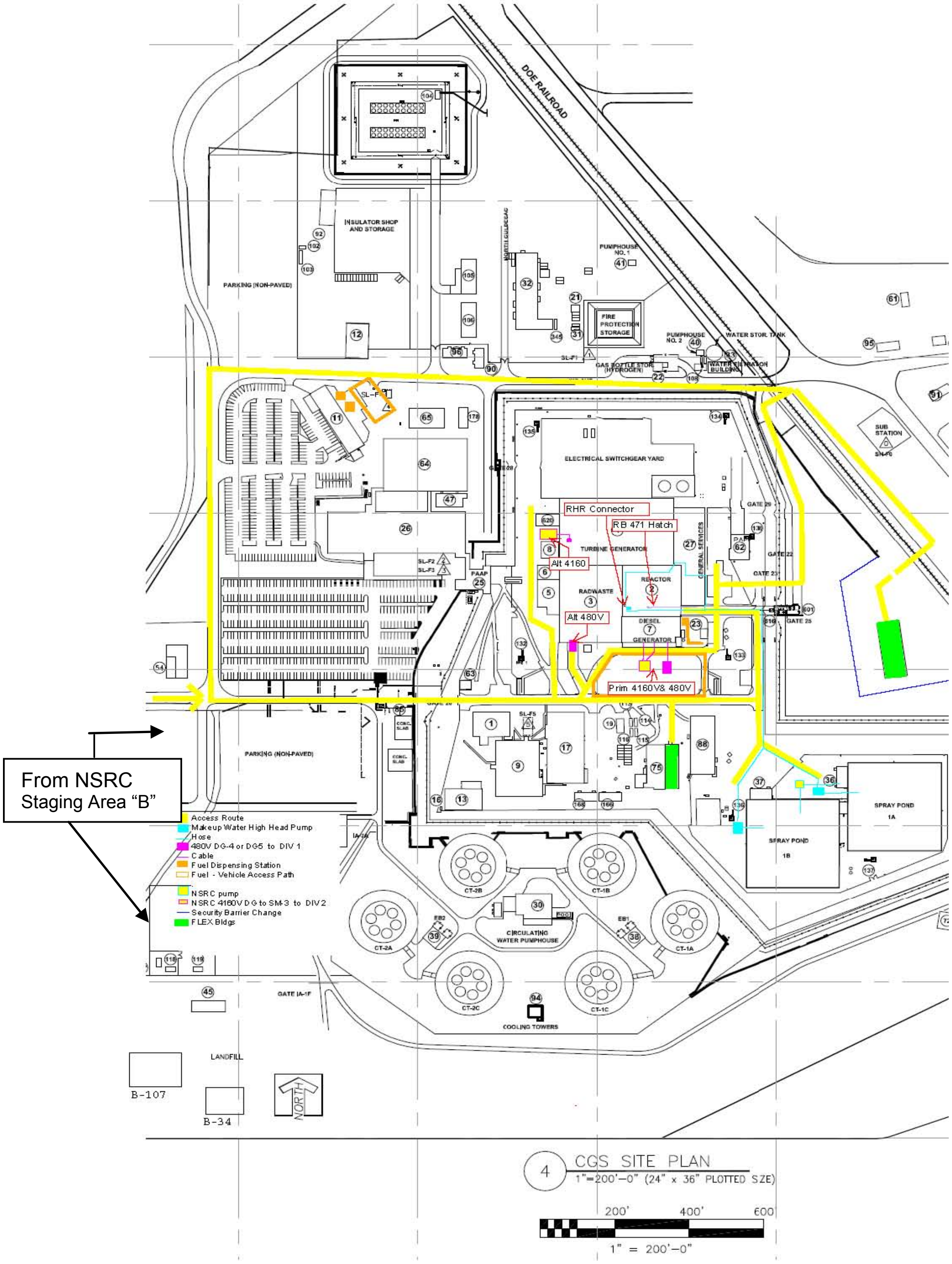


Figure 2: FLEX Deployment Routs

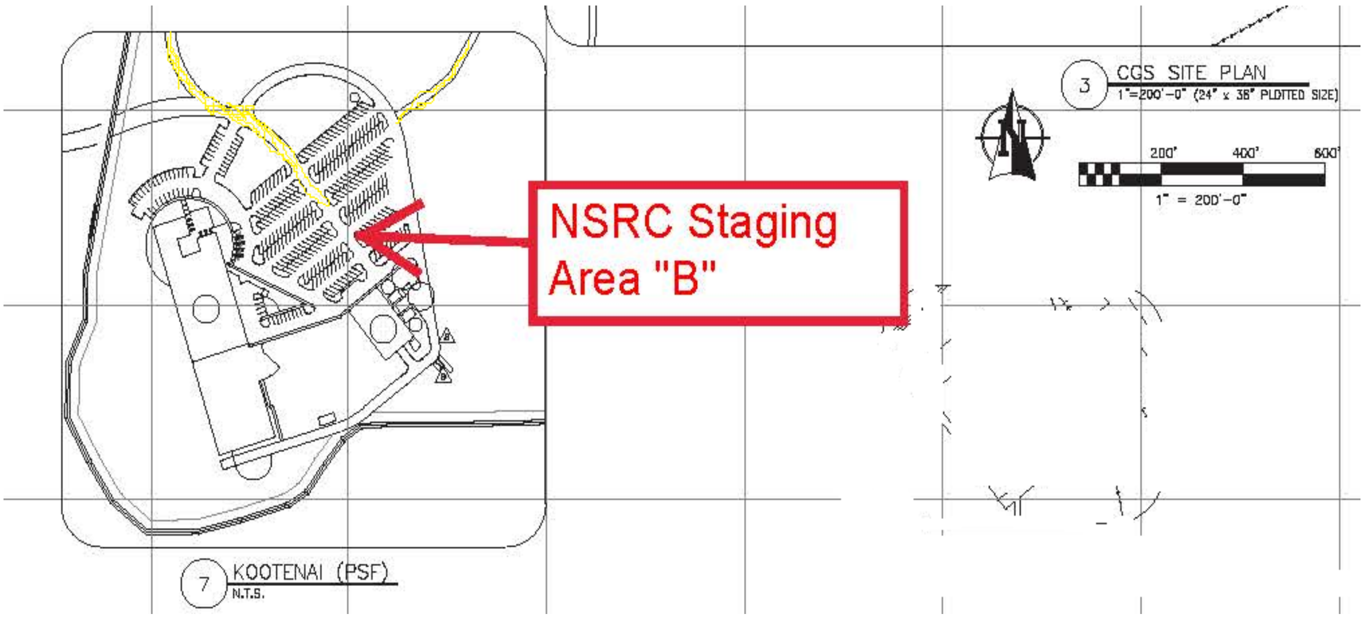


Figure 2 A: FLEX Deployment Rout from NSRC Staging Area B
(Staging Area "A" is the final location shown on Figure 2)

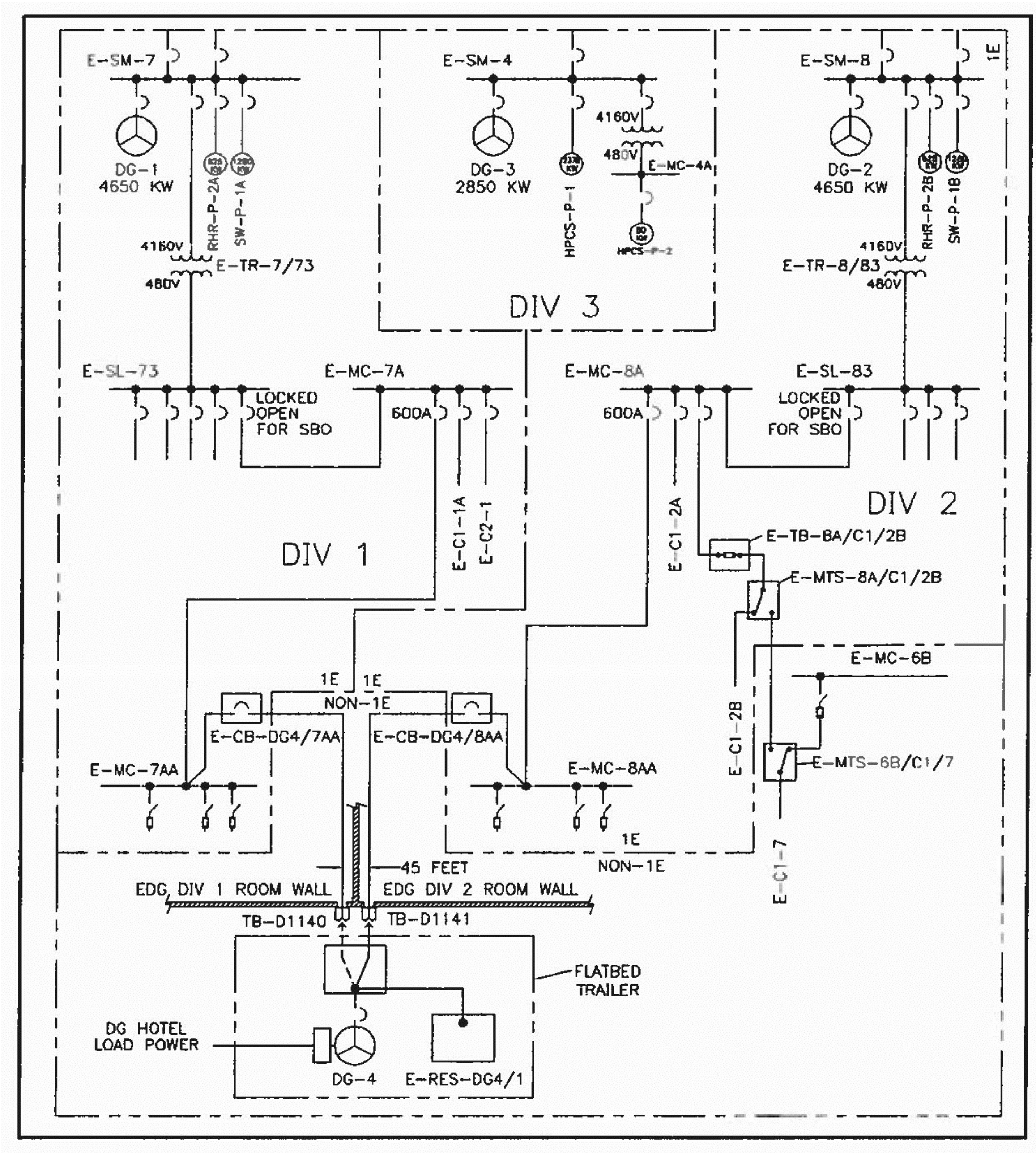


Figure 3: FLEX DG Connections Outside of DG Building

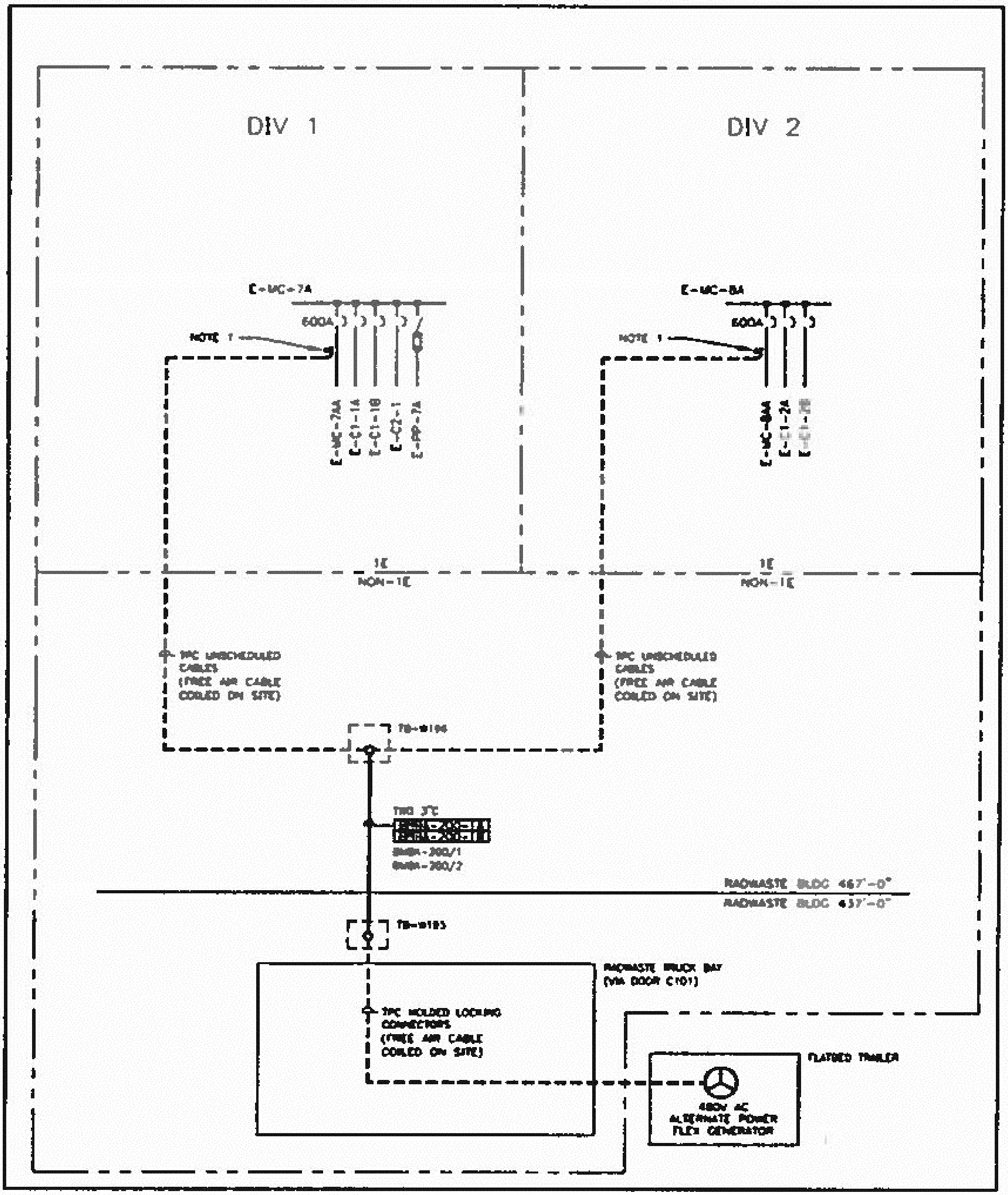


Figure 4: FLEX DG Connections Inside of Radwaste Building

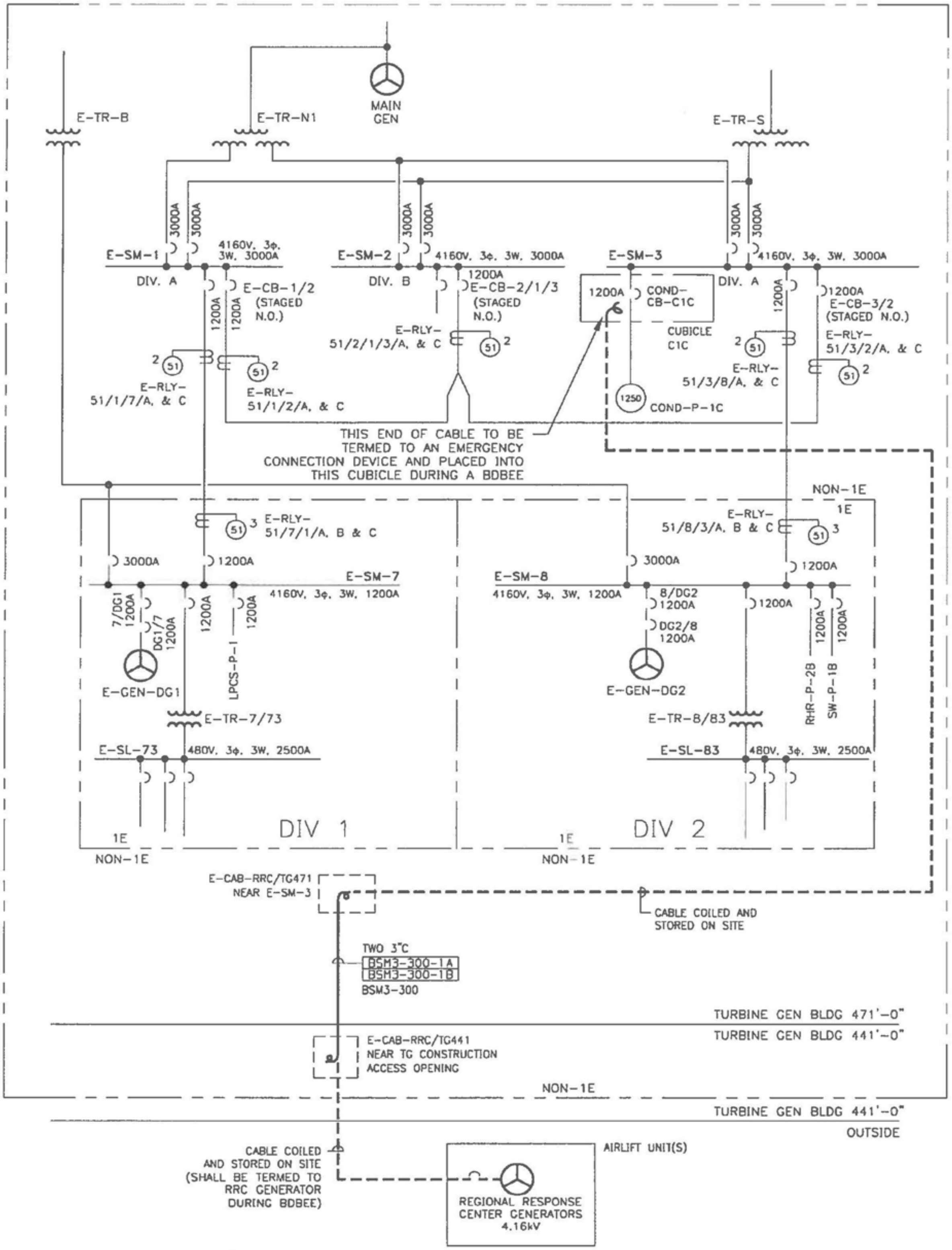


Figure 5: 4.16-kV NSRC Generator Connection 1

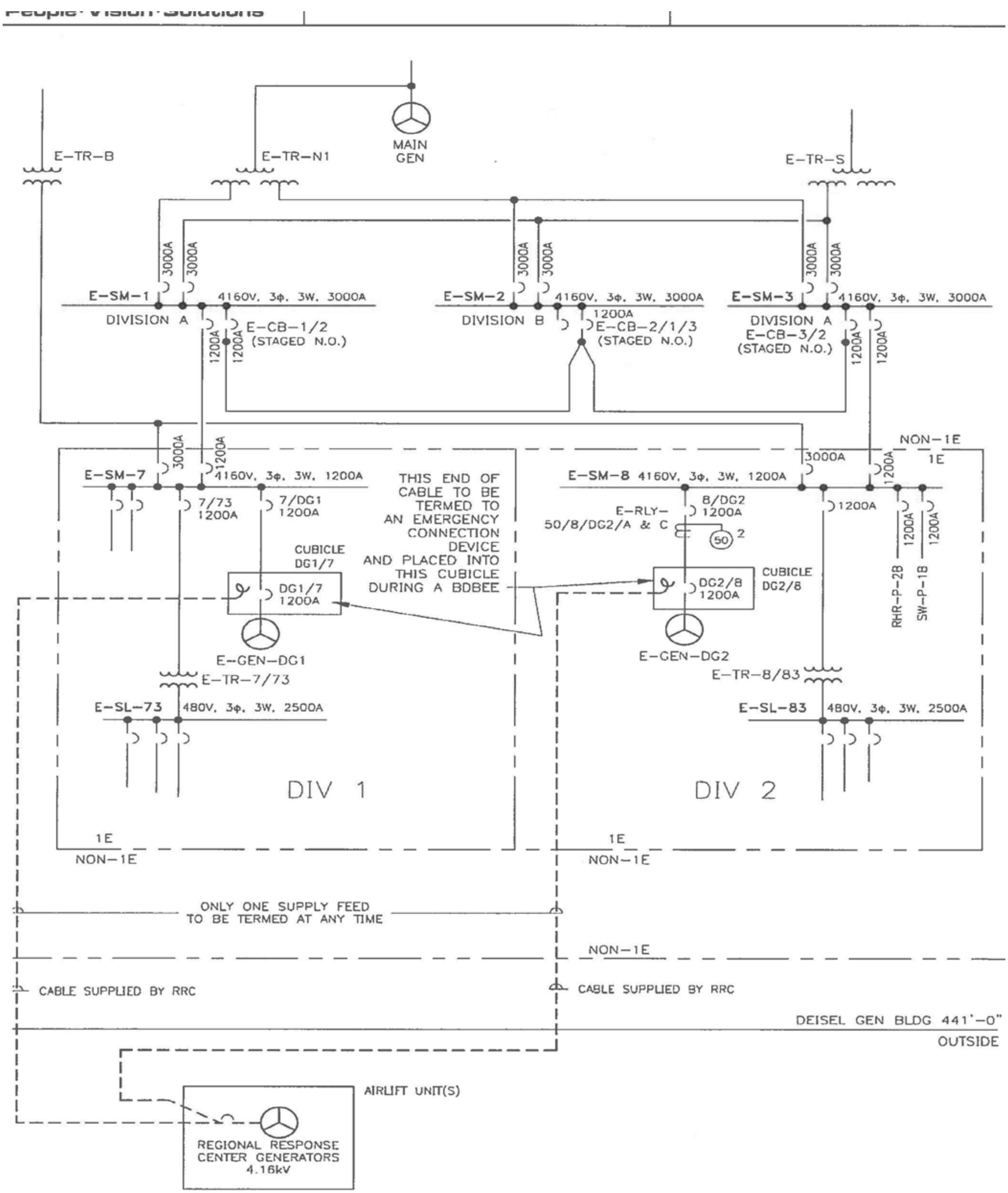


Figure 6: 4.16-kV NSRC Generator Connection 2

Attachment B
FLEX Strategy Timelines

Table B-1: Integrated FLEX Strategy Timeline			
	Action	**Time Constraint (t=0 is loss of power)**	**Discussion**
	Event Starts	0	Plant at 100 percent power at time = 0. All AC power is lost including that from the installed emergency DGs.
1	Reactor Core Isolation Cooling (RCIC) starts	Within 1 min.	Existing Station Blackout (SBO) coping strategy.
	Suction swaps from CST to suppression pool		Automatic
2	Operations crew enters SBO/ELAP procedure PPM 5.6.1	Within 15 min.	The timing is consistent with requirements for classifying an emergency.
3	Monitor RPV and containment parameters and initiate RPV cool down	Continuous	Consider controlling pressure between 100 and 300 psig using PPM 5.6.1 and 5.6.2 RPV cool down is initiated at an appropriate point in the procedure. Temperature change is limited to less than or equal to 100°F per hour. Pressure is maintained as required to ensure continued RCIC operation.
4	Complete compensatory measures to promote CR cooling	Within 30 min.	PPM 5.6.2 Att. 8.5 TM-2187, *Actions, Limitations, and Notes Associated with an Extended Loss of AC Power* (TM-2187 Table 2)
5	Consult with regional load centers on offsite power recovery	Within 45 min.	Priority restoration of power to Columbia is provided for in agreements with the Bonneville Power Administration (BPA). If an AC power source will be recovered within the functional life of the batteries, load reductions must be completed within the first hour.
6	Determine if AC power will not be restored within the normal SBO coping period (4 hours) (i.e., declare ELAP)		If AC power will not be restored within 4 hours, declare an ELAP condition exists and initiate additional compensatory measures to promote cooling in required areas of the Reactor Building, control room and vital island. PPM 5.6.2 Att. 8.11 Secondary Containment Supplementary Cooling PPM 5.6.2 Att.8.12 Vital Island Temperature Control Actions
7	Complete 125 volt dc load shed	Within 60 min.	PPM 5.6.2 Att. 8.4 See TM-2187
8	Perform 250 volt dc load shed	Within 2 hrs.	In order to extend the ability of the 250 volt batteries to meet the extended demand during an ELAP, additional loads are shed.

Table B-1: Integrated FLEX Strategy Timeline			
	Action	**Time Constraint (t=0 is loss of power)**	**Discussion**
9	Vent the main generator		PPM 5.6.2 Att. 8.9 Includes stopping SO-P-ASBU the air side seal oil backup pump
101	If not already initiated, depressurize RPV at a rate not to exceed 100°F per hour and maintain the RPV pressure required for RCIC operation.	Within 3 hrs.	Under ELAP conditions the RPV should be depressurized to facilitate long term RCIC operation. Generally, this action will begin no later than 30 minutes. The primary factor the shift manager will use for determining if this is a time constraint will be the time to reach the Heat Capacity Temperature Limit (HCTL). Depressurizing the RPV will increase the margin to HCTL. After depressurization to 175 psig cycle RPV pressure between 175 and 300 psig. See TM-2187
11	Remove the pins on piping support RCIC-967N	Prior to reaching 170°F at approximately 3 hrs.	Pins on RCIC-967N must be removed to reduce pipe stress under high water temperature conditions. PPM 5.6.2 Att. 8.2 See TM-2187
12	Vent containment using the hardened containment vent system	Within 6 hrs.	Modular Accident Analysis Program (MAAP) analyses have been performed that determine the relationship between the timing of initiation of containment (wetwell) venting and the maximum suppression pool temperature. Maximum suppression pool temperature can affect the long term availability of RCIC. It is assumed that the normal RCIC suction source from condensate storage tank (CST) is unavailable, and RCIC is taking suction from the suppression pool. Analysis indicates that maximum suppression pool temperature is acceptable if venting is initiated within 6 hours. ME-02-12-18 See TM-2187
13	Open additional breakers for Control Room cooling		Designated breakers are given in TM-2187 Section Q

Table B-1: Integrated FLEX Strategy Timeline

	Action	Time Constraint (t=0 is loss of power)	Discussion
14	Begin deployment of FLEX fueling equipment	Beginning after 6 hrs.	After supplemental staff arrives, begin staging of refueling equipment to support continued operation. See TM-2187
15	Open designated doors in Vital Island, Radwaste Bldg.	Within 8 hrs.	PPM 5.6.2 Att. 8.12
16	Connect FLEX equipment for battery charging		Battery calculations indicate that the batteries will provide power for at least 8 hours. Thus, battery charging must be established at or before 8 hours. This action will generally be started as soon as possible with the available on-shift staff.
17	Refuel credited FLEX equipment within required time.	Within 10 hrs. and periodically thereafter	All credited equipment can operate for at least 10 hrs. before requiring refueling SOP-FLEX-EQUIPMENT-REFUEL See TM-2187
18	Bypass RCIC trips	Within 10 hrs.	Selected RCIC trips are to be bypassed per the SBO/ELAP procedure to ensure continued operation of RCIC. (FSAR 8A.2.2) Generally this action is completed within 30 minutes, but may be allowed to take as long as 10 hours. RCIC trips will be bypassed consistent with the Boiling Water Reactors Owners Group (BWROG) recommendations. PPM 5.6.2 Att. 8.2
19	Install 6000 cfm fan in doorway to room C221, Battery Charging Rm 1	Within 12 hrs.	This action facilitates cooling of battery charging room. PPM 5.6.2 Att. 8.12 See TM-2187

Table B-1: Integrated FLEX Strategy Timeline			
	Action	Time Constraint (t=0 is loss of power)	Discussion
20	Perform actions to promote RCIC room and general Reactor Building cooling		Actions to open identified doors to the RCIC pump room, building stairwells, refueling floor ceiling hatch, 471 ft. floor hatch, doors in other areas in the Reactor Building and remove one RCIC room ceiling plug; all necessary to provide added ventilation for the RCIC pump room and the Reactor Building in general. Install temporary flood barriers at RCIC plug and stairwell S-3 openings (if needed to mitigate internal flooding PPM 5.6.2 Att. 8.11 See TM-2187
21	Connect FLEX equipment to refill RPV.		In order to protect the fuel in case of RCIC failure the makeup potential of the FLEX high-head pumps will be established as early as possible, considering staffing and site conditions. ABN-FSG-002
22	Connect FLEX equipment to provide cooling water to SFP.		GOTHIC analyses indicate that SFP makeup is required within 12 hours to preserve accessibility in the Reactor Building higher elevations. Establish a 300 gpm make-up to the SFP to support habitability in the Reactor Building. The SFP makeup will be cascaded to the Suppression Pool for makeup and cooling. In three days of operation, the cascaded makeup will not flood the Wetwell vent. See Discussion Item 4. ABN-FSG-002 See TM-2187

Table B-2: Full Core Off-Load FLEX Strategy Timeline			
	Action	Time Constraint (t=0 is loss of power)	Discussion
	Stage FLEX equipment required to provide cooling water to the Spent Fuel Pool	Before off-load	The response time is very limited. These actions are identified in TM-2187
1	Event Starts	0	Plant is at 0 percent power at time=0. Reactor vessel is void of fuel and all fuel has been moved to the SFP. All AC power is lost including that from the installed emergency DGs.
2	Complete activities on refueling floor, evacuate refueling floor area.	Within 2 hrs.	The response time is very limited. Equipment is pre-staged to enable 2 hour response on providing makeup water. SeeTM-2187

Attachment C:
FIP References

FIP References			
Ref. No.	**Document No.**	**Title**	**Document Date**
1	SECY-11-0093	Recommendations for Enhancing Reactor Safety in the 21st Century; The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident [Package ML11186A950]	07/12/2011
2	EA-12-049	Order to Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events	3/12/2012
3	NEI 12-06	Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, Revision 2	12/2015
4	JLD-ISG-2012-01	Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events	08/29/2012
5	NEI 12-01	Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities, Revision 0	05/20/2012
6	ML13358A206	NRC letter from J. R. Davis to J. E. Pollock (NEI) "Containment Venting Strategies"	01/09/2014
7	ML13241A188	NRC letter from J. R. Davis to J. E. Pollock (NEI) "Battery Life"	09/16/2013
8	GO2-14-026	Request for Relaxation from NRC Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events"	02/21/2014
9	ML14071A572	Relaxation of Certain Schedule Requirements for Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events	04/15/2014
10	GO2-14-107	Energy Northwest's Phase 1 Response to NRC Order EA-13-109 – Overall Integrated Plan for Reliable Hardened Containment Vents Under Severe Accident Conditions	06/30/2014
11	GO2-15-175	Energy Northwest's Response to NRC Order EA-13-109 – Overall Integrated Plan for Reliable Hardened Containment Vents under Severe Accident Conditions Phases 1 and 2, Revision 1	12/16/2015
12	GO2-16-143	Flooding Hazard Reevaluation Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident	10/06/2016
13	CVI 1228-00,10	SAFER Response Plan for Columbia Generating Station, Revision 1	01/07/2015
14	EP-01	Emergency Plan Columbia Generating Station, Revision 61	12/22/2014
15	NEI 10-05	Assessment of On-Shift Emergency Response Organization Staffing and Capabilities	06/2011

FIP References			
Ref. No.	**Document No.**	**Title**	**Document Date**
16	GO2-12-069	Energy Northwest's 60-Day Response to the March 12, 2012 Information Request Related to Recommendation 9.3	05/10/2012
17	GO2-14-174	Energy Northwest's NEI 12-01 Phase 2 Staffing Assessment	12/23/2014
18	ML13273A514	NEI Position Paper "Shutdown/Refueling Modes"	09/18/2013
19	ML13267A382	NRC letter from J. R. Davis (NRC) to J. E. Pollock (NEI) "NEI Position Paper on Shutdown/Refueling Modes"	09/30/2013
20	CVI 1217-00,1,1 R0	MOHR Test and Measurement LLC EFP-IL SFPI System Manual	
21	GO2-16-021	Expedited Seismic Evaluation Process (ESEP) Report in Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident	01/20/2016
22	GO2-16-070	Response to Request for Additional Information Associated with Expedited Seismic Evaluation Process Submittal	05/17/2016
23	ML15113B344	Screening and Prioritization Results for the Western United States Sites Regarding Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Seismic Hazard re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident	05/13/2015
24	GO2-15-045	Seismic Hazard and Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident	03/12/2015
25	ML15215A043	Columbia Generating Station - Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Reevaluations	08/18/2015
26	ML15254A257	Columbia Generating Station - Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Reevaluations	09/16/2015
27	GO2-15-137	Response to the Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Reevaluations	09/24/2015

FIP References			
Ref. No.	**Document No.**	**Title**	**Document Date**
28	GO2-15-143	Response to the Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Reevaluations	10/14/2015
29	ML16285A410	Columbia Generating Station Staff Assessment of Information Provided Under Title 10 of the *Code Of Federal Regulations* Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dal-Ichi Accident (CAC No. MF5274)	11/4/2016
30	ML12053A340	Request for Information Pursuant to Title of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3 of the Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident	03/12/2012
31	ML16202A414	Transmittal of U.S. Army Corps of Engineers Flood Hazard Reevaluation Information	08/11/2016
32	PKG ML16337A111 LTR ML16337A109	Columbia Generating Station - Interim Staff Response to Reevaluated Flood Hazards Submitted in Response to 10 CFR 50.54(f) Information Request - Flood-Causing Mechanism Reevaluation (CAC No. Mf3039)	12/7/2016
32	CVI 1217-00.1.2 R0	MOHR Test and Measurement, LLC EFP-IL SFPLI System Reports/FMEA	10/19/2016
33	NE-02-13-04 R0	Cycle 22 Spent Fuel Pool Time-to-200°F	06/05/2013
34	NE-02-17-02 R0	Cycle 24 SFP Time-to200°F	06/07/2017
35	AR 278368-31	Columbia Generating Station Assessment of NRC Limitations on the Use of MAAP4 for ELAP Analysis R1	8/26/2014
36	ML13241A186	NEI White paper "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic concern"	8/27/2013