UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 22, 2018

Mr. Mark E. Reddemann
Chief Executive Officer
Energy Northwest
MD 1023
76 North Power Plant Loop
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SUBJECT: COLUMBIA GENERATING STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 (CAC NOS. MF0796, MF0797, EPIDS L-2013-JLD-0005 AND L-2013-JLD-0006)

Dear Mr. Reddemann:

On March 12, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events" and Order EA-12-051, "Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation," (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12054A736 and ML12054A679, respectively). The orders require holders of operating reactor licenses and construction permits issued under Title 10 of the *Code of Federal Regulations* Part 50 to modify the plants to provide additional capabilities and defense in depth for responding to beyond-design-basis external events, and to submit for review Overall Integrated Plans (OIPs) that describe how compliance with the requirements of Attachment 2 of each order will be achieved.

By letter dated February 28, 2013 (ADAMS Accession No. ML13071A614), Energy Northwest (the licensee) submitted its OIP for Columbia Generating Station (Columbia) in response to Order EA-12-049. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated January 29, 2014 (ADAMS Accession No. ML13337A266), and June 16, 2015 (ADAMS Accession No. ML15139A462), the NRC issued an Interim Staff Evaluation (ISE) and an audit report, respectively, on the licensee's progress. By letter dated August 17, 2017 (ADAMS Accession No. ML17229B506), Energy Northwest submitted a compliance letter and Final Integrated Plan (FIP) in response to Order EA-12-049. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-049.

By letter dated February 28, 2013 (ADAMS Accession No. ML13071A470), Energy Northwest submitted its OIP for Columbia in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached safety evaluation. By letters dated November 7, 2013 (ADAMS Accession No. ML13302C136), and June 16, 2015 (ADAMS Accession No. ML15139A462), the NRC staff issued an ISE and an audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated August 12, 2015 (ADAMS Accession No. ML15245A530), Energy Northwest submitted a compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051.

The enclosed safety evaluation provides the results of the NRC staff's review of Energy Northwest's strategies for Columbia. The intent of the safety evaluation is to inform Energy Northwest on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 2515-191, "Inspection of the Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communication/Staffing/Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML15257A188). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact John Boska, BDB Management Branch, Columbia Project Manager, at 301-415-2901 or at John.Boska@nrc.gov.

Sincerely,

Eric E. Bowman, Acting Chief
Beyond-Design-Basis Management Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No.: 50-397

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ORDERS EA-12-049 AND EA-12-051

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

## 1.0 INTRODUCTION

The earthquake and tsunami at the Fukushima Dai-ichi nuclear power plant in March 2011 highlighted the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers already in place in nuclear power plants in the United States. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. During the events in Fukushima, the challenges faced by the operators were beyond any faced previously at a commercial nuclear reactor and beyond the design basis of the plants. The U.S. Nuclear Regulatory Commission (NRC) determined that additional requirements needed to be imposed at U.S. commercial power reactors to mitigate such beyond-design-basis external events (BDBEEs).

On March 12, 2012, the NRC issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" [Reference 4]. This order directed licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities in the event of a BDBEE. Order EA-12-049 applied to all power reactor licensees and all holders of construction permits for power reactors.

On March 12, 2012, the NRC also issued Order EA-12-051, "Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation" [Reference 5]. This order directed licensees to install reliable SFP level instrumentation with a primary channel and a backup channel, and with independent power supplies that are independent of the plant alternating current (ac) and direct current (dc) power distribution systems. Order EA-12-051 applies to all holders of operating licenses and construction permits for power reactors.

## 2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC regulations and processes and determining if the agency should make additional improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the NTTF developed a comprehensive set of recommendations, documented in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 [Reference 1]. Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

On February 17, 2012, the NRC staff provided SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," [Reference 2] to the Commission. This paper included a proposal to order licensees to implement enhanced BDBEE mitigation strategies. As directed by the Commission in staff requirements memorandum (SRM)-SECY-12-0025 [Reference 3], the NRC staff issued Orders EA-12-049 and EA-12-051.

### 2.1 Order EA-12-049

Order EA-12-049, Attachment 2, [Reference 4] requires that operating power reactor licensees and construction permit holders use a three-phase approach for mitigating BDBEEs. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and SFP cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely. Specific requirements of the order are listed below:

1) Licensees or construction permit (CP) holders shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event.

2) These strategies must be capable of mitigating a simultaneous loss of all alternating current (ac) power and loss of normal access to the ultimate heat sink [UHS] and have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

3) Licensees or CP holders 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, and SFP cooling capabilities at all units on a site subject to this Order.

4) Licensees or CP holders must be capable of implementing the strategies in all modes of operation.

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

On December 10, 2015, following submittals and discussions in public meetings with NRC staff, the Nuclear Energy Institute (NEI) submitted document NEI 12-06, Revision 2, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," [Reference 6] to the NRC to provide revised specifications for an industry-developed methodology for the development, implementation, and maintenance of guidance and strategies in response to Order EA-12-049. The NRC staff reviewed NEI 12-06, Revision 2, and on January 22, 2016, issued Japan Lessons-Learned Division (JLD)[1] Interim Staff Guidance (ISG) JLD-ISG-2012-01, Revision 1, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," [Reference 7], endorsing NEI 12-06, Revision 2, with exceptions, additions, and clarifications, as an acceptable means of meeting the requirements of Order EA-12-049, and published a notice of its availability in the *Federal Register* (81 FR 10283).

### 2.2 Order EA-12-051

Order EA-12-051, Attachment 2, [Reference 5] requires that operating power reactor licensees and construction permit holders install reliable SFP level instrumentation. Specific requirements of the order are listed below:

> All licensees identified in Attachment 1 to the order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.
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> 1. The spent fuel pool level instrumentation shall include the following design features:
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> 1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.
>
> 1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding

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[1] Revision 0 was issued by the Japan Lessons-Learned Project Directorate, which also used the term "JLD." No distinction is made in this safety evaluation between documents issued by the two organizations because the Japan Lessons-Learned Division continued the numbering scheme used by the Japan Lessons-Learned Project Directorate.

from missiles provided by existing recesses and corners in the spent fuel pool structure.

1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

1.4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.

1.6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

1.7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

1.8 Testing: The instrument channel design shall provide for routine testing and calibration.

1.9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.

2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

On August 24, 2012, following several NEI submittals and discussions in public meetings with NRC staff, the NEI submitted document NEI 12-02, "Industry Guidance for Compliance With NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1 [Reference 8] to the NRC to provide specifications for an industry-developed methodology for compliance with Order EA-12-051. On August 29, 2012, the NRC staff issued its final version of JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation" [Reference 9], endorsing NEI 12-02, Revision 1, as an acceptable means of meeting the requirements of Order EA-12-051 with certain clarifications and exceptions, and published a notice of its availability in the *Federal Register* (77 FR 55232).

## 3.0 TECHNICAL EVALUATION OF ORDER EA-12-049

By letter dated February 28, 2013 [Reference 10], Energy Northwest (the licensee) submitted an Overall Integrated Plan (OIP) for Columbia Generating Station (Columbia, CGS) in response to Order EA-12-049. By letters dated August 28, 2013 [Reference 11], February 27, 2014 [Reference 12], August 28, 2014 [Reference 13], March 2, 2015 [Reference 14], August 25, 2015 [Reference 34], February 24, 2016 [Reference 35], August 30, 2016 [Reference 36], December 29, 2016 [Reference 37], and June 27, 2017 [Reference 38], the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 [Reference 15], the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 36]. By letters dated January 29, 2014 [Reference 16], and June 16, 2015 [Reference 17], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated August 17, 2017 [Reference 18] the licensee reported that full compliance with the requirements of Order EA-12-049 had been achieved, and submitted a Final Integrated Plan (FIP).

### 3.1 Overall Mitigation Strategy

Attachment 2 to Order EA-12-049 describes the three-phase approach required for mitigating BDBEEs in order to maintain or restore core cooling, containment, and SFP cooling capabilities. The phases consist of an initial phase (Phase 1) using installed equipment and resources, followed by a transition phase (Phase 2) in which portable onsite equipment is placed in service, and a final phase (Phase 3) in which offsite resources may be placed in service. The timing of when to transition to the next phase is determined by plant-specific analyses.

While the initiating event is undefined, it is assumed to result in an extended loss of ac power (ELAP) with a loss of normal access to the UHS (LUHS). Thus, the ELAP with LUHS is used as a surrogate for a BDBEE. The initial conditions and assumptions for the analyses are stated in NEI 12-06, Section 3.2.1, and include the following:

1. The reactor is assumed to have safely shut down with all rods inserted (subcritical).
2. The dc power supplied by the plant batteries is initially available, as is the ac power from inverters supplied by those batteries; however, over time the batteries may be depleted.
3. There is no core damage initially.
4. There is no assumption of any concurrent event.
5. Because the loss of ac power presupposes random failures of safety-related equipment (emergency power sources), there is no requirement to consider further random failures.

Columbia is a General Electric boiling-water reactor (BWR) Model 5 with a Mark II type containment. The containment consists of primary containment and secondary containment. The primary containment is a free-standing steel pressure vessel divided into a drywell and a pressure suppression chamber known as the wetwell. The wetwell contains a large volume of water known as the suppression pool. The secondary containment is the reactor building, which completely encloses the primary containment. The licensee's three-phase approach to mitigate a postulated ELAP event, as described in the FIP, is summarized below.

At the onset of an ELAP the reactor is assumed to trip from full power. The main condenser is unavailable as a heat sink due to the loss of flow from the circulating water system. Decay heat is removed when the safety relief valves (SRVs) open on high pressure and dump steam from the reactor pressure vessel (RPV) to the suppression pool located in the wetwell. Makeup to the RPV is provided by the reactor core isolation cooling (RCIC) turbine-driven pump, which normally has its suction aligned to the condensate storage tanks (CSTs). Because the CSTs are not designed to survive all applicable hazards, the licensee's mitigating strategy assumes that the RCIC pump suction realigns to the suppression pool. If ac power cannot be restored, the operators take manual control of the SRVs to perform a controlled cooldown and depressurization of the reactor, at a rate not to exceed 100 degrees Fahrenheit (°F) per hour. The cooldown of the primary system is stopped when reactor pressure reaches a control band of 175 pounds per square inch gauge (psig) to 300 psig to ensure sufficient steam pressure to operate the RCIC pump. When the suppression pool heats up to a predetermined setpoint, operators open the wetwell vent to atmosphere to mitigate the temperature rise. The RPV makeup will continue to be provided from the RCIC system until the gradual reduction in RPV pressure resulting from diminishing decay heat requires a transition to Phase 2 methods. The RCIC injection source will be maintained for as long as possible, since it is a closed loop system using relatively clean suppression pool water. As suppression pool level decreases during venting, makeup water to the suppression pool will be supplied by aligning a portable diesel-driven FLEX pump to take suction from a service water spray pond and pump about 300 gallons per minute (gpm) to the SFP. The SFP will be aligned to overflow to the suppression pool.

When the RCIC system is no longer available, the preferred RPV makeup supply in Phase 2 comes from aligning the FLEX pump to supply water to the RPV from a service water spray pond.

The reactor has a Mark II containment which is inerted with nitrogen at power. The licensee performed a containment evaluation and determined that opening the wetwell vent to atmosphere will allow containment temperature and pressure to stay within acceptable levels until equipment from a National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) can be set up to provide cooling of the suppression pool.

Columbia has an SFP in its reactor building. The SFP will heat up due to the unavailability of the normal cooling system and the decay heat from the stored fuel assemblies. To maintain SFP cooling capabilities, the licensee stated that the required action is to establish the water injection lineup before the environment on the SFP operating deck degrades due to boiling in the pool so that personnel can access the refuel floor to accomplish the coping strategies.

To makeup to the SFP, the primary FLEX strategy is to connect FLEX hoses to the portable diesel-driven FLEX pump discussed above. The discharge ends of these hoses are either routed all the way to the SFP, with the discharge ends positioned over the edge of the pool, or connected to nozzles that spray the water over the spent fuel, or connected to plant piping that discharges into the SFP.

The operators will perform load stripping of the 125 volt dc (Vdc) buses within the initial 60 minutes following event initiation and load stripping of the 250 Vdc bus to ensure safety-related battery life is extended up to 8 hours. Following dc load stripping and prior to battery depletion, a 480 volt ac (Vac) FLEX generator will be used to repower essential battery chargers prior to depletion of the batteries.

In addition, an NSRC will provide high capacity pumps and large turbine-driven electrical generators which could be used to restore a residual heat removal (RHR) cooling train in the long term. There are two NSRCs in the United States.

Below are specific details on the licensee's strategies to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of a BDBEE, and the results of the staff's review of these strategies. The NRC staff evaluated the licensee's strategies against the endorsed NEI 12-06, Revision 2, guidance.

3.2 Reactor Core Cooling Strategies

Order EA-12-049 requires licensees to maintain or restore cooling to the reactor core in the event of an ELAP concurrent with a loss of normal access to the UHS. Although the ELAP results in an immediate trip of the reactor, sufficient core cooling must be provided to account for fission product decay and other sources of residual heat. Consistent with endorsed guidance from NEI 12-06, Phase 1 of the licensee's core cooling strategy credits installed equipment (other than that presumed lost to the ELAP with loss of normal access to the UHS) that is robust in accordance with the guidance in NEI 12-06. In Phase 2, robust installed equipment is supplemented by onsite FLEX equipment (e.g., pumps and hoses). The equipment available onsite for Phase 2 is further supplemented in Phase 3 by equipment transported from an NSRC.

As reviewed in this section, the licensee's core cooling analysis for the ELAP with loss of normal access to the UHS event presumes that, per endorsed guidance from NEI 12-06, the reactor would have been operating at full power prior to the event. Therefore, the suppression pool may be credited as the heat sink for core cooling during the ELAP with loss of normal access to the UHS event. Maintenance of sufficient RPV inventory, considering steam release from the SRVs and ongoing system leakage expected under ELAP conditions, is accomplished through a combination of installed systems and FLEX equipment. The specific means used by the licensee to accomplish adequate core cooling during the ELAP with loss of normal access to the UHS event are discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions where the unit is shut down or being refueled is reviewed separately in Section 3.11 of this evaluation.

3.2.1 Core Cooling Strategy and RPV Makeup

3.2.1.1 Phase 1

Per the Columbia FIP, the injection of cooling water into the RPV will be accomplished using the RCIC system. The RCIC turbine and pump are protected from all applicable hazards. Because the turbine for the RCIC pump is driven by steam from the RPV, operation of the RCIC system further assists the SRVs with RPV pressure control. The RCIC system suction is initially lined up to the CST and will pump water from the CST into the RPV. However, the CST is not protected from the seismic hazard. If the CST is unavailable the suction of the RCIC pump will be swapped to the suppression pool. The suppression pool is in the primary containment structure and is protected from all applicable hazards.

Per the FIP, pressure control of the RPV is accomplished using the SRVs. Within 3 hours after the initiation of the event, operators will utilize the SRVs to cool down and depressurize the RPV to 175-300 psig at a rate of less than 100 °F per hour. After this point, the RPV pressure is maintained between 175 and 300 psig to allow for continued operation of the RCIC system. The SRVs need both 125 Vdc power and gas pressure in order to operate. There is a backup nitrogen system in place to supply gas pressure for at least 24 hours after the initiation of the ELAP event.

Per the FIP, station batteries and the Class 1E (safety-related) 125 Vdc distribution system provide power to RCIC systems and instrumentation. Per Table B-1 in the FIP, dc load shedding is initiated by the station blackout procedures (PPM 5.6.1 and PPM 5.6.2), and is completed within 60 minutes of the loss of all ac power. This load shedding will extend the battery capacity to power the Phase 1 systems and instrumentation up to 8 hours and allow time for the FLEX diesel generators (DGs) to be connected and operated to provide power to charge the batteries.

3.2.1.2 Phase 2

In the Columbia FIP, the licensee states that RCIC will continue to be used until it becomes necessary to transfer to a portable FLEX pump. To support this operation, the connection of a 480 Vac FLEX DG will be accomplished prior to the station batteries dropping below the required voltage. The FLEX DG will repower a station battery charger to support operation of RCIC and the key parameters as discussed further below.

The licensee plans to open the hardened containment vent to vent steam from the wetwell to the atmosphere and maintain the suppression pool below 240 °F. The vent would be opened at 6 hours after the start of the event. To maintain suppression pool level the licensee will initiate water makeup to the SFP. When the SFP is full, valve alignments will allow the overflow water to drain through the fuel pool cooling system and residual heat removal system piping to the suppression pool. Supplying cooler water to the suppression pool and maintaining water level in the suppression pool will ensure that net positive suction head and adequate lube oil cooling is maintained for the RCIC pump and turbine.

Per the FIP, prior to depletion of the backup nitrogen system in containment, additional bottles of nitrogen will be attached to allow for continued SRV operation. The additional bottles allow for SRV use for about 50 hours. Prior to depletion of the nitrogen, the licensee plans to use air compressors from the FLEX Support Buildings (FSBs) to maintain adequate gas pressure to the SRVs.

When RCIC is no longer available, the RPV makeup strategy transitions to low pressure makeup to the RPV. The strategy involves using a portable pump such as the B.5.b pumper truck to transfer water from a spray pond to the RPV. The pump is deployed near a spray pond and then a hose is run from the pump discharge to a tee connector in the reactor building. From this tee, the SFP and the RPV can be provided with makeup water to maintain cooling. The primary and alternate strategies involve using the installed piping of the RHR system and connecting to one of three valves, RHR-V-63A, RHR-V-63B, or RHR-V-63C. The RHR piping is fully protected from all hazards for the event.

Per the FIP, the usage of a spray pond will provide adequate volume for the core cooling strategy into phase 3. The size and capacity of the pumps as well as the two safety-related spray ponds are discussed further in later sections of this safety evaluation. Other water tanks are available on site but not fully protected from all hazards.

In the event that raw water such as the spray ponds is used to provide core cooling, Columbia will utilize the guidance contained in the BWR Owners Group (BWROG) document BWROG-TP-14-006. This guidance contains direction to maintain the water level in the RPV above the level of the moisture separator drains to ensure that core cooling is maintained despite the possible clogging of fuel element inlet orifices and filters. At this water level, water can enter the fuel elements from both the bottom and the top.

#### 3.2.1.3 Phase 3

According to the Columbia FIP, the Phase 3 strategy would be to maintain and supplement or replace the Phase 2 strategy with Phase 3 equipment. The Phase 3 equipment arrives from the NSRC starting at 24 hours after notifying the NSRC and then can be connected to replace Phase 2 components. The Phase 2 connection points are compatible with the equipment that will be arriving from the NSRC.

### 3.2.2 Variations to Core Cooling Strategy for Flooding Event

As described in Section 3.5.2 below, the current flooding design basis for Columbia is that the Seismic Category I structures are not susceptible to external flooding from the probable maximum flood event. In its FIP, the licensee stated that both FSBs are also located above the water level resulting from the probable maximum flood, and that although the equipment deployment route from one building (Building 600) could be affected, the deployment route from the other building would not be affected. The licensee will evaluate the effects of local intense precipitation in a future submittal. Therefore, the staff concurs with the licensee that the current strategies should not be affected by the design-basis flooding.

### 3.2.3 Staff Evaluations

#### 3.2.3.1 Availability of Structures, Systems, and Components (SSCs)

Guidance document NEI 12-06 provides guidance that the baseline assumptions have been established on the presumption that other than the loss of the ac power sources and normal access to the UHS, installed equipment that is designed to be robust with respect to design basis external events is assumed to be fully available. Installed equipment that is not robust is assumed to be unavailable. Below are the baseline assumptions for the availability of SSCs for core cooling during an ELAP caused by a BDBEE.

##### 3.2.3.1.1 Plant SSCs

Phase 1

In the FIP, Section 3.1.1 states that the primary strategy for core cooling is to supply high quality water via the RCIC system. The RCIC system consists of a steam-driven turbine pump that gets motive steam from the RPV and takes suction from the CST or from the suppression pool as a backup. However, the CST is not seismically robust and on low level in the CST the system will automatically swap to the suppression pool using two dc-powered valves. Operators have redundant diverse indication and can manually switch RCIC suction to the suppression pool if necessary. Furthermore, station procedure ABN-RCIC-START, "RCIC Start without AC and DC Power," Revision 7, directs operators to manually operate RCIC in the event that automatic operation is not available. As described in FIP Section 3.1.4.2, the RCIC system is located in the reactor building which is robust and provides protection from all applicable hazards. In addition, most of the RCIC system and its instrumentation and controls are designed to Seismic Category I specifications. Seismic Category I equipment is designed to remain functional after a safe shutdown earthquake. Drain and instrumentation lines are the only piping systems not designed to Seismic Category I standards. However, these piping systems are supported by pipe hangers designed to Seismic Category I criteria. Additionally, the drain line isolates upon RCIC initiation using two air-operated Seismic Category I valves that fail closed on a loss of operating air. Therefore, the NRC staff finds the RCIC system, including the steam-driven turbine pump and the suppression pool, is a robust system and is expected to be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

During decay heat removal, the primary strategy for reactor pressure control is by operation of the SRVs. These valves require dc control power from the station's batteries to operate. In addition, they require pneumatic pressure for operation. In the FIP, Section 3.1.4.3 states that seven of the 18 total SRVs are used for the automatic depressurization system (ADS). These seven ADS valves are equipped with air accumulators and backup nitrogen bottles to ensure the valves will operate following the loss of the normal air supply. Additionally, the licensee has FLEX air compressors to provide pneumatic capability if the backup nitrogen bottles are exhausted. The SRVs are located in the primary containment structure, and some controls are located in the reactor building. These structures are robust to all applicable hazards. In the FIP, the licensee stated that four of the ADS SRVs are environmentally qualified for 24 hours following a loss-of-coolant accident (LOCA), while the other three are environmentally qualified for the full post-LOCA time frame. The NRC staff finds the ADS SRVs are robust and are expected to be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

Phase 2

The licensee's Phase 2 strategy continues to use the suppression pool as the heat sink for SRV discharges and RCIC turbine steam exhaust. The RCIC will continue to be used for RPV makeup with suction from the suppression pool. The wetwell will be vented to atmosphere to remove heat from the suppression pool. The licensee has designed the wetwell vent system to remain functional following a safe shutdown earthquake. The suppression pool will be refilled from cascading overflow from makeup to the SFP using a FLEX pump. In the FIP, Section 3.1.2 states that a FLEX pump will be deployed that can inject water into the RPV through primary or alternate connection points. The licensee plans to rely on the FLEX connection points and water sources discussed in Sections 3.7 and 3.10, respectively, of this safety evaluation (SE). In addition, the SRVs will continue to be used for pressure control.

Phase 3

The licensee's Phase 3 core cooling strategy initially relies on Phase 2 strategies with the NSRC equipment providing backup equipment.

3.2.3.1.2 Plant Instrumentation

The licensee's plan for Columbia is to monitor instrumentation in the control room, and by alternate means if necessary, to support the FLEX cooling strategy. The instrumentation is powered by station batteries and will be maintained for indefinite coping via battery chargers powered by the FLEX DGs. A more detailed evaluation of the instrumentation power supply is contained in Section 3.2.3.6 of this SE.

As described in the Columbia FIP, the following instrumentation will be relied upon to support FLEX core cooling and inventory control strategy:

- RPV Level (Wide Range)
- RPV Pressure
- Drywell Pressure
- Suppression Pool (Wetwell) Pressure
- Suppression Pool Level
- Drywell Temperature
- Suppression Pool Temperature.

These instruments can be monitored from the Main Control Room, the Remote Shutdown Panel, the alternate remote shutdown panel, or locally at instrument racks.

The instrumentation identified by the licensee to support its core cooling strategy is consistent with the recommendation specified in the endorsed guidance of NEI 12-06.

Per the FIP, instrumentation is normally powered by station batteries during the ELAP event. The batteries are charged within 8 hours after the ELAP event initiation by use of a FLEX generator which provides power to the battery chargers through emergency buses. If FLEX generators are not immediately available, the battery power is extended by load shedding to maintain availability of instruments. The FLEX generators continue to provide power through all phases. Additional backup generators will be available from an NSRC during Phase 3. Based upon the information provided by the licensee, the NRC staff understands that the locations of the instrument indications would be accessible continuously throughout the ELAP event.

In accordance with NEI 12-06 Section 5.3.3.1, guidelines for obtaining critical parameters locally are provided in a Flex Support Guideline (FSG). The Columbia guideline ABN-FSG-001, "Accessing Essential Instrumentation during Extended Loss of AC Power with no Power Available," provides alternate methods for obtaining critical parameters if key parameter instrumentation is unavailable. The SFP level instruments are discussed in Section 4 below.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee concluded that its mitigating strategy for reactor core cooling would be adequate based in part on thermal-hydraulic analysis performed using Version 4 of the Modular Accident Analysis Program (MAAP). Because the thermal-hydraulic analysis for the reactor core and containment during an ELAP event are closely intertwined, as is typical of BWRs, Columbia has addressed both in a single, coupled calculation. This dependency notwithstanding, the NRC staff's discussion in this section of the SE focuses on the licensee's analysis of reactor core cooling. The NRC staff's review of the licensee's analysis of containment thermal-hydraulic behavior is provided subsequently in Section 3.4.4.2 of this evaluation.

The MAAP is an industry-developed, general-purpose thermal-hydraulic computer code that has been used to simulate the progression of a variety of light water reactor accident sequences, including severe accidents such as the Fukushima Dai-ichi event. Initial code development began in the early 1980s, with the objective of supporting an improved understanding of and predictive capability for severe accidents involving core overheating and degradation in the wake of the accident at Three Mile Island Nuclear Station, Unit 2. Currently, maintenance and development of the code is carried out under the direction of the Electric Power Research Institute (EPRI).

To provide analytical justification for their mitigating strategies in response to Order EA-12-049, a number of licensees for BWRs and pressurized-water reactors (PWRs) completed analysis of the ELAP event using Version 4 of the MAAP code (MAAP4). Although MAAP4 and predecessor code versions have been used by industry for a range of applications, such as the analysis of severe accident scenarios and probabilistic risk analysis (PRA) evaluations, the NRC staff had not previously examined the code's technical adequacy for performing best-estimate simulations of the ELAP event. In particular, due to the breadth and complexity of the physical phenomena within the code's calculation domain, as well as its intended capability for rapidly simulating a variety of accident scenarios to support PRA evaluations, the NRC staff observed that the MAAP code makes use of a number of simplified correlations and approximations that should be evaluated for their applicability to the ELAP event. Therefore, in support of the reviews of licensees' strategies for ELAP mitigation, the NRC staff audited the capability of the MAAP4 code for performing thermal-hydraulic analysis of the ELAP event for both BWRs and PWRs. The NRC staff's audit review involved a limited review of key code models, as well as confirmatory analysis with the TRACE code to obtain an independent assessment of the predictions of the MAAP4 code.

To support the NRC staff's review of the use of MAAP4 for ELAP analyses, in June 2013, the EPRI issued a technical report entitled "Use of Modular Accident Analysis Program (MAAP) in Support of Post-Fukushima Applications" [Reference 55]. The document provided general information concerning the code and its development, as well as an overview of its physical models, modeling guidelines, validation, and quality assurance procedures.

Based on the NRC staff's review of EPRI's June 2013 technical report, as supplemented by further discussion with the code vendor, audit review of key sections of the MAAP code documentation, and confirmation of acceptable agreement with NRC staff simulations using the TRACE code, the NRC staff concluded that, under certain conditions, the MAAP4 code may be used for best-estimate prediction of the ELAP event sequence for BWRs.

The NRC staff issued an endorsement letter dated October 3, 2013 (ADAMS Accession No. ML13275A318), which documented these conclusions and identified specific limitations that BWR licensees should address to justify the applicability of simulations using the MAAP4 code for demonstrating that the requirements of Order EA-12-049 have been satisfied.

During the audit process, the NRC staff verified that the licensee's MAAP4 calculation, along with an associated addendum, addressed the limitations from the NRC staff's endorsement letter. The licensee utilized the generic roadmap and response template that had been developed by EPRI to support consistency in individual licensee's responses to the limitations from the endorsement letter. In particular, based upon review of the MAAP4 calculation documentation, the staff concluded that appropriate inputs and modeling options had been selected for the code parameters expected to have dominant influence for the ELAP event. The NRC staff further observed that the limitations imposed in the endorsement letter, particularly those concerning the RPV collapsed liquid level being maintained above the reactor core and the primary system cooldown rate being maintained within Technical Specification limits, were satisfied. Specifically, the licensee's analysis calculated that Columbia would maintain the collapsed liquid level in the reactor vessel above the top of the active fuel region throughout the analyzed ELAP event. By maintaining the reactor core fully covered with water, adequate core cooling is assured for this event. Additionally, Columbia's fulfillment of the endorsement letter condition regarding the primary system cooldown rate signifies that thermally induced volumetric contraction and other changes in primary system thermal-hydraulic conditions should proceed relatively slowly with time, which supports the NRC staff's confidence in the predictions of the MAAP4 code. Furthermore, that the licensee should be capable of maintaining the entire reactor core submerged throughout the ELAP event is consistent with the staff's expectation that the licensee's flow capacity for primary makeup (i.e., installed RCIC pump and, subsequently, FLEX pumps) should be sufficient to support adequate heat removal from the reactor core during the analyzed ELAP event, including potential losses due to expected primary leakage.

Therefore, based on the evaluation above, the NRC staff concludes that the licensee's analytical approach should appropriately determine the sequence of events for reactor core cooling, including time-sensitive operator actions, and evaluate the required equipment to mitigate the analyzed ELAP event, including pump sizing and cooling water capacity.

3.2.3.3 Recirculation Pump Seals

An ELAP event would result in the interruption of cooling to the recirculation pump seals, potentially resulting in increased leakage due to the distortion or failure of the seals, elastomeric O-rings, or other components. Sufficient primary make-up must be provided to offset recirculation pump seal leakage and other expected sources of primary leakage in addition to removing decay heat from the reactor core.

The licensee's calculations for Columbia assumed a total primary system leakage rate of 25 gpm at normal RPV operating pressure. This leakage rate was not based on the 18 gpm per recirculation pump seal listed in NRC Generic Letter 91-07. Columbia has two recirculation pumps. The licensee provided data to demonstrate the leakage that Columbia has historically experienced with the recirculation pump seals. The licensee's calculation assumed a primary system leakage rate equal to the Technical Specification LCO 3.4.4 limit of 25 gpm. This leakage rate was used in the Columbia MAAP4 analysis.

During the audit, the NRC staff discussed recirculation pump seal leakage with the licensee and requested that the licensee justify the applicability of the assumed leakage rate to the ELAP event. In the FIP, the licensee stated that the seal leakage rate and total RCS leakage rate will be proportional to the RPV pressure. Based on the licensee's analysis, the limiting flow rate for the FLEX pump would have a margin of 135 gpm above the leak rate of the seals and the flow needed for decay heat removal.

Considering the above factors the NRC staff concludes that the leakage rate of 25 gpm is reasonable as gross seal failures are not anticipated to occur during the postulated ELAP event. As is typical of the majority of U.S. BWRs, Columbia has an installed steam-driven pump (i.e., RCIC) capable of injecting into the RPV at a flow rate well in excess of the primary system leakage rate expected during an ELAP event, and the other pumps used for core cooling in its FLEX strategy have a similar functional capability. As discussed previously, at the limiting pressure the FLEX pump is able to inject at a rate which maintains adequate margin.

Based upon the discussion above, the NRC staff concludes that the recirculation pump seal leakage rates assumed in the licensee's thermal-hydraulic analysis may be applied to the beyond-design-basis ELAP event for the site.

3.2.3.4 Shutdown Margin Analyses

As described in Columbia's Final Safety Analysis Report (FSAR), the control rods provide adequate shutdown margin under all anticipated plant conditions, with the assumption that the highest-worth control rod remains fully withdrawn. Columbia Technical Specification Section 1.1, "Definitions," further clarifies that shutdown margin is to be calculated for a cold, xenon-free condition to ensure that the most reactive core conditions are bounded.

Based on the NRC staff's audit review, the licensee's ELAP mitigating strategy maintains the reactor within the envelope of conditions analyzed by the licensee's existing shutdown margin calculation. Furthermore, the existing calculation retains conservatism because the guidance in NEI 12-06 permits analyses of the beyond-design-basis ELAP event to assume that all control rods fully insert into the reactor core. Therefore, based on the evaluation above, the NRC staff concludes that the sequence of events in the proposed mitigating strategy should result in acceptable shutdown margin for the analyzed ELAP event.

3.2.3.5 FLEX Pumps and Water Supplies

The licensee relies on one of two available portable diesel-driven pumps during Phase 2. One is a trailer-mounted pump, and the other is a truck-mounted pump. In the FIP, Section 3.1.4.5 states the performance criteria (e.g., flow rate and head) for these pumps. See Section 3.10 of this SE for a detailed discussion of the availability and robustness of each water source.

When RCIC is no longer available, a FLEX pump is used to inject water into the RPV from a spray pond. As described in FIP Section 3.1.2, this pump takes suction from one of the two spray ponds. One FLEX pump is rated at 600 gpm at 274 psig, the other is rated at 500 gpm at 270 psig. To verify this pump capacity is within requirements, the licensee performed hydraulic analysis ME-02-12-06, Revision 2, "Evaluation of the Use of Portable Equipment during an Extended Station Blackout," to verify the volumetric flow rate and head needed to remove decay heat following a BDBEE.

The staff was able to confirm that either FLEX pump capacity was within the required capacity from the hydraulic analysis. During the onsite audit, the staff conducted a walk down of the hose deployment routes for the FLEX pump to confirm the evaluations of the pump staging locations, hose distance runs, and connection points as described in the hydraulic analysis and the FIP.

In the FIP, Section 3.1.4.5 states that two FLEX pumps are available, satisfying the need for a spare pump (N+1, where N is the number of units on a site). The FLEX pumps described in this section are also used as described in SE Section 3.3.4.3 for SFP make-up in order to conform to the guidance in NEI 12-06, Section 3.2.2.16. FLEX equipment, including the FLEX pumps, is primarily stored in one of two FSBs, Buildings 600 and 82. The spare equipment is not in the same building as the primary equipment. In the FIP, Section 5.1 describes the capability of these buildings to withstand applicable hazards and protect the equipment contained within.

Based on the staff's review of the FLEX pumping capabilities at Columbia, as described in the above hydraulic analyses and the FIP, the NRC staff concludes that either of the two portable FLEX pumps should perform as intended to support core cooling and RCS inventory control during an ELAP event, consistent with NEI 12-06, Section 11.2.

3.2.3.6 Electrical Analyses

The licensee's electrical strategies provide power to the equipment and instrumentation used to mitigate the ELAP. The electrical strategies described in the FIP are practically identical for maintaining or restoring core cooling, containment, and SFP cooling, except as noted in Sections 3.3.4.4 and 3.4.4.4 of this SE.

The NRC staff reviewed the licensee's FIP, conceptual electrical single-line diagrams and the summary of calculations for sizing the FLEX generators and station batteries. The NRC staff also reviewed the licensee's evaluations that addressed the effects of temperature on the electrical equipment credited in the FIP as a result of the loss of heating, ventilation, and air conditioning (HVAC) caused by the event.

According to the licensee's FIP, operators would declare an ELAP following a loss of offsite power, emergency diesel generators (EDGs), and any alternate ac source. The plant's indefinite coping capability is attained through the implementation of pre-determined diverse and flexible coping strategies (FLEX strategies) that are focused on maintaining or restoring key plant safety functions. A safety function-based approach provides consistency with, and allows coordination with, existing plant emergency operating procedures (EOPs). The FLEX strategies are implemented in support of EOPs using FSGs.

During the first phase of an ELAP event, Columbia would rely on the safety-related Class 1E batteries to provide power to key instrumentation and applicable dc components. The Columbia Class 1E station batteries and associated dc distribution systems are located within safety-related structures designed to meet applicable design-basis external hazards. The licensee's procedure PPM 5.6.2, "Station Blackout and Extended Loss of AC Power ELAP," Major Revision 010, directs operators to conserve dc power during the event by stripping non-essential loads. Operators will strip or shed unnecessary loads to extend battery life until backup power is available in Phase 2. The plant operators would commence load shedding of the batteries within 45 minutes and complete load shedding of the 125 Vdc batteries within 1 hour and load shedding of the 250 Vdc batteries within 2 hours from the onset of an ELAP event.

As part of its mitigating strategies, the licensee is crediting two Class 1E 125 Vdc batteries (E-B1-1 and E-B1-2) and one Class 1E 250 Vdc battery (E-B2-1). Enersys manufactured each battery. The 125 Vdc station batteries are model GN-15 with a nominal capacity of 1260 ampere-hours (AH). The 250 Vdc station battery is model GN-15 with a nominal capacity of 2520 AH. The battery coping time for batteries E-B1-1 and E-B1-2 is about 10 hours while the coping time for battery E-B2-1 is about 17 hours.

The NEI white paper, "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," (ADAMS Accession No. ML13241A186), provides guidance for calculating extended duty cycles of batteries (i.e., beyond 8 hours). This paper was endorsed by the NRC (ADAMS Accession No. ML13241A188). In addition to the white paper, the NRC sponsored testing at Brookhaven National Laboratory that resulted in the issuance of NUREG/CR-7188, "Testing to Evaluate Extended Battery Operation in Nuclear Power Plants," in May of 2015. The testing provided additional validation that the NEI white paper method was technically acceptable. The NRC staff reviewed the licensee's battery calculations and confirmed that they had followed the guidance in the NEI white paper.

The NRC staff reviewed the licensee's dc coping calculation 2.05.01, "Calculation for Battery and Battery Charger 250VDC, 125VDC, and 24VDC," Revision 11, which verified the capability of the dc system to supply power to the required loads during the first phase of the Columbia FLEX mitigation strategy plan for an ELAP. The licensee's evaluation identified the required loads and their associated ratings (ampere (A) and minimum required voltage) and the non-essential loads that would be shed within 1 hour for the 125 Vdc station batteries to ensure battery operation for at least 8 hours and within 2 hours for the 250 Vdc station battery to ensure battery operation for at least 17 hours.

Based on the staff's review of the licensee's analysis and procedures, and the battery vendor's capacity and discharge rates for the Class 1E station batteries, the NRC staff finds that the

Columbia dc systems have adequate capacity and capability to power the loads required to mitigate the consequences during Phase 1 of an ELAP, provided that necessary load shedding is completed within the times assumed in the licensee's analysis.

The licensee's Phase 2 strategy includes repowering the Class 1E battery chargers within 8 hours after initiation of an ELAP to maintain availability of instrumentation to monitor key parameters. The licensee's strategy relies on one 400 kilowatt (kW), 480 Vac FLEX diesel generator (DG). The licensee has a total of two 400 kW, 480 Vac FLEX DGs (one prestaged outdoors and one stored in Building 600). Only one of the FLEX DGs are required for the Phase 2 electrical strategy. The 480 Vac FLEX DG would provide power to the 125/250 Vdc battery chargers and other selected loads.

The NRC staff reviewed the licensee's FIP, calculation E/I-02-91-03, "Calculation for Division 1 and 2 and 3 Diesel Generator Loading," Revision 19, CMR/EC 11763, Revision 1, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the EDGs. In its FIP, the licensee noted that the expected loading on the FLEX DG would be 230 kW. However, the NRC staff's review of the calculation (E/I-02-91-03) found that the minimum required loads during Phase 2 for the licensee's 400 kW FLEX DGs is 285.7 kW (302.8 kW when connected to support transitioning to Phase 3). The NRC staff notified the licensee of the discrepancy which resulted in the licensee revising calculation E/I-02-91-03 via CMR/EC 11763, Revision 1. The revised limits are 195.4 kW for the primary configuration and 279.1 kW for the alternate configuration. Based on the maximum assumed site temperature, the licensee has de-rated the FLEX DGs by 2 percent (i.e., total available capacity is 392 kW at 115 °F). The licensee took the FLEX cable lengths into consideration (i.e., ensured that the voltage drop did not exceed the minimum voltage required at the limiting component). Based on its review of the licensee's calculation, the NRC staff finds that a 400 kW FLEX DG is adequate to support the electrical loads required for the licensee's Phase 2 strategies. Licensee procedures ABN-FSG-003, "DG4 Crosstie to E-MC-7A and E-MC-8A," Major Revision 004, and ABN-FSG-004, "DG5 Crosstie to E-MC-7A and E-MC-8A," Major Revision 004, provide direction for connecting a FLEX DG to the electrical buses to supply required loads within the required timeframes.

For Phase 3, the licensee plans to continue the Phase 2 coping strategy with additional assistance provided from offsite equipment/resources. The offsite resources that will be provided by an NSRC includes two 1-megawatt (MW) 4160 Vac combustion turbine generators (CTGs), one 1100 kW 480 Vac CTG, and distribution panels (including cables and connectors). Licensee procedure ABN-FSG-NSRC-001, "NSRC 4160V DG Crosstie via DG1, DG2 or SM-3 to SM-7 or SM-8," Major Revision 001, includes a table of plant loads with kW ratings that could be powered by the NSRC 4160 Vac CTGs. The licensee's procedure allows for these loads to be managed such that they fall within the rating of the 4160 Vac CTGs. The NSRC 4160 Vac CTGs would supply power to one RHR pump, related valves, and miscellaneous required loads when connected in parallel. The miscellaneous required loads include a fuel pool recirculating pump, the battery chargers, and the room coolers for the control room, cable spreading room, switchgear room, and RHR room. The licensee could also utilize the NSRC 480 Vac CTG as a backup to the Phase 2 FLEX DGs (using the same staging area and connections), if necessary. Licensee procedures ABN-FSG-NSRC-001, and ABN-FSG-NSRC-003, "NSRC 480V DG Crosstie to E-MC-7A and E-MC-8A," Major Revision 004, provide direction for connecting NSRC-supplied CTGs to the electrical buses to supply required loads within the required timeframes, if necessary. Based on its review, the NRC staff finds that the equipment being supplied from an NSRC should have sufficient capacity and capability to supply the required loads during Phase 3.

Based on its review, the NRC staff finds that the Class 1E station batteries should have sufficient capacity to support the licensee's strategy, and that the FLEX DGs and CTGs should have sufficient capacity and capability to supply the necessary loads during an ELAP event.

### 3.2.4 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that should maintain or restore core cooling and RCS inventory during an ELAP event with LUHS consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Table 3-1 and Appendix C summarize an approach consisting of two separate capabilities for the SFP cooling strategies. This approach uses a portable injection source to provide the capability for 1) makeup via hoses on the refueling floor capable of exceeding the boil-off rate for the design-basis heat load; and 2) makeup via connection to spent fuel pool cooling piping or other alternate location capable of exceeding the boil-off rate for the design basis heat load. However, in JLD-ISG-2012-01, Revision 1 [Reference 7], the NRC staff did not fully accept this approach, and added another requirement to either have the capability to provide spray flow to the SFP, or complete an SFP integrity evaluation which demonstrates that a seismic event would have a very low probability of inducing a crack in the SFP or its piping systems so that spray would not be needed to cool the spent fuel. The evaluation must use the reevaluated seismic hazard described in Section 3.5.1 below if it is higher than the site's current safe shutdown earthquake (SSE). During the event, the licensee selects the SFP makeup method to use based on plant conditions. This approach also requires a strategy to mitigate the effects of steam from the SFP, such as venting.

As described in NEI 12-06, Section 3.2.1.7, and JLD-ISG-2012-01, Section 2.1, strategies that must be completed within a certain period of time should be identified and a basis that the time can be reasonably met should be provided. In NEI 12-06, Section 3 provides the performance attributes, general criteria, and baseline assumptions to be used in developing the technical basis for the time constraints. Since the event is beyond design basis, the analysis used to provide the technical basis for time constraints for the mitigation strategies may use nominal initial values (without uncertainties) for plant parameters, and best-estimate physics data. All equipment used for consequence mitigation may be assumed to operate at nominal setpoints and capacities. In NEI 12-06, Section 3.2.1.2 describes the initial plant conditions for the at-power mode of operation; Section 3.2.1.3 describes the initial conditions; and Section 3.2.1.6 describes SFP initial conditions.

In NEI 12-06, Section 3.2.1.1 provides the acceptance criterion for the analyses serving as the technical basis for establishing the time constraints for the baseline coping capabilities to maintain SFP cooling. This criterion is keeping the fuel in the SFP covered with water. The ELAP causes a loss of cooling in the SFP. As a result, the pool water will heat up and eventually boil off. The licensee's response is to provide makeup water. The timing of operator actions and the required makeup rates depend on the decay heat level of the fuel assemblies in the SFP. The sections below address the response during operating, pre-fuel transfer or post-fuel transfer operations. The effects of an ELAP with full core offload to the SFP is addressed in Section 3.11. The licensee has decided to provide the spray flow described in JLD-ISG-2012-01.

3.3.1 Phase 1

The licensee stated in its FIP that no actions are required during ELAP Phase 1 for SFP makeup because the time to boil is sufficient to enable deployment of Phase 2 equipment. Adequate SFP inventory exists to provide radiation shielding for personnel well beyond the time of boiling. The licensee will monitor SFP water level using reliable SFP level instrumentation installed per Order EA-12-051.

3.3.2 Phase 2

During Phase 2, the FIP Section 3.2.1 states that operators will deploy a portable FLEX pump to supply water from the service water spray ponds to the SFP. The FLEX pump discharge can be routed to a connection to the RHR system which has a cross-tie to the SFP cooling system (not requiring refueling floor access), or routed to the refuel floor to provide direct makeup to the pool or provide spray flow via portable nozzles.

3.3.3 Phase 3

The FIP states that SFP cooling can be maintained indefinitely using the makeup strategies described in Phase 2 above. A portion of the NSRC equipment available for Phase 3 can provide SFP cooling and a portion can provide backup capability to the Phase 2 FLEX equipment.

3.3.4 Staff Evaluations

3.3.4.1 Availability of Structures, Systems, and Components

3.3.4.1.1 Plant SSCs

Condition 6 of NEI 12-06, Section 3.2.1.3, states that if permanent plant equipment is contained in structures with designs that are robust with respect to seismic events, floods, and high winds and associated missiles, that equipment should be available. In addition, Section 3.2.1.6 states that the initial SFP conditions are: 1) all boundaries of the SFP are intact, including the liner, gates, transfer canals, etc., 2) although sloshing may occur during a seismic event, the initial loss of SFP inventory does not preclude access to the refueling deck around the pool and 3) the SFP cooling system is intact, including attached piping.

The staff reviewed the licensee's calculation on habitability on the SFP refuel floor. This calculation and the FIP indicate that boiling begins after 30 hours during a normal, non-outage situation. The staff noted that the licensee's sequence of events timeline in the FIP indicates that operators will deploy hoses and spray nozzles as a contingency for SFP makeup within 12 hours from event initiation while the SFP area is habitable for personnel entry. The licensee's first choice would be to provide water directly into the top of the SFP. If that is not accessible, the second choice would be to spray water into the SFP from a distance. If the refueling floor is not accessible, the third choice is to supply water to the RHR connection and cross-tie to the SFP cooling system piping to get water into the SFP.

As described in the licensee's FIP, the licensee's Phase 1 SFP cooling strategy does not require any operator actions. However, the licensee does establish a ventilation path to cope with temperature, humidity and condensation from evaporation and/or boiling of the SFP. The operators are directed to open personnel doors and a refueling floor hatch in the reactor building to establish the ventilation path.

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves the use of the FLEX pump (or NSRC-supplied pump for Phase 3), with suction from the SW spray ponds, to supply water to the SFP. The staff's evaluation of the robustness and availability of FLEX connection points for the FLEX pump is discussed in Section 3.7.3.1 below. Furthermore, the staff's evaluation of the robustness and availability of the SW spray ponds for an ELAP event is discussed in Section 3.10.3.

3.3.4.1.2 <u>Plant Instrumentation</u>

In its FIP, the licensee stated that the instrumentation for SFP level will meet the requirements of Order EA-12-051. Furthermore, the licensee stated that these instruments will have initial local battery power with the capability to be powered from the FLEX DGs. The NRC staff's review of the SFP level instrumentation, including the primary and back-up channels, the display to monitor the SFP water level, and environmental qualifications to operate reliably for an extended period are discussed in Section 4 of this safety evaluation.

3.3.4.2 <u>Thermal-Hydraulic Analyses</u>

As described in Section 3.2 of the FIP, for design operating heat load, the SFP will boil sometime after 30 hours. Section 3.2 of the FIP states that the two bounding scenarios analyzed are: (1) maximum normal operation heat load and (2) the maximum refueling heat load which includes a full core offload. The heat loads can be found in the table below.

| | Heat Load |
|---|---|
| Case 1 | 8.2 million Btu/hr |
| Case 2 | 44.3 million Btu/hr |

The licensee performed calculation ME-02-14-02, "General Technical Support for Fukushima Related Licensing Documents," Revision 1, which determined that an SFP makeup flow rate of at least 100 gpm for the maximum heat load outlined in Case 2 will maintain adequate SFP level makeup for water inventory lost to evaporation and boiling and maintain water level above the fuel for an ELAP occurring during normal power operation. Consistent with this guidance in NEI 12-06, Section 3.2.1.6, the staff finds the licensee has considered the maximum design-basis SFP heat load.

3.3.4.3 <u>FLEX Pumps and Water Supplies</u>

As described in the FIP, the SFP cooling strategy relies on a FLEX pump to provide SFP makeup during Phase 2. In the FIP, Section 3.1.2 describes the hydraulic performance criteria (e.g., flow rate and discharge pressure) for the FLEX pump. The FLEX pumps are the same pumps used for core cooling as described above, and the licensee's analysis showed that a FLEX pump could simultaneously provide makeup water to the RPV and the SFP. The calculation ME-02-12-06 found that either FLEX pump could provide 600 gpm to the SFP for the full core offload scenario. For the at-power scenario, either FLEX pump can provide 300 gpm to the SFP while simultaneously providing 165 gpm to the RPV. The staff noted that the performance criteria of a FLEX pump supplied from an NSRC for Phase 3 would allow the NSRC pump to fulfill the mission of the onsite FLEX pump if the onsite FLEX pump were to fail. As stated above, the SFP makeup rates of 600 gpm and 300 gpm exceed the maximum SFP makeup requirement of 100 gpm as outlined in the previous section of this SE. The licensee also credited some of the extra flow to the SFP as spillover to the suppression pool to replace water lost during the wetwell venting process.

#### 3.3.4.4 Electrical Analyses

The licensee's mitigating strategies for the SFP do not rely on electrical power except for power to SFP level instrumentation. The licensee's electrical SFP cooling strategy for all Phases is to monitor SFP level using installed instrumentation (the capability of this instrumentation is described in Section 4 of this SE). According to the FIP, the Columbia SFP level instrumentation has battery capacity for 7 days. Prior to the battery fully depleting, the licensee could replace the batteries or use portable generators to supply power to the instrumentation and display panels and recharge the installed battery.

Based on its review, the NRC staff finds that the licensee's electrical strategy is acceptable to restore or maintain SFP cooling indefinitely during an ELAP.

### 3.3.5 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that if implemented appropriately should maintain or restore SFP cooling following an ELAP with LUHS consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.4 Containment Function Strategies

The industry guidance document, NEI 12-06, Table 3-1, provides some examples of acceptable approaches for demonstrating the baseline capability of the containment strategies to effectively maintain containment functions during all phases of an ELAP event. One such approach is for a licensee to perform an analysis demonstrating that containment pressure control is not challenged.

The licensee performed a containment evaluation, ME-02-12-18, "Containment Response During Extended Loss of AC Power (ELAP) – A Beyond Design Basis Assessment," which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy of maintaining the RCIC pump suction from the suppression pool and venting the wetwell to atmosphere and concluded that the containment parameters of pressure and temperature remain well below the respective FSAR Section 6.2, Table 6.2-1, design limits of 45 psig and 340 °F for the drywell and 275 °F for the wetwell for more than 72 hours. From its review of the evaluation, the NRC staff noted that the required actions to maintain containment functionality and required instrumentation functions have been developed, and are summarized below. Eventual containment cooling and depressurization to normal values may utilize off-site equipment and resources during Phase 3 if onsite capability is not restored.

3.4.1 Phase 1

For Phase 1, following a loss of ac power, the containment isolation system is assumed to operate normally following the automatic trip of the reactor. The RCIC system is operated to maintain adequate core cooling by providing the RPV with makeup water. Steam flow through the SRVs and the RCIC turbine exhaust is discharged to the suppression pool, removing decay heat from the reactor. At approximately 6 hours after the start of the ELAP event, the suppression pool will be vented to atmosphere through the hardened containment vent system (HCVS) to limit the increase in the suppression pool temperature.

3.4.2 Phase 2

Phase 2 will maintain the Phase 1 strategy of venting the suppression pool. Phase 2 will utilize a FLEX DG to charge safety-related batteries using the existing battery chargers.

The FLEX pump will be deployed to provide an alternate method of supplying makeup water to the RPV in the event the RCIC pump becomes unavailable.

3.4.3 Phase 3

Phase 3 will maintain Phase 1 venting of the suppression pool. A backup generator and diesel-powered pump will be provided from an off-site NSRC to provide additional capacity and redundancy to the on-site FLEX equipment.

The NRC staff noted in the thermal-hydraulic analysis that with the suppression pool venting mitigating strategy the drywell temperature exceeds 300 °F after 72 hours and continues to increase. The drywell maximum design temperature is 340 °F. As part of the audit process, NRC staff reviewed licensee document FLEX-01, "FLEX Program." This document stated that the licensee has developed procedures that can be used to restore the RHR system to remove decay heat from containment for an extended time. A large diesel-driven pump from an NSRC will be available to pump water from the UHS to the service water piping through a connection adapter at the service water pump discharge check valve to the RHR heat exchanger for decay heat removal. In addition the two 4160 Vac generators and switchgear from an NSRC will be available to repower the RHR pump. The NRC staff finds that with off-site equipment provided by the NSRC, the licensee will be able to maintain the containment within design parameters.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

Guidance document NEI 12-06 baseline assumptions have been established on the presumption that other than the loss of the ac power sources and normal access to the UHS, installed equipment that is designed to be robust with respect to design-basis external events is assumed to be fully available. Installed equipment that is not robust is assumed to be unavailable. Below are the baseline assumptions for the availability of SSCs for maintaining containment functions during an ELAP.

3.4.4.1.1 Plant SSCs

Containment

Columbia has a Mark II style primary containment which is filled with nitrogen gas when operating at power. The primary containment structure is a free-standing steel pressure vessel. The vessel contains both a drywell and a suppression chamber, also known as the wetwell. The primary containment employs the pressure suppression concept. Its design employs an over-and-under, steel pressure vessel which houses the reactor vessel, the reactor recirculation system (RRC) loops, and other branch connections of the reactor primary system. The pressure suppression system consists of the drywell, a pressure suppression chamber which stores a large volume of water known as the suppression pool, a connecting submerged vent system between the drywell and the suppression pool, isolation valves, containment cooling system, and other service equipment. In the event of a reactor coolant pressure boundary (RCPB) piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increase in drywell pressure would then force a mixture of gas, steam, and water through the vents into the suppression pool, condensing the steam, resulting in a rapid pressure reduction in the drywell. Non-condensable gases such as nitrogen which are transferred to the suppression chamber will pressurize the suppression chamber, and are subsequently vented back to the drywell when the drywell pressure decreases below the suppression chamber pressure. As mentioned above, FSAR Section 6.2, Table 6.2-1, lists containment design limits of 45 psig and 340 °F for the drywell and 275 °F for the suppression chamber.

Hardened Containment Vent System

The licensee indicated that a hardened containment vent system has been installed in order to meet the Phase 1 requirements of NRC Order EA-13-109. In a response to NRC Order EA-13-109, Energy Northwest's submittal, "Overall Integrated Plan For Reliable Hardened Containment Vents Under Severe Accident Conditions," dated June 30, 2014 [Reference 56], stated that the wetwell HCVS piping will be sized to vent steam with the energy of 1 percent of rated thermal power at a wetwell pressure of 45 psig, with rated thermal power of 3556 megawatts thermal (MWt). That rated thermal power is about 0.3 percent higher than the current rated thermal power of 3544 MWt. This pressure is equal to the containment design pressure and is lower than the primary containment pressure limit stated in the EOPs. The vent piping will also be sized to support system operation and pass 80,000 lb/hr at a maximum pressure of 10 psig in the wetwell. The wetwell vent exits the reactor building at approximately 72 feet above plant grade level near the southeast corner of the reactor building, terminates above the roof's parapet wall, and vents to atmosphere. The vent pipe is independent of the reactor building elevated release point and is located away from any ventilation system intake and exhaust openings or emergency response facilities. The HCVS will be further reviewed by NRC staff as part of Order EA-13-109.

3.4.4.1.2 Plant Instrumentation

In NEI 12-06, Table 3-1 specifies that containment pressure, suppression pool level, and suppression pool temperature are key containment parameters that should be monitored by repowering the appropriate instruments. The licensee's FIP states that some control room instrumentation would be available due to the coping capability of the station batteries and associated inverters in Phase 1, or the portable DGs deployed in Phase 2. If no ac or dc power were available, the FIP states that key credited plant parameters, including these containment parameters, would be available using alternate methods.

Section 3.1.7 of Columbia's FIP identifies drywell pressure, drywell temperature, suppression pool pressure, suppression pool level, and suppression pool temperature as key instrumentation credited for all phases of the FLEX strategies. Plant procedure ABN-FSG-001, "Accessing Essential Instrumentation during Extended Loss of AC Power with no DC Power Available," contains instructions for alternate methods of obtaining key parameters such as drywell and suppression pool pressure and temperature, and suppression pool level, with no ac or dc power.

Based on the battery capacities and availability of FLEX DGs for long-term power supplies, and the ability to use plant procedure ABN-FSG-001 if there is no ac or dc power, the licensee should have the ability to appropriately monitor the key containment parameters as delineated in NEI 12-06, Table 3-1.

#### 3.4.4.2 Thermal-Hydraulic Analyses

The licensee evaluated the response of the containment in calculation ME-02-12-18, "Containment Response during Extended Loss of AC Power (ELAP) – A Beyond Design Basis Assessment," which was based on the boundary conditions described in Section 2 of NEI 12-06. As discussed in Section 3.2.3.2 of this SE the calculation used the MAAP4 computer code, which is a thermal hydraulic analysis program, to perform numeric computations of the fundamental thermodynamic equations which predict the heat up and pressurization of the containment atmosphere under ELAP conditions. The calculation assumes reactor recirculation pump seal leakage to be 25 gpm. As discussed in Section 3.2.3.3 of this SE, the NRC staff accepted this leakage rate for this analysis. The analysis modeled 72 hours post-ELAP to verify FLEX strategies are effective in controlling conditions inside containment within acceptable parameters. Wetwell venting was modeled starting at various times. The calculation determined that with venting starting at 6 hours, suppression pool temperature peaks at 240 °F at approximately 12 hours. After cascade makeup to the suppression pool from spent fuel pool overflow, the suppression pool temperature will gradually decrease to 222 °F. The wetwell pressure peaks at 28.4 pounds per square inch absolute (psia), which is 13.7 psig. The wetwell remains below the design limits of 45 psig and 275 °F, which are listed in FSAR Table 6.2-1. The calculation shows the maximum drywell pressure is 28.2 psia and the drywell temperature reaches 300 °F by 72 hours. The drywell stays below the design limits of 45 psig and 340 °F for the drywell as listed in FSAR Table 6.2-1.

The licensee also performed calculation ME-02-14-13, "Containment Response during Extended Loss of AC Power (ELAP) with Suppression Pool Makeup." This analysis investigated cascading spent fuel pool makeup to the suppression pool for cooling and makeup. The calculation determined the maximum hardened vent flow resistance to keep the maximum suppression pool temperature at or below 240 °F. The information for the vent flow resistance was used as input for ME-02-12-18, above.

The NRC staff reviewed the analyses and concluded that the strategies indicated in the FIP will maintain the containment within the design pressure and temperature limits indicated in FSAR Table 6.2-1.

3.4.4.3 FLEX Pumps and Water Supplies

See Section 3.3.4.3, "FLEX Pumps and Water Supplies," of this SE for a discussion of portable pumps and water supplies.

3.4.4.4 Electrical Analyses

The licensee performed the containment evaluation discussed in Section 3.4.4.2 of this SE based on the boundary conditions described in Section 2 of NEI 12-06. Based on the results of its evaluation, the licensee developed required actions to ensure the maintenance of containment functionality and that required instrumentation continues to function. With an ELAP initiated, while Columbia is in Modes 1-4, containment cooling is lost for an extended period of time. Therefore, containment temperature and pressure will slowly increase.

The licensee's Phase 1 coping strategy is to monitor containment pressure and temperature using installed instrumentation, and maintain containment functionality using normal design features of the containment, such as the containment isolation valves and the HCVS. Control room indication for containment pressure and temperature is available for the duration of the ELAP. The licensee's strategy to power instrumentation using the Class 1E station batteries is identical to what was described in Section 3.2.3.6 of this SE and is adequate to ensure continued containment monitoring. The installed HCVS system has a dedicated 125 Vdc battery. In calculation E/I-02-13-03, "Battery Sizing Calculation for the Hardened Containment Vent (HCV) System," Revision 1, the licensee evaluated the battery/battery charger sizing and device terminal voltages for the 125 Vdc HCVS. Based on the NRC staff's review of this calculation, the results show that the 125 Vdc HCVS battery is adequately sized to supply power to the HCVS for 24 hours following an ELAP.

The licensee's Phase 2 coping strategy is to continue monitoring containment pressure and temperature using installed instrumentation. The licensee's strategy to power instrumentation using a 400 kW FLEX DG is identical to what was described in Section 3.2.3.6 of this SE and is adequate to ensure continued containment monitoring. The licensee would transition to Phase 2 prior to depleting the HCVS battery (i.e., within 24 hours). Licensee procedures ABN-FSG-003 and ABN-FSG-004 provide direction for connecting a FLEX DG to the electrical buses to supply required loads within the required timeframes.

The licensee's Phase 3 strategy is to continue its Phase 2 strategy throughout the event. Columbia will start receiving offsite resources and equipment from an NSRC about 24 hours after notifying the NSRC. This equipment includes two 4160 Vac 1 MW CTGs and a 480 Vac 1100 kW CTG. Given the capacity of these generators, the NRC staff finds that it is reasonable to expect that the licensee could utilize these resources to supply power to the HCVS components to maintain the ability to vent containment indefinitely. Licensee procedures ABN-FSG-NSRC-001 and ABN-FSG-NSRC-003 provide direction for connecting NSRC-supplied CTGs to the electrical buses to supply required loads within the required timeframes, if necessary.

Based on its review, the NRC staff determined that the electrical equipment available onsite (e.g., Class 1E station batteries, HCVS battery, and 400 kW FLEX DGs) supplemented with the electrical equipment that will be supplied from an NSRC, has sufficient capacity and capability to supply the required loads to maintain containment.

### 3.4.5 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain or restore containment functions following an ELAP event with LUHS consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.5 Characterization of External Hazards

Sections 4 through 9 of NEI 12-06 provide the methodology to identify and characterize the applicable BDBEEs for each site. In addition, NEI 12-06 provides a process to identify potential complicating factors for the protection and deployment of equipment needed for mitigation of applicable site-specific external hazards leading to an ELAP and loss of normal access to the UHS.

Characterization of the applicable hazards for a specific site includes the identification of realistic timelines for the hazard, characterization of the functional threats due to the hazard, development of a strategy for responding to events with warning, and development of a strategy for responding to events without warning.

The licensee reviewed the plant site against NEI 12-06 and determined that FLEX equipment should be protected from the following hazards: seismic; external flooding; snow, ice and extreme cold; volcanic ash fall, and extreme high temperatures.

References to external hazards within the licensee's mitigating strategies and this SE are consistent with the guidance in NEI-12-06 and the related NRC endorsement of NEI 12-06 in JLD-ISG-2012-01. Guidance document NEI 12-06 directed licensees to proceed with evaluating external hazards based on currently available information. For most licensees, this meant that the OIP used the current design-basis information for hazard evaluation. Coincident with the issuance of Order EA-12-049, on March 12, 2012, the NRC staff issued a Request for Information under Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f) [Reference 19] (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies. Due to the time needed to reevaluate the hazards, and for the NRC to review and approve them, the reevaluated hazards were generally not available until after the mitigation strategies had been developed. The NRC staff has developed a draft final rule, titled "Mitigation of Beyond-Design-Basis Events," hereafter called the MBDBE rule, which was provided to the Commission for approval on December 15, 2016 [Reference 47]. The MBDBE rule would make the intent of Orders EA-12-049 and EA-12-051 generically applicable to all present and future power reactor licensees, while also requiring that licensees consider the reevaluated hazard information developed in response to the 50.54(f) letter.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in response to the 50.54(f) letter and the requirements for Order EA-12-049 and the MBDBE rulemaking (see COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluaton of Flooding Hazards" [Reference 46]. The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 20]. The Commission approved the staff's recommendations that licensees would need to address the reevaluated flooding hazards within their mitigating strategies for BDBEEs, and that licensees may need to address some specific flooding scenarios that could significantly impact the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition. In a letter to licensees dated September 1, 2015 [Reference 40], the NRC staff informed the licensees that the implementation of mitigation strategies should continue as described in licensee's OIPs, and that the NRC safety evaluations and inspections related to Order EA-12-049 will rely on the guidance provided in JLD-ISG-2012-01, Revision 0, and the related industry guidance in NEI 12-06, Revision 0. The hazard reevaluations may also identify issues to be entered into the licensee's corrective action program consistent with the OIPs submitted in accordance with Order EA-12-049.

As discussed above, licensees are reevaluating the site seismic and flood hazards as requested in the NRC's 50.54(f) letter. After the NRC staff approves the reevaluated hazards, licensees will use this information to perform flood and seismic mitigating strategies assessments (MSAs) per the guidance in NEI 12-06, Revision 2, Appendices G and H [Reference 6]. The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 [Reference 7]. The licensee's MSAs will evaluate the mitigating strategies described in this SE using the revised flooding and seismic hazard information and, if necessary, make changes to the strategies or equipment.

The licensee developed its OIP for mitigation strategies by considering the guidance in NEI 12-06 and the site's design-basis hazards. Therefore, this SE makes a determination based on the licensee's OIP and FIP. The characterization of the applicable external hazards for the plant site is discussed below.

3.5.1 Seismic

In its FIP, the licensee described the current design-basis seismic hazard, the safe shutdown earthquake (SSE). As described in FSAR Section 3.7.1.1, the SSE seismic criteria for the site is twenty-five hundredths of the acceleration due to gravity (0.25g) for the peak horizontal ground acceleration, and two-thirds of that for the peak vertical ground acceleration. It should be noted that the actual seismic hazard involves a spectral graph of the acceleration versus the frequency of the motion. Peak acceleration in a certain frequency range, such as the numbers above, is often used as a shortened way to describe the hazard.

As the licensee's seismic reevaluation activities are completed, the licensee is expected to assess the mitigation strategies to ensure they can be implemented under the reevaluated hazard conditions as will potentially be required by the proposed MBDBE rulemaking. The licensee has appropriately screened in this external hazard and identified the hazard levels to be evaluated.

3.5.2 Flooding

In its FIP, the licensee described that the current design basis for the limiting site flooding event is the probable maximum flood (PMF) event, which is of limited duration and water level. As described in FSAR Section 2.4, the current design basis is that the Seismic Category I structures are not susceptible to external flooding from the PMF event. The licensee stated in its FIP that there is no groundwater inleakage that could affect the FLEX strategy. In this SE the NRC staff will make a determination based on the licensee's current design basis for flooding. By letter dated January 25, 2018 (ADAMS Accession No. ML18025B515), the licensee submitted analyses of the effect of the beyond-design-basis reevaluated flooding hazards. The NRC staff will issue an evaluation of those analyses at a later time. The licensee has appropriately screened in this external hazard and identified the hazard levels to be evaluated.

3.5.3 High Winds

In NEI 12-06, Section 7 provides the NRC-endorsed screening process for evaluation of high wind hazards. This screening process considers the hazard due to hurricanes and tornadoes.

The screening for high wind hazards associated with hurricanes should be accomplished by comparing the site location to NEI 12-06, Figure 7-1 (Figure 3-1 of U.S. NRC, "Technical Basis for Regulatory Guidance on Design Basis Hurricane Wind Speeds for Nuclear Power Plants," NUREG/CR-7005, December, 2009); if the resulting frequency of recurrence of hurricanes with wind speeds in excess of 130 mph exceeds 1E-6 per year, the site should address hazards due to extreme high winds associated with hurricanes using the current licensing basis for hurricanes.

The screening for high wind hazard associated with tornadoes should be accomplished by comparing the site location to NEI 12-06, Figure 7-2, from U.S. NRC, "Tornado Climatology of the Contiguous United States," NUREG/CR-4461, Rev. 2, February 2007; if the recommended tornado design wind speed for a 1E-6 per year probability exceeds 130 mph, the site should address hazards due to extreme high winds associated with tornadoes using the current licensing basis for tornados or Regulatory Guide 1.76, Rev. 1.

In FSAR Section 2.1.1.1 it states that the site is located at 46° 28' 18" North latitude and 119° 19' 58" West longitude. In NEI 12-06, Figures 7-1 and 7-2 indicate the site is not in a region where the wind speed is expected to exceed 130 mph. Therefore, the plant screens out for an assessment for high winds (hurricanes and tornados) for the protection and deployment of FLEX equipment, including missiles produced by these events.

Therefore, high-wind hazards are not applicable to the plant site for the protection and deployment of FLEX equipment. The licensee has appropriately screened out the high wind hazard.

3.5.4 Snow, Ice, and Extreme Cold

As discussed in NEI 12-06, Section 8.2.1, all sites should consider the temperature ranges and weather conditions for their site in storing and deploying FLEX equipment consistent with normal design practices. All sites outside of Southern California, Arizona, the Gulf Coast and Florida are expected to address deployment for conditions of snow, ice, and extreme cold. All sites located north of the 35th Parallel should provide the capability to address extreme snowfall with snow removal equipment. Finally, all sites except for those within Level 1 and 2 of the maximum ice storm severity map contained in Figure 8-2 of NEI 12-06 should address the impact of ice storms.

As previously noted, the site is located at 46° 28' 18" North latitude and 119° 19' 58" West longitude. The licensee noted in the FIP that the site is located within the region characterized by EPRI as ice severity level 3 (NEI 12-06, Figure 8-2, Maximum Ice Storm Severity Maps). Consequently, the site is subject to icing conditions that could cause low to medium damage to electrical transmission lines. The licensee concludes that the plant screens in for an assessment for snow, ice, and extreme cold hazard. In FSAR Chapter 9.4, the winter outside design temperature is listed as 0 °F with an extreme outdoor winter condition of -27 °F.

In summary, based on the available local data and Figures 8-1 and 8-2 of NEI 12-06, the plant site does experience significant amounts of snow, ice, and extreme cold temperatures; therefore, the hazard is screened in. The licensee has appropriately screened in the hazard and characterized the hazard in terms of expected temperatures.

3.5.5 Extreme Heat

In the section of its FIP regarding the determination of applicable extreme external hazards, the licensee stated that, as per NEI 12-06, Section 9.2, all sites are required to consider the impact of extreme high temperatures. The plant site screens in for an assessment for extreme high temperature hazard.

In FSAR Chapter 9.4, the summer outdoor design temperature is listed as 105 °F (dry-bulb) with an extreme outdoor summer condition of 115 °F (dry-bulb). In summary, based on the available local data and the guidance in Section 9 of NEI 12-06, the plant site does experience extreme high temperatures. The licensee has appropriately screened in the high temperature hazard and characterized the hazard in terms of expected temperatures.

3.5.6 Volcanic Ash

In its FIP, the licensee stated that there are several major volcanoes close enough to the site to require the ability to cope with a design-basis ash fall. In FSAR Section 9.2.5.3, it is stated that the design-basis ash fall is 3 inches, which bounds the Mount St. Helens eruption of May 18, 1980. The licensee has appropriately screened in the ash fall hazard and characterized the hazard in terms of expected ash fall.

3.5.7 Conclusions

Based on the evaluation above, the NRC staff concludes that the licensee has developed a characterization of external hazards that is consistent with NEI 12-06 guidance, as endorsed by

JLD-ISG-2012-01, and should adequately address the requirements of the order in regard to the characterization of external hazards.

### 3.6 Planned Protection of FLEX Equipment

#### 3.6.1 Protection from External Hazards

In its FIP, the licensee stated that:

> In accordance with NEI 12-06 , if on-site FLEX equipment is prestaged 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.

In its FIP, the licensee stated that:

> 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.

Below are additional details on how FLEX equipment is protected from each of the applicable external hazards.

##### 3.6.1.1 Seismic

The NRC staff noted that the original design of the two FLEX buildings did not include design criteria to withstand a large seismic event such as the SSE. The licensee reviewed the seismic and wind load design loads of the two FLEX buildings and determined that wind load governs the lateral load capacity of each building in both directions, which is typical in these types of lightweight structures. Wind load on each building is lowest in the longitudinal direction. The licensee determined the seismic ground motion that corresponds to the longitudinal wind load design base shear of each building. The licensee then compared the resulting equivalent seismic ground motion capacities for both buildings to the Columbia SSE and confirmed that they exceed the SSE ground motion. This conclusion is documented in analysis CVI 1266-00,20, "Evaluation of Seismic Ground Motion Capabilities for FLEX Building 82 and Building 600." The NRC staff concludes that these buildings should remain functional following an SSE, allowing the deployment of FLEX equipment from inside the buildings.

3.6.1.2 Flooding

In its FIP, the licensee stated that both FSBs are located above the design-basis flood level from the PMF event. The FLEX DG (E-GEN-DG4) prestaged outside the EDG building is also above that flood level. The details on that design-basis flood are in Section 3.5.2 above.

3.6.1.3 High Winds

Due to the physical location of Columbia, the licensee is not required to show protection for the FLEX equipment from high winds such as tornados or hurricanes. See Section 3.5.3 above for additional details.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

As noted in Sections 3.5.4 and 3.5.5 above, the summer outdoor design temperature for Columbia is 105 °F (dry-bulb) with an extreme outdoor summer condition of 115 °F (dry-bulb) and the winter outdoor design temperature is 0 °F with an extreme outdoor winter condition of -27 °F. In its FIP, the licensee stated that FLEX equipment was procured to function in weather conditions applicable to Columbia and includes block heaters.

3.6.1.5 Volcanic Ash

In addition to the normal air filters supplied with the FLEX equipment, the licensee has purchased oil bath filters, to be used if volcanic ash is falling, to filter combustion air for the protection of diesel-powered FLEX equipment (DG4, DG5, FLEX-P-1, B.5.b Pumper Truck, Building 82 house generator, and the Building 600 house generator). The oil bath filters are stored in the FSBs along with containers of oil. The licensee stated that Buildings 82 and 600 meet the design standards of ASCE 7-10 and are designed to be structurally capable of withstanding expected wind loading and ash fall deposits.

Procedure ABN-ASH, "Ash Fall," Major Revision 026, contains the instruction for installation of the oil bath filters and refilling or replacing the filter's oil. As Columbia occasionally has snow, equipment is already available to remove ice and snow and that equipment can be used to remove any excessive accumulation of ash from the FLEX equipment deployment routes.

Licensee calculation ME-02-15-04, "Potential Effects of Volcanic Ash in Spray Pond Water," Revision 0, demonstrated that by considering the volume of the water in the RPV and SFP and the very small ash particle size (< 1 millimeter), the relatively small amount of ash entering the RPV and SFP will not adversely affect cooling the fuel. Based on this, no special actions are required in the RPV or SFP during or following an ash fall event to support the cooling of the fuel.

Since the 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 the licensee has evaluated the deployment of equipment to ensure required manual actions can be accomplished under ash fall conditions, the NRC staff finds that the licensee's mitigating strategies are adequate to address ash fall during an ELAP event.

3.6.1.6 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should protect the FLEX equipment during a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.6.2 Availability of FLEX Equipment

Section 3.2.2.16 of NEI 12-06 states, in part, that in order to assure reliability and availability of the FLEX equipment, the site should have sufficient equipment to address all functions at all units on-site, plus one additional spare (i.e., an N+1 capability, where "N" is the number of units on site). It is also acceptable to have multiple strategies to accomplish a function, in which case the equipment associated with each strategy does not require an additional spare.

As noted in the evaluation above, the licensee relies on one FLEX pump, with necessary suction and discharge hoses, to provide makeup water to the RPV and the SFP, and one FLEX DG, with necessary electrical hookup cables, to provide electrical generation to maintain vital battery buses and supply power to critical equipment. One FLEX pump (FLEX-P-1), with necessary suction and discharge hoses, is located in Building 82. The second FLEX pump (B5B-P-1, the B5b pumper truck), with necessary suction and discharge hoses, is located in Building 600. One FLEX DG, FLEX-GEN-DG5 (DG5), with necessary electrical hookup cables, is located in Building 600. The second DG, E-GEN-DG4 (DG4), with necessary electrical hookup cables, is prestaged outside near the south side of the EDG building. Due to the physical location of Columbia, the licensee is not required to show protection for the FLEX equipment from high winds such as tornados or hurricanes. The NRC staff noted that DG4 is trailer mounted and can be moved if necessary. It is located in a flat area, with no non-seismic components or structures close enough to damage it in a seismic event. It is located above the flood level for the design-basis flood, and is designed to use a block heater and battery charger when necessary. For the volcanic ash hazard, an oil bath filter is available and stored in an FSB. The NRC staff finds that DG4 satisfies the guidance for FLEX equipment provided in NEI 12-06. Therefore, the licensee has available a suitable amount of equipment to address all functions for Columbia, plus one additional spare (N+1).

Based on the number of portable FLEX pumps, FLEX DGs, and support equipment identified in the FIP and during the audit review, the NRC staff finds that, if implemented appropriately, the licensee's FLEX strategies include a sufficient number of portable FLEX pumps, FLEX DGs, and equipment for RPV makeup and core cooling, SFP makeup, and maintaining containment consistent with the N+1 recommendation in Section 3.2.2.16 of NEI 12-06.

## 3.7 Planned Deployment of FLEX Equipment

Building 82 is located in the plant protected area, while Building 600 is located outside the protected area. The licensee stated that electrical power is not required to deploy equipment, as all the building doors can be manually operated. The licensee identified deployment paths for the FLEX equipment. As documented in FSAR Section 3.4.1.4.2, soil liquefaction following a seismic event is not postulated at the Columbia site due to soil type and unsaturated conditions. However, the normal access road to Building 600 may be flooded in the design-basis site flood. If that happens, the licensee has a plan to create an alternate route to the plant that is above the flood level by removing a portion of the vehicle barrier system. The deployment path from Building 82 would not be affected. The licensee also investigated the effects of a break in a circulating water pipe, possibly caused by a seismic event. Due to the volume and pressure of the water, it may saturate the surrounding soil. The licensee's analysis indicates that soil stability would be maintained, but there may be a small degree of erosion at the ground surface near the failure location. The licensee stated that this level of erosion can be quickly repaired using the dedicated debris removal equipment, and the FLEX deployment route restored.

### 3.7.1 Means of Deployment

In its FIP, the licensee stated that debris removal equipment includes a wheel loader and an excavator, and towing equipment includes a flatbed truck and a 5-yard dump truck. The towing and debris removal equipment were purchased with block heaters to aid starting in cold weather.

### 3.7.2 Deployment Strategies

A FLEX pump will be deployed to one of the two service water spray ponds, which are located southeast of the reactor building. 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 continuing to pump through the second strainer, preventing the loss of the pump suction. If ice forms on the surface of the spray ponds due to cold weather, the excavator can be used to break the ice to allow the suction hose to be placed in the spray pond.

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 could be deployed to one of two locations identified by the licensee. Both FLEX generators have ample cabling so either FLEX generator can be connected to either connection point discussed in Section 3.7.3.2 below.

### 3.7.3 Connection Points

#### 3.7.3.1 Mechanical Connection Points

Core Cooling

In the FIP, Section 3.1.5.3 describes the primary and alternate core cooling connection points for the portable FLEX pump. The FLEX pump will supply water to the connection points via a hose tee that splits the flow to the SFP and the RPV. The primary and alternate connections are located in the three RHR loops at valves RHR-V-63A, RHR-V-63B, or RHR-V-63C using an adaptor with a hose fitting. Water is injected through the RHR injection valves. All three connections are located in the reactor building. In FIP Section 3.1.4.2, the licensee stated the connection points are protected from all applicable hazards.

SFP Cooling

In the FIP, Sections 3.1.5.3 and 3.2.2 discuss the SFP connections. The portable FLEX pump is also used for SFP cooling using a service water spray pond as a suction source. For the primary strategy, hose is run from the discharge tee of the FLEX pump and routed directly into the top of the SFP. No physical connections to permanent plant equipment are required for this strategy. Alternatively, if the refueling floor is not accessible, the RHR to fuel pool cooling (FPC) cross-connect can be used to fill the SFP with the FLEX pump via the RHR-V-63B connection.

Given the design and location of the primary and alternate connection points, as described in the above paragraphs, the staff finds that at least one of the connection points should be available to support core and SFP cooling via a portable pump during an ELAP caused by an external event, consistent with NEI 12-06 Section 3.2.2.

3.7.3.2 Electrical Connection Points

Electrical connection points are only applicable for Phases 2 and 3 of the licensee's mitigation strategies for a BDBEE that result in an ELAP with LUHS.

During Phase 2, the licensee's strategy is to supply power to equipment required to maintain or restore core cooling, containment, and SFP cooling using a combination of permanently installed and portable components.

The licensee's FLEX strategy to re-power the station's battery chargers and other vital equipment requires the use of a 480 Vac, 400 kW DG. Two FLEX DGs are available but only one is necessary. One FLEX generator, DG4, is a previously existing generator used to support Technical Specification 3.8.1, "AC Sources – Operating," Action B.4 Completion Time, allowing the extension of the Completion Time up to 14 days. DG4 is located outside near the EDG building. The other FLEX generator, DG5, is located in Building 600 and will require deployment.

Both of the FLEX DGs are 400 kW standby rating DGs. One 480 Vac connection point is located on the outside wall of the EDG building, which is a Seismic Category 1 structure. A second 480 Vac connection point is located in the radwaste/control building. The cabling and connections are installed seismically and the connection point in the radwaste/control building is a Seismic Category 1 area that also provides protection from snow and ice. Either connection point can be used to supply Division 1 or Division 2 480 Vac electrical loads. Both on-site FLEX 480 Vac DGs have sufficient cabling and connectors to access either connection point. Each on-site FLEX DG is stored with the cabling required to connect the generator to the FLEX connection points. Procedures ABN-FSG-003 and ABN-FSG-004 provide direction for connecting a 480 Vac FLEX DG. During the audit process, the NRC staff confirmed that these procedures will include a step directing operators to verify proper phase rotation when the 480 Vac FLEX cables are connected.

For Phase 3, the licensee will receive two 1 MW 4160 Vac CTGs and one 1100 kW 480 Vac CTG from an NSRC. The NSRC-supplied 4160 Vac CTGs can be connected to the 4160 Vac buses via two diverse bolted-lug connection points. One connection pathway is via the turbine building, the other, via the EDG building. Two storage lockers containing cabling for the connection through the turbine building are located in the turbine building at different locations. The cabling for the connection through the EDG building is supplied with the NSRC generators. Cabling for the connections is color-coded. ABN-FSG-NSRC-001 and ABN-FSG-NSRC-003 provide direction for staging and connecting the 4160 Vac CTGs. ABN-FSG-NSRC-001 provides direction to verify proper phase rotation of the 4160 Vac CTGs.

### 3.7.4 Accessibility and Lighting

In the NRC FLEX audit report [Reference 17], the NRC staff discussed its review of the licensee's strategy for using portable lighting and concluded that the licensee's plan for portable lighting was adequate. In the FIP, the timeline in Attachment 4 shows that the short-term actions involve verifying that RCIC is injecting to the RPV, checking that the SRVs are controlling RPV pressure, and performing a load shed of the vital 125 Vdc batteries. The RCIC system and the SRVs can be operated from the control room, which is located in a Seismic Category I structure and therefore is expected to remain accessible. Although the load shed actions are outside the control room, the areas that need to be accessed are also located in Seismic Category I structures and the NRC staff finds that those areas should remain accessible.

### 3.7.5 Access to Protected and Vital Areas

During the audit process, the licensee provided information describing that access to protected areas will not be hindered. The licensee has contingencies in place to provide access to areas required for the ELAP response if the normal access control systems are without power.

### 3.7.6 Fueling of FLEX Equipment

In the FIP, Section 6.5, the licensee described that procedure SOP-FLEX-EQUIPMENT-REFUEL, "FLEX Equipment Refueling," Revision 1, directs operators to refuel the FLEX equipment using various identified fuel oil sources, including the high pressure core spray (HPCS) diesel fuel storage tank, two EDG fuel storage tanks, and some non-robust tanks. The first three tanks, the HPCS tank and the two EDG storage tanks are Seismic Category I and are located underground. Seismic Category I structures and equipment are designed so that following an SSE the structures will not fail and the equipment will remain functional. Plant Technical Memorandum TM-2185, "Equipment Refueling Strategy of Phase 1 and 2 FLEX Components During the First 72 Hours Following an ELAP," Revision 2, analyzed the amount of fuel oil and gasoline needed and the source. It states that the licensee will first attempt to obtain diesel fuel from the auxiliary boiler fuel tank FO-TK-1, if it remains available. However, it is not Seismic Category I. The licensee's next preferred source of diesel fuel is DO-TK-2, the storage tank for the HPCS diesel. After that the two EDG storage tanks could be used, or other storage tanks at the site if they are available. The plant Technical Specifications require certain minimum levels in each of these three tanks. Memorandum TM-2185 states that the HPCS tank will have a minimum level of about 32,000 gallons of fuel oil, while the two EDG tanks will each have minimum levels slightly above 54,000 gallons. The licensee will deploy two FLEX tanks mounted on trailers with a capacity of 500 gallons each to transfer fuel from the any of the above tanks to refuel portable equipment. Columbia also has 20 gpm transfer pumps and a fuel truck as well. Based on the design and location of these available fuel oil tanks and their protection from hazards, the staff finds the Seismic Category I tanks are robust and the fuel oil contents should be available to support the licensee's FLEX strategies during an ELAP event with LUHS.

As stated above, the EDG fuel oil tanks have more than 100,000 gallons. The licensee stated in its FIP that the 7 day consumption for all site diesel-powered FLEX equipment is 6,825 gallons. Furthermore, the licensee has calculated that operators will have to begin refueling the first components at 10 hours. Given the fuel demand for the Phase 2 FLEX components cited above and the large amount of available fuel, Columbia has a sufficient inventory of fuel for diesel-powered equipment required for the FLEX strategy, including the potential additional consumption by the Phase 3 equipment, until additional fuel arrives from off-site. In TM-2185 the licensee calculated that the total diesel fuel usage by Phase 3 equipment transported to the site would be about 8,856 gallons per day. Furthermore, the staff finds that the licensee has adequate plans to refuel the diesel-powered FLEX equipment to ensure uninterrupted operation to support the licensee's FLEX strategies.

The plant also has a need for a supply of gasoline, primarily for the small portable fuel transfer pumps that will be used to pump fuel and for some small portable generators. Building 82 and Building 600 are each stocked with a 30 gallon drum of gasoline and a 30 gallon drum of diesel fuel, to ensure ready availability. The 30 gallon drum of gasoline is reserved for fueling one fuel transfer pump for 72 hours.

Additionally, FIP Section 6.5 states that all FLEX equipment will be stored with sufficient fuel to operate for 10 hours, giving the licensee time to establish the refueling operation. The FIP did not explicitly describe that the maintenance and test plan involves checking and replacing fluids for the FLEX equipment. However, in Section 11.5 of its FIP the licensee committed to develop a maintenance and test program, which includes periodically checking and maintaining fluids in the FLEX equipment.

### 3.7.7 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should allow deploying the FLEX equipment following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.8 Considerations in Using Offsite Resources

### 3.8.1 Columbia SAFER Plan

The industry has collectively established the needed off-site capabilities to support FLEX Phase 3 equipment needs via the SAFER Team. The SAFER team is an alliance between the Pooled Equipment Inventory Corporation (PEICo) and AREVA, Inc., and provides FLEX Phase 3 management and deployment plans through contractual agreements with every commercial nuclear operating company in the United States.

There are two National SAFER Response Centers (NSRCs), located near Memphis, Tennessee and Phoenix, Arizona, established by SAFER to support nuclear power plants in the event of a BDBEE. Each NSRC holds five sets of equipment, four of which will be able to be fully deployed to the plant when requested. The fifth set allows removal of equipment from availability to conduct maintenance cycles. In addition, the plant's FLEX equipment hose and cable end fittings are standardized with the equipment supplied from an NSRC.

By letter dated September 26, 2014 [Reference 23], the NRC staff issued its assessment of the NSRCs established in response to Order EA-12-049. In its assessment, the staff concluded that SAFER has procured equipment, implemented appropriate processes to maintain the equipment, and developed plans to deliver the equipment needed to support site responses to BDBEEs, consistent with NEI 12-06 guidance; therefore, the staff concluded in its assessment that licensees can reference the SAFER program and implement their SAFER response plans to meet the Phase 3 requirements of Order EA-12-049.

The NRC staff noted that the licensee's SAFER Response Plan contains (1) SAFER control center procedures, (2) NSRC procedures, (3) logistics and transportation procedures, (4) staging area procedures, which include travel routes between staging areas to the site, (5) guidance for site interface procedure development, and (6) a listing of site-specific equipment (generic and non-generic) to be deployed for FLEX Phase 3.

### 3.8.2 Staging Areas

In general, up to four staging areas for NSRC supplied Phase 3 equipment are identified in the SAFER Plans for each reactor site. These are a Primary (Area C) and an Alternate (Area D), if available, which are offsite areas (within about 25 miles of the plant) utilized for receipt of ground transported or airlifted equipment from the NSRCs. From Staging Areas C and/or D, the SAFER team will transport the Phase 3 equipment to the on-site Staging Area B for interim staging prior to it being transported to the final location in the plant (Staging Area A) for use in Phase 3. For Columbia, Alternate Staging Area D is not used. Staging Area C is the Connell City Airport, which will only be used if the roads to the plant are impassable and helicopter delivery is required. Staging Area B is the parking lot at the plant training facility. Staging Area A is the specific location in the plant where the equipment will be operated.

Use of helicopters to transport equipment from Staging Area C to Staging Area B is recognized as a potential need within the Columbia SAFER Plan and is provided for.

### 3.8.3 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should allow utilization of offsite resources following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.9 Habitability and Operations

### 3.9.1 Equipment Operating Conditions

#### 3.9.1.1 Loss of Ventilation and Cooling

Following a BDBEE and subsequent ELAP with LUHS event, ventilation that provides cooling to occupied areas and areas containing required equipment will be lost. Per the guidance given in NEI 12-06, FLEX strategies must be capable of execution under the adverse conditions (unavailability of installed plant lighting, ventilation, etc.) expected following a BDBEE resulting in an ELAP.

The primary concern with regard to ventilation is the heat buildup which occurs with the loss of forced ventilation in areas that continue to have heat loads. The licensee performed a loss of ventilation analyses to quantify the maximum steady-state temperatures expected in specific areas of the plant related to FLEX implementation to ensure the environmental conditions remain acceptable for personnel habitability and within equipment qualification limits that would impact the FLEX strategy.

The key areas identified for all phases of execution of the licensee's FLEX strategy activities are located in the radwaste/control building, the reactor building, and the primary containment structure. These buildings contain the main control room, RCIC pump room, battery rooms, battery charger rooms, critical switchgear rooms, and the reactor pressure relief system. The licensee evaluated these areas to determine the temperature profiles following an ELAP with LUHS event. The results of the licensee's room heat-up evaluations have concluded that temperatures remain within acceptable limits based on conservative input heat load assumptions for all rooms/areas either with no mitigating actions or using passive or active means of ventilation, depending on the area.

The licensee performed calculations CVI 1201-00,1, "GOTHIC Analysis of CGS Radwaste Building Response to SBO," Revision 1, to evaluate the environmental response in the control room and vital island during an ELAP event, and CVI 1201-00,2, "Reactor Building GOTHIC Temperature Analysis during PSBO/ELAP," to address the environmental response in the reactor building due to heat loads during an ELAP. The calculations use the thermal-hydraulic computer code Generation of Thermal-Hydraulic Information for Containments (GOTHIC), version 8.0. A base case was run without mitigating actions. Additional cases were run to determine the effect of various mitigating actions. The licensee's mitigating actions to establish natural convection include opening selected doors, opening a roof hatch, and for the RCIC pump room, removing the ceiling plug. Plant Procedure Manual (PPM) 5.6.1, "SBO/ELAP" and PPM 5.6.2, "Station Blackout (SBO) and Extended Loss of AC Power ELAP Attachments," provide guidance for implementing the mitigating actions. The calculation assumes a 105.5 °F maximum outdoor air temperature with a 20 °F diurnal variation.

Main Control Room

The Equipment Room Temperature Table in Section 3.1.4.1 of the FIP shows that the main control room reaches 120 °F at 3.9 days and 129 °F at 7 days after initiation of an ELAP event. Appendix F of calculation CVI 1201-00,1 shows that shortly after electrical load reduction as a result of the ELAP and load shedding actions, the main control room temperature would be approximately 105 °F. The temperature gradually would increase to 110 °F at approximately 30 hours and 120 °F at approximately 96 hours after initiation of an ELAP event. Calculation CVI 1201-00,1 assumed mitigating actions that included opening doors to establish natural convection ventilation and removing tiles from the suspended ceiling to permit convective air flow to the concrete ceiling (i.e. heat sink). Procedure PPM 5.6.2 provides guidance for operator actions to mitigate the temperature increase in the Main Control Room.

Columbia will receive offsite resources and equipment from an NSRC starting about 24 hours after notifying the NSRC. The NRC staff finds that it is reasonable to expect that the licensee could utilize these resources prior to 72 hours into the event to reduce or maintain temperatures within the main control room to ensure that required electrical equipment survives indefinitely. Procedures ABN-FSG-NSRC-001 and ABN-FSG-NSRC-002, "NSRC Portable SW Pump Alignment to SW Loop A or SW Loop B," Major Revision 00, provide guidance for utilization of NSRC-supplied equipment. Restoration of main control room ventilation to address the room environment would be directed by the plant's emergency response organization (ERO) using plant procedures.

Based on expected room temperature remaining below 120 °F (the temperature limit, as identified in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, for electronic equipment to be able to function indefinitely), the NRC staff finds that the required electrical equipment in the main control room will continue to function with a loss of ventilation as a result of an ELAP event.

RCIC Room

The licensee's analysis (CVI 1201-00,2) showed that the RCIC room would exceed 120 °F without mitigating action. With mitigating actions the licensee's analysis showed that room temperature would reach 105 °F by 12 hours. Mitigating actions are assumed to be established at 12 hours. The room temperature drops to approximately 102 °F once mitigating actions are established. Procedure PPM 5.6.2 instructs operators to bypass RCIC trips for high area temperature and high RCIC exhaust pressure to ensure continued RCIC system operation.

The most limiting equipment relied on for the Columbia response is E-C2-1 (the 250 Vdc battery), which provides RCIC functional support. The licensee's analysis shows that the 250 Vdc battery room would reach its temperature limit in approximately 77 hours. The licensee expects that the Phase 2 FLEX pump will be connected and functional within 12 hours of the start of the ELAP. The FLEX pump could supply SFP and RPV makeup. Additionally, the NSRC equipment is expected to be operational within 72 hours after the onset of an ELAP event. Therefore, the licensee does not need to rely on RCIC in Phase 2 once the Phase 2 FLEX pump and FLEX DG are placed in service. In procedure PPM 5.6.2, Attachment 8.11 provides guidance to establish ventilation in the RCIC room. Guidelines ABN-FSG-003 and ABN-FSG-004 provide direction for connecting a FLEX DG to the electrical buses to supply required loads within the required timeframes.

Based on expected room temperature remaining at or below 120 °F (the temperature limit, as identified in NUMARC 87-00, for electronic equipment to be able to function indefinitely), the NRC staff finds that the credited equipment in the RCIC room will continue to function with a loss of ventilation as a result of an ELAP event.

Class 1E Battery Rooms

The 125 Vdc Class 1E station batteries (E-B1-1 and E-B1-2) supply power to the solenoid pilot valves (SPVs) that actuate the SRVs, and to other important loads. The 125 Vdc batteries can support SRV actuations and other required loads for at least 8 hours without recharging.

Licensee calculation CVI 1201-00,1 shows that the analyzed peak temperature of battery room 1 and battery room 2 during normal operation will remain lower than the Licensee Controlled Specifications (LCS) limit (110 °F). For E-B1-1, the maximum temperature would reach 97.2 °F at 72 hours after an ELAP event. For E-B1-2, the maximum temperature would reach 99.6 °F at 72 hours after an ELAP event.

Attachment D of CVI 1201-00,1 notes that when power to vital island equipment, cables, and lights is restored (i.e., during Phase 2) this will add heat loads to areas containing this equipment. Based on temperatures remaining below LCS limits, mitigating actions were not required to maintain room temperatures prior to 72 hours following initiation of an ELAP event. Columbia will receive offsite resources and equipment from an NSRC starting about 24 hours after notifying the NSRC. The NRC staff finds that it is reasonable to expect that the licensee could utilize these resources prior to 72 hours into the event to reduce or maintain temperatures within the battery rooms to ensure that required electrical equipment continues to function indefinitely.

Based on the above, the NRC staff finds that the 125 Vdc Class 1E battery room temperatures should remain below the LCS limit (110 °F) and the maximum temperature limit (125 °F) specified by the battery manufacturer (Enersys). Therefore, the 125 Vdc Class 1E batteries should not be adversely impacted by the loss of ventilation as a result of an ELAP event. While the battery vendor's analysis shows that the batteries are capable of performing their function up to 125 °F, periodic monitoring of electrolyte level may be necessary to protect the battery since the battery may gas more at higher temperatures. See the RCIC Pump discussion above for the NRC staff's review of the 250 Vdc battery room.

Critical Switchgear, Battery Charger, and Reactor Protection System Rooms

Licensee calculation CVI 1201-00,1 showed that the critical switchgear, battery charger, and reactor protection system (RPS) rooms will not reach the LCS limit for equipment in these rooms for at least 3 days.

CVI 1201-00,1 notes that when power to vital island equipment, cables, and lights is restored (i.e., during Phase 2) this will add heat loads to areas containing this equipment. CVI 1201-00,1 notes that with no mitigating actions, RPS rooms and battery charger rooms would experience temperatures of 137 °F and 157 °F, respectively, at 72 hours after initiation of an ELAP event. These temperatures are above LCS limits of 129 °F, 131 °F, and 140 °F for the RPS rooms and 122 °F, 129 °F, and 131 °F for the battery charger rooms. To maintain temperature within the LCS limits, the licensee determined that most of the doors in the vital island will need to be opened within two hours after initiation of an ELAP event. In addition to opening the doors, one 6000 cubic feet per minute portable floor fan must be deployed near battery charger room 1. The licensee plans to have this fan in operation 12 hours after initiation of an ELAP event. Procedure PPM 5.6.2 provides direction for deploying the fan. As a result, temperatures in the critical switchgear, battery charger, and RPS rooms should remain below the LCS limits for at least 72 hours.

Columbia will receive offsite resources and equipment from an NSRC starting about 24 hours after notifying the NSRC. The NRC staff finds that it is reasonable to expect that the licensee could utilize these resources prior to 72 hours into the event to reduce or maintain temperatures within these rooms to ensure that required electrical equipment continues to function indefinitely.

Reactor Pressure Relief System

A pressure relief system for the reactor system, consisting of SRVs 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 inside the primary containment structure. The SRVs can be opened by energizing an SPV and will operate automatically or can be manually opened and closed by operators to 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 their SPVs, which actuate the SRVs, are environmentally qualified for the full post-LOCA time frame. All other SRVs and their SPVs are qualified for 24-hours post-LOCA.

The SRV actuators were LOCA tested for 96 hours at over 308 °F. 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 ELAP profile for 72 hours.

Columbia will receive offsite resources and equipment from an NSRC starting about 24 hours after notifying the NSRC. The NRC staff finds that it is reasonable to expect that the licensee could utilize these resources prior to 72 hours into the event to reduce or maintain temperatures within primary containment to ensure that required electrical equipment continues to function indefinitely.

Primary Containment

The NRC staff reviewed the summary of the licensee's 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. Licensee calculations ME-02-12-18, "Containment Response during Extended Loss of AC Power (ELAP) – A Beyond Design Basis Assessment," Revision 1, and ME-02-14-13, "Containment Response During an Extended Loss of AC Power (ELAP) with Suppression Pool Makeup," Revision 0, modeled the transient temperature response in the containment following the first 72 hours of an ELAP event. The containment design pressure and temperature limits (45 psig for both the drywell and suppression chamber and 340 °F for the drywell and 275 °F for the suppression chamber) bound the expected containment pressure and temperature values calculated for an ELAP event (28.2 psig and 300 °F for the drywell and 21 psig and 240 °F for the suppression chamber at 72 hours after initiation of an ELAP event). The licensee plans to vent the suppression chamber at 6 hours into an ELAP event via the HCVS to maintain containment parameters below design limits. The affected instrumentation that the licensee has designated as necessary for event mitigation actions is listed in Section 3.2.3.1.2 of this SE. Of those instruments, only the drywell temperature and suppression pool temperature instruments are located inside the primary containment. For the others, the electronic transmitters are located outside the primary containment (in the reactor building) with only parts such as metal tubing and valves that are not sensitive to elevated temperature and pressure located within the primary containment. The drywell and suppression pool temperature instruments are qualified to meet or exceed LOCA conditions, and should remain functional during an ELAP.

Columbia will receive offsite resources and equipment from an NSRC starting about 24 hours after notifying the NSRC. The NRC staff finds that it is reasonable to expect that the licensee could utilize these resources prior to 72 hours into the event to reduce or maintain temperatures within primary containment to ensure that required electrical equipment continues to function indefinitely.

Based on temperatures remaining below the design limits of equipment, the ability to vent the suppression chamber via the HCVS, and the availability of offsite resources after 72 hours, the NRC staff finds that the credited electrical equipment in the primary containment should continue to function after the loss of ventilation as a result of an ELAP event.

Reactor Building

The refueling floor area in the reactor building is also the access point to the top of the SFP. The licensee's calculations indicate that the refueling floor area is expected to reach 183 °F without mitigating actions. Following a trip from full power due to an ELAP event, and with mitigating actions and makeup to the spent fuel pool, the room will reach 111 °F. The only electrical equipment needed to remain functional in this area is the SFP level instruments. The environmental qualifications of those instruments are discussed in Section 4.2.4 of this SE. Other equipment to be used in the refueling floor area, such as fire hoses and spray nozzles, are not affected by these elevated temperatures.

Most other areas in the reactor building below the elevation of the refueling floor were determined to remain at or below 102 °F. Procedure PPM 5.6.2 provides guidance to establish natural convection ventilation by opening selected doors and hatches. Based on expected room temperature remaining at or below 120 °F (the temperature limit, as identified in NUMARC 87-00, for electronic equipment to be able to function indefinitely), the NRC staff finds that the credited equipment in these areas of the reactor building will continue to function with a loss of ventilation as a result of an ELAP event.

3.9.1.2 Loss of Heating

The FIP indicated that during an ELAP, the primary concern for a loss of heating during cold weather is the effects of a loss of heat tracing used to ensure cold weather conditions do not result in freezing important piping and instrumentation systems. The licensee did not identify any important piping systems or instrumentation systems that could be adversely affected such that the implementation of mitigating actions would be impaired. The primary source of water for the Phase 1 FLEX strategy is the suppression pool. The suppression pool will remain at elevated temperatures due to the decay heat from the reactor core. Therefore, no specific action is required to compensate for a loss of heat trace during ELAP. During the Phase 2

FLEX strategy, suction hoses for the FLEX pump will be placed into a service water spray pond. Equipment to break through surface ice on the spray pond is available, if needed, to implement the Phase 2 strategy.

The Columbia Class 1E battery rooms and HCVS battery are located inside Seismic Category I structures and will not be exposed to extreme low temperatures. At the onset of the event, the Class 1E battery rooms would be at their normal operating temperature and the temperature of the electrolyte in the cells would build up due to the heat generated by the batteries discharging and during recharging. Temperatures in the battery rooms are not expected to be sensitive to extreme cold conditions due to their location, the concrete walls isolating the rooms from the outdoors, and lack of forced outdoor air ventilation during early phases of the ELAP event.

During an ELAP, the licensee's coping strategy calls for the connection of one of the FLEX DGs within 8 hours to support continued availability of dc power. Once equipment is re-powered by a FLEX DG, heat loads will be added in some of the vital island rooms. Room doors are opened in the vital island to dissipate heat generated in the highest heat load rooms to other areas within the vital island. Therefore, the battery rooms should be maintained above the minimum temperature limit. Based on the above, the NRC staff finds that the Class 1E station batteries should perform their required functions even with a loss of normal heating during an ELAP event.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern that is applicable to Phases 2 and 3 is the potential buildup of hydrogen in the battery rooms as a result of loss of ventilation during an ELAP event. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging. As shown in the licensee's hydrogen generation analysis (ME-02-13-14, "Hydrogen Generation during Extended Loss of AC Power"), the licensee concluded that the hydrogen accumulation in the battery rooms would not exceed the flammability limit. The analysis included both battery divisions and the batteries for the HCVS. Even with a delay in opening doors in the vital island, from 2 hours to 8 hours, the hydrogen concentration in the battery rooms remained below the flammability limit.

Based on its review of the licensee's battery room ventilation strategy, the NRC staff finds that hydrogen accumulation in the Class 1E battery rooms should not reach the flammability limit for hydrogen (4 percent) during an ELAP event.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

In Section 3.1.4.1 of the FIP, the "Equipment Room Temperature Table" shows the main control room reaches 120 °F at 3.9 days and after 7 days the temperature is 129 °F. The table indicates the acceptability of the 120 °F main control room temperature based on FSAR Chapter 8, Appendix 8A, "Station Blackout." The NRC staff questioned the ability of the operators to cope with an ELAP when control room temperature exceeds 110 °F for an extended time. The licensee responded that Appendix F of calculation CVI 1201-00,1 is the evaluation for the ELAP. Appendix F shows that shortly after electrical load reduction the control room temperature decreases to approximately 105 °F. The temperature gradually increases to 110 °F at approximately 30 hours and 120 °F at approximately 96 hours. NRC staff reviewed calculation CVI-1201-00,1, "GOTHIC Analysis of CGS Radwaste Building Response to SBO." The calculation assumed mitigating strategies of opening doors to establish natural convection ventilation and removing tiles from the suspended ceiling permitting convective air flow to the concrete ceiling. The concrete ceiling provides a heat sink. Procedure PPM 5.6.2 provides guidance for operator actions to mitigate the temperature increase in the main control room.

The licensee indicated that control room personnel can be relieved as needed as staff arrive from offsite. If conditions in the control room become problematic, ice vests are available.

The licensee also stated that they expect SAFER equipment to start arriving from an NSRC within 24 hours of notifying the NSRC. The ERO will direct the utilization of that equipment. Guidelines ABN-FSG-NSRC-001, "NSRC 4160V DG Crosstie via DG1, DG2 or SM-7 or SM-8" and ABN-FSG-NSRC-002, "NSRC Portable SW Pump Alignment to SW Loop A or SW Loop B," provide guidance for utilization of SAFER equipment. Restoration of main control room ventilation to address the control room environment will directed by the ERO using existing plant procedures.

The NRC staff finds that the main control room environment can be addressed such that operators will be able to continue actions needed to address the ELAP.

3.9.2.2 Spent Fuel Pool Area

The refueling floor area in the reactor building is also the access point to the top of the SFP. The licensee's calculations indicate that the refueling floor area is expected to reach 183 °F without mitigating actions. Following a trip from full power due to an ELAP event, and with mitigating actions and makeup to the spent fuel pool, the licensee calculated that the room will reach 111 °F. During refueling operations with a full core off-load and makeup to the spent fuel pool, the refueling floor temperature will exceed 140 °F. The NRC staff reviewed calculation ME-02-14-07, "Reactor Building Accessibility Following an Extended Loss of AC Power (ELAP) With Full Core Off-Load". The calculation determined mitigating actions to be taken before a full core is off-loaded, the timing of mitigating actions needed to establish natural convection, and that actions needed to support spent fuel pool cooling during an ELAP need to be completed within 2 hours. The calculation used wet bulb temperature (WBT) to establish allowable action times. It also assumed work expected by plant staff on the refueling floor can be done in regular work clothing. WBT below 82 °F was assumed to have unlimited work time. Rooms above 115 °F WBT were considered inaccessible. NUMARC 87-00, Revision 1, indicates that light work may be performed at temperatures of 110 °F for up to four hours. Based on the licensee's calculations and procedures, the NRC staff finds that plant personnel taking actions to respond to an ELAP event in accordance with plant procedures will not be impeded due to elevated temperatures on the refueling floor.

3.9.2.3 Other Plant Areas

Reactor Building

Most other areas in the reactor building below the elevation of the refueling floor were determined to remain at or below 102 °F. Procedure PPM 5.6.2 provides guidance to establish natural convection ventilation by opening selected doors and hatches. NUMARC 87-00, Revision 1, indicates that light work may be performed at temperatures of 110 °F for up to four hours. Based on the licensee's calculations and procedures, the NRC staff finds that plant personnel taking actions to respond to an ELAP event in these areas will not be impeded due to elevated temperatures in the reactor building.

RCIC Room

See Section 3.9.1.1 "Loss of Ventilation and Cooling" above for a discussion of the RCIC room temperature evaluation. Entry to the RCIC room is not required if alternate RPV makeup using a FLEX pump is in operation. During the audit the licensee indicated that prior to completion of mitigating actions to establish natural ventilation for the RCIC room, the licensee will apply the current plant program to limit stay times in areas with elevated temperatures. After mitigating actions are complete, with an expected room temperature at or below 102 °F, personnel could access the RCIC room for extended periods of time.

### 3.9.3 Conclusions

The NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain or restore equipment and personnel habitability conditions following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.10 Water Sources

### 3.10.1 RPV Make-Up

Phase 1

As described in the FIP, the non-seismically qualified CSTs are the normal suction source for the RCIC pump. The FIP states that the RCIC suction will automatically swap to the suppression pool in event the CSTs are unavailable. The licensee states that the suppression pool will be the suction source for RCIC for about the first 12 hours of the event. After the suppression pool vent is opened at about 6 hours into the event, the suppression pool level will drop slowly. The licensee will align the FLEX pump to provide makeup to the suppression pool via overflow from the SFP. The suppression pool is located in the primary containment and is a Seismic Category I structure, and it is protected from all applicable hazards.

Phase 2

During Phase 2, the licensee will transition from the RCIC pump to a portable FLEX pump to provide makeup water to the RPV. As discussed in Section 2.3.2 of this SE, the robust water sources for the FLEX pump are from either of the two safety-related service water spray ponds (the ultimate heat sink for the plant). In the FIP, Sections 3.1.4.7 and 4.0 describe the service water spray ponds and why they are robust for all applicable hazards. The service water spray ponds are man-made Seismic Category I ponds which are 250 feet by 250 feet and 15 feet deep. They contain a total of over 12.5 million gallons of water. In Section 4.4 of its FIP, the licensee states that the surface of the spray ponds could freeze during extreme cold and stated that actions have been developed to ensure the ability to break the surface ice to place the FLEX pump suction hose.

Phase 3

For Phase 3, RPV makeup strategy is the same as the Phase 2 strategy.

3.10.2 Suppression Pool Make-Up

The licensee plans to make up to the suppression pool using a FLEX pump with suction from one of the service water spray ponds by filling the SFP and aligning valves to allow overflow to the suppression pool.

3.10.3 Spent Fuel Pool Make-Up

The licensee plans to make up to the SFP using a FLEX pump with suction from one of the service water spray ponds.

3.10.4 Primary Containment Cooling

The licensee will use the wetwell vent to control pressure and temperature in the primary containment structure in Phase 1 and Phase 2. After NSRC-supplied equipment arrives at the site, the licensee will make decisions about further equipment lineups for containment cooling, such as placing an RHR loop in service for decay heat removal from the reactor core or for suppression pool cooling.

3.10.5 Conclusions

Based on the evaluation above, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain satisfactory water sources following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.11 Shutdown and Refueling Analyses

Order EA-12-049 requires that licensees must be capable of implementing the mitigation strategies in all modes. In general, the discussion above focuses on an ELAP occurring during power operations. This is appropriate, as plants typically operate at power for 90 percent or more of the year. If an ELAP occurs with the plant at power, the mitigation strategy initially focuses on the use of the steam-driven RCIC pump to provide the water initially needed for decay heat removal. If the plant has been shut down and all or most of the fuel has been removed from the RPV and placed in the SFP, there may be a shorter timeline to implement the makeup of water to the SFP. However, this is balanced by the fact that if immediate cooling is not required for the fuel in the reactor vessel, the operators can concentrate on providing makeup to the SFP. The licensee stated in the FIP that following a full core offload to the SFP, about 34.6 hours are available to implement makeup before boil-off results in the water level in the SFP dropping far enough to uncover fuel assemblies. The licensee has the ability to implement makeup to the SFP within that time by deploying and operating a FLEX pump.

When a plant is in a shutdown mode in which steam is not available to operate a steam-powered pump such as RCIC (which typically occurs when the RPV has been cooled below about 300 °F), another strategy must be used for decay heat removal. The NRC-endorsed strategy is described in NEI 12-06. Section 3.2.3 provides guidance to licensees for reducing shutdown risk by incorporating FLEX equipment in the shutdown risk process and procedures. Considerations in the shutdown risk assessment process include maintaining necessary FLEX equipment readily available and potentially pre-deploying or pre-staging equipment to support maintaining or restoring key safety functions in the event of a loss of shutdown cooling. On September 18, 2013, NEI submitted to the NRC a position paper entitled "Shutdown/Refueling Modes" [Reference 52], which described methods to ensure plant safety in those shutdown modes. By letter dated September 30, 2013 [Reference 53], the NRC staff endorsed this position paper as a means of meeting the requirements of the order. In Section 9 of its FIP, the licensee stated that it would follow this guidance.

Based on the licensee's plan to incorporate the use of FLEX equipment in the shutdown risk process and procedures, the NRC staff concludes that the licensee has developed guidance that if implemented appropriately should maintain or restore core cooling, SFP cooling, and containment following a BDBEE in shutdown and refueling modes consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

## 3.12 Procedures and Training

### 3.12.1 Procedures

Section 8 of the licensee's program document for FLEX, FLEX-01 [Reference 54], describes the procedures and guidelines the licensee developed to guide the FLEX strategies. The plant EOPs or abnormal condition procedures (ABNs) will be used for the initial operational response to a BDBEE. The FLEX strategies will be deployed in support of the EOPs/ABNs using separate FLEX support guidelines (FSGs). The FSGs provide guidance for the use of FLEX equipment to maintain or restore key safety functions. When FLEX equipment is needed to supplement the EOP/ABN strategies, the EOP/ABN will direct entry into and exit from the appropriate FSG.

### 3.12.2 Training

Section 9.2 of the licensee's program document for FLEX, FLEX-01 [Reference 54], describes the method for developing training plans for various plant personnel. The training plan development was done in accordance with licensee procedures using the systems approach to training.

### 3.12.3 Conclusions

Based on the description above, the NRC staff finds that the licensee has adequately addressed the procedures and training associated with FLEX. The procedures have been developed in accordance with NEI 12-06, Section 11.4, and a training program has been established in accordance with NEI 12-06, Section 11.6.

3.13 Maintenance and Testing of FLEX Equipment

In Section 11.5 of its FIP, the licensee stated that FLEX equipment was either initially tested or other reasonable means were used to verify that the performance conforms to the limiting FLEX requirements. The licensee stated that maintenance and testing guidance for FLEX equipment that directly performs a FLEX mitigation strategy for the core, containment, or the SFP was developed using the guidance provided in INPO [Institute of Nuclear Power Operations] AP 913, "Equipment Reliability Process."

The licensee stated in the FIP that 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." The NRC staff finds this to be acceptable, as the licensee's controls have been developed in accordance with NEI 12-06, Section 11.5.

The NRC staff finds that the licensee has adequately addressed equipment maintenance and testing activities associated with FLEX equipment because a maintenance and testing program has been established in accordance with NEI 12-06, Section 11.5.

3.14 Alternatives to NEI 12-06, Revision 2

The licensee and the NRC staff did not identify any alternatives to NEI 12-06, Revision 2.

3.15 Conclusions for Order EA-12-049

Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance to maintain or restore core cooling, SFP cooling, and containment following a BDBEE which, if implemented appropriately, should adequately address the requirements of Order EA-12-049.

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 [Reference 24], the licensee submitted its OIP for Columbia in response to Order EA-12-051. By letter dated June 20, 2013 [Reference 25], the NRC staff sent a request for additional information (RAI) to the licensee. The licensee provided a response by letter dated July 19, 2013 [Reference 26]. By letter dated November 7, 2013 [Reference 27], the NRC staff issued an ISE and RAI to the licensee.

By letters dated August 23, 2013 [Reference 29], February 27, 2014 [Reference 30], August 28, 2014 [Reference 31], and February 27, 2015 [Reference 32], the licensee submitted status reports for the Integrated Plan. The Integrated Plan describes the strategies and guidance to be implemented by the licensee for the installation of reliable SFP level instrumentation which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated August 12, 2015 [Reference 33] the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has installed a SFP level instrumentation system designed by Mohr. The NRC staff reviewed the vendor's SFP level instrumentation system design specifications, calculations and analyses, test plans, and test reports. The staff issued an audit report on August 27, 2014 [Reference 28].

The staff performed an onsite audit to review the implementation of SFP level instrumentation related to Order EA-12-051. The scope of the audit included verification of (a) site's seismic and environmental conditions enveloped by the equipment qualifications, (b) equipment installation met the requirements and vendor's recommendations, and (c) program features met the requirements. By letter dated June 16, 2015 [Reference 17], the NRC issued an audit report on the licensee's progress. Refer to Section 2.2 above for the regulatory background for this section.

4.1 Levels of Required Monitoring

For Level 1, in its OIP, the licensee stated, in part, that:

> The minimum level where the skimmer surge tank functions is at elevation 605 feet 5½ inches.
> Level 1 is considered the minimum level where the skimmer surge tank and the spent fuel pool water levels are still coupled.

In its letter dated July 19, 2013 [Reference 26], the licensee provided a sketch with an elevation view of the Columbia SFP depicting the Level 1, 2, and 3 datum points and instrument's measurement range. This sketch shows Level 1 at an elevation of 605 feet (ft.) 5 ½ inches (in.)

The NRC staff notes that the elevation identified as Level 1 is adequate for normal SFP cooling system operation and it is also adequate to ensure the required fuel pool cooling pump net positive suction head (NPSH) as the skimmer surge tanks supply the SFP cooling pumps. This level represents the higher of the two points described in NEI 12-02 for Level 1.

For Level 2, in its OIP, the licensee stated, in part, that:

> At Columbia, Level 2 is defined at elevation 593 feet 2 inches, which is 10 feet above the top of the fuel racks (583 feet 2 inches). In addition to the spent fuel racks, the spent fuel pool is also used to store other materials that could affect the radiation dose in this area. Accordingly, station procedures will be used to maintain a water level to adequately protect personnel from significant dose consequence. The detailed design will determine the basis for the station procedures.

Additionally, in its letter dated July 19, 2013 [Reference 26], the licensee stated, in part, that:

> The radiation dose rates from irradiated spent fuel and other materials stored in the SFP area will be assessed in a calculation for habitability and equipment qualification purposes. Necessary compensatory measures and actions will be incorporated into station procedures to ensure that there is adequate radiation shielding to maintain personnel radiological dose levels within acceptable limits while performing local operations in the vicinity of the pool when SFP water level is below the normal water level band. Level 2 will remain at 10 feet above the SFP rack because this is the non-changeable value based on SFP rack locations.

Conversely, material and the elevations at which it is stored in the pool will vary based on outage work and pool clean-up. Necessary limitation on SFP makeup actions due to temporarily stored materials will be addressed through administrative controls incorporated into the EA-12-051 response related procedures. Any dose from uncovering these materials will be mitigated once water level is restored through diverse and flexible coping strategies (FLEX procedures) as needed. Additionally, diverse means exist for adding water to the SFP including the use of spray nozzles that do not require access to the immediate vicinity of the SFP.

In its letter dated August 17, 2017 [Reference 18], the licensee also stated that:

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 to 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 instruments and selected SFP operating deck areas....

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 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 ft. and at Level 2 are similar. The maximum dose rates of about 6.5 E+04 mR/hr and 1.5E+05 mR/hr occur near the southeast and southwest comers of the SFP.

The NRC staff noted that at Level 2, the maximum dose rates of about 6.5 E+04 mR/hr and 1.5E+05 mR/hr occur near the southeast and southwest corners of the SFP. The staff requested the licensee to address compensatory measures and actions to be taken, to be incorporated into the station procedures, as a result of projected dose rate impact and material stored in SFP. In response to the staff request, the licensee stated that plant procedure PPM 5.3.1 directs that actions be taken upon a SFP level of less than 22 ft. 4 in. above the top of the fuel racks, to return water level above 22 ft. 4 in. This procedure also contains actions if level cannot be maintained. Supporting procedures are ABN-FPC-LOSS, which directs actions upon a loss of fuel pool cooling and PPM 4.602.A5, which directs actions if a refueling floor area high radiation alarm is received which includes:

- SUSPEND all work activities on the refueling floor.
- EVACUATE the floor.
- NOTIFY Health Physics.
- REFER to ABN-RAD-HIGH, Abnormally High Area Radiation Levels.

Procedure PPM 5.3.1 also contains actions to take upon exceeding any reactor building area radiation alarm level. Procedure ABN-RAD-High also directs the notification of Health Physics.

The NRC staff finds the licensee's response acceptable because the licensee addressed compensatory measures and actions to be taken incorporated into the station procedures as a result of the projected dose rate impact and other material stored in SFP. Level 2 is 10 feet above the spent fuel racks. To address adequacy of radiation shielding to maintain personnel radiological dose levels within the acceptable limits while performing local operation in the vicinity of the SFP, the licensee has procedures directing actions to maintain SFP water level 22 ft. 4 in. to 22 ft. 10-1/8 in. This water level should provide radiation shielding to personnel within acceptable limits while performing local operation in the vicinity of the SFP. In its letter dated August 17, 2017 [Reference 18], the licensee stated that they could add water to the SFP using an oscillating spray nozzle. Also, if the refueling floor is inaccessible, an alternate water supply path to the SFP is available by connecting the supply hose to valve RHR-V-63B. Based on this information, the staff finds that SFP Level 2 is acceptable.

For Level 3, in its OIP, the licensee stated, in part, that:

> At Columbia, Level 3 corresponds to an elevation of 583 feet 2 inches which is located at the top of the fuel rack that is 15 feet ½ inch from the bottom of the spent fuel pool.

In its letter dated July 19, 2013 [Reference 26], the licensee provided a sketch with an elevation view of the Columbia SFP showing the Level 1, 2, and 3 datum points and the level instrument's measurement range. This sketch shows Level 3 at an elevation of 583 ft. 2 in. The NRC staff notes that the elevation for Level 3 is the highest point of any spent fuel storage rack seated in the SFP, which is consistent with NEI 12-02.

Based on the evaluation above, the NRC staff finds that the licensee's proposed Levels 1, 2 and 3 appear to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

### 4.2 Evaluation of Design Features

Order EA-12-051 required that the SFP level instrumentation (SFPLI) shall include specific design features, including specifications on the instruments, arrangement, mounting, qualification, independence, power supplies, accuracy, testing, and display. Refer to Section 2.2 above for the requirements of the order in regards to the design features. Below is the staff's assessment of the design features of the SFP level instrumentation.

#### 4.2.1 Design Features: Instruments

In its OIP, the licensee stated that the primary instrument channel level sensing component will consist of a new permanently mounted guided-wave radar device and that the backup instrument channel will consist of an independent and redundant channel of the same technology. The licensee also stated that the instrument channels will provide a continual display of water level indication over the complete range from the top of the fuel rack at 583 ft. 2 in. to the maximum nominal water level at 606 ft. 1 in.

The NRC staff notes that the range specified for the licensee's instrumentation will cover Levels 1, 2, and 3 as described in Section 3.2 above. The staff finds that the licensee's design, with respect to the number of instrument channels and measurement range for its SFPLI, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.2 Design Features: Arrangement

In its OIP, the licensee stated, in part, that:

> The primary channel level transmitter probe will be located near or in the north-west corner of the pool. The mounting of the level transmitter probe does not require drilling or tapping into the spent fuel pool liner. The backup channel level transmitter probe will be located diagonally across the spent fuel pool from the primary probe in or near the south-east corner of the pool which is in the cask loading area. This location in or near the corner provides protection from the spent fuel transfer cask due to the geometry of the corner and the circular transfer cask. The mounting plate for the level transmitter probe will be anchored to the concrete around the edge of the pool and will not require drilling or tapping into the spent fuel pool liner. The locations near the north-west and south-east corners afford the greatest separation between probes to maximize the protection, through separation, against missile damage to both channels. The minimum distance between the primary and backup channels is the shortest length of the spent fuel pool side (34 feet). Mounting in or near the corner of the spent fuel pool is preferred over other pool locations. The primary and backup channel electronics cannot be located on the refuel floor because they are sensitive to radiation. The electronics will be located in an area where they are capable of performing their function in the selected environment (humidity, temperature & radiation). The primary and backup instrument channel cables will be routed separately in rigid conduits and utilize the walls and building structure to provide reasonable protection from missiles that may result from damage to the structure over the spent fuel pool and refuel floor. The maximum distance from the probe to the electronics is limited to 1000 feet of cabling. There are no cables routed external to the Reactor, Radwaste, and Turbine Buildings.

During the onsite audit, the licensee provided drawings related to the SFPLI. The NRC staff reviewed Drawing E701, "Reactor Building EL 606-10 ½ Instrumentation and Control Conduit and Tray Plan," Rev. 0, and Drawing FSKE-11797-001 "Electrical Cable/Raceway Routing Functional Diagram Reactor Building – Fuel Pool Level Instrumentation," Rev. 0. The staff found the locations for the SFPLI acceptable and verified them during the walkdown. The NRC staff noted that there is sufficient channel separation within the SFP area between the primary and back-up level instruments, sensor electronics, and routing cables to provide reasonable protection against loss of indication of SFP level due to missiles that may result from damage to the structure over the SFP.

Based on the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee's proposed arrangement for the SFPLI appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.3 Design Features: Mounting

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> 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 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.
>
> 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.

During the onsite audit, the NRC staff reviewed the mounting specifications and seismic analyses for the SFPLI, including the methodology and design criteria used to estimate the total loading on the mounting devices. The staff also reviewed the design inputs and the methodology used to qualify the structural integrity of the affected structures for each of the SFPLI mounting attachments. Based on the review, the staff found the criteria established by the licensee adequately account for the appropriate structural loading conditions, including seismic and hydrodynamic loads.

Based on the evaluation above, the NRC staff finds that the licensee's proposed mounting design appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

Appendix A-1 of the guidance in NEI 12 02 describes a quality assurance process for non-safety systems and equipment that are not already covered by existing quality assurance requirements. In JLD-ISG-2012-03, the NRC staff found the use of this quality assurance process to be an acceptable means of meeting the augmented quality requirements of Order EA-12-051.

In its OIP, the licensee stated that the primary and backup SFPLI channels will be classified as Augmented Quality (AQ), similar to the quality assurance process that applies to fire protection systems.

Based on the evaluation above, the NRC staff finds that, if implemented appropriately, this approach appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.4.2 Instrument Channel Reliability

Section 3.4 of NEI 12-02 states, in part:

> The instrument channel reliability shall be demonstrated via an appropriate combination of design, analyses, operating experience, and/or testing of channel components for the following sets of parameters, as described in the paragraphs below:
>
> - conditions in the area of instrument channel component use for all instrument components,
> - effects of shock and vibration on instrument channel components used during any applicable event for only installed components, and
> - seismic effects on instrument channel components used during and following a potential seismic event for only installed components.

Equipment reliability performance testing was performed to (1) demonstrate that the SFP instrumentation will not experience failures during beyond-design-basis (BDB) conditions of temperature, humidity, emissions, surge, and radiation, and (2) to verify those tests envelope the plant-specific requirements.

The NRC staff reviewed the MOHR SFPLI's qualification and testing during the vendor audit for temperature, humidity, radiation, shock and vibration, and seismic [Reference 28]. The staff further reviewed the anticipated Columbia's environmental condition during the onsite audit [Reference 17]. Below is the staff's assessment of the equipment reliability of the Columbia SFPLI.

4.2.4.2.1 Radiation, Temperature, and Humidity

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

Temperature:

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 1E wire and cable meet the requirements of IEEE 383-1974, *IEEE Standard for Type Test of Class 1E 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.

Humidity:

Signal processor and Extended Batteries: Based on vendor testing/analysis the 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.

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.

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 the main control room is 120° Fahrenheit (F). The processor continues to functions successfully in conditions up to 131°F at 100% humidity.

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 (CB) 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 (L1 - Normal), for 8 years. Case 2c included integrated dose calculations for the NW and SE Level Indicators from the fuel and CB rollers, including sky shine 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) |

During the onsite audit, the NRC staff reviewed calculation NAI-1751-001, "GOTHIC Analysis of CGS Radwaste Building Response to SBO," Rev. 1, and CVI 1201-003 "EN-CGS Spent Fuel Pool Dose for New Level Instrumentation," Rev. 0. The staff verified that the environmental conditions for both the normal condition and post-BDB event for the locations where the SFPIS located are bounded by the vendor equipment qualification.

#### 4.2.4.2.2 Seismic

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> 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.

The NRC staff noted that the licensee adequately addressed the SFPLI seismic design. The staff audited the MOHR SFP instrumentation design verification analyses and performance test results [Reference 28] and found the seismic testing of SFPLI acceptable.

4.2.4.2.3 Shock and Vibration

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> 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*.
>
> 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.

The NRC staff noted that the licensee adequately addressed the SFPLI shock and vibration design. The staff audited the MOHR SFP instrumentation design verification analyses and performance test results [Reference 28] and found the shock and vibration testing of SFPLI acceptable.

4.2.4.2.4 Electromagnetic Compatibility

In its letter dated August 12, 2015 [Reference 33], the licensee stated that:

> Columbia has reassessed the RF [Radio Frequency] susceptibility of the new SFP instrumentation required by NRC Order EA-12-051. Columbia reviewed the requirements in the Electric Power Research Industry (EPRI) Test Report (TR) 102323, "Guidelines for Electromagnetic Interference Testing of Power Plants," previously endorsed by the NRC. The new instrumentation is classified as non-safety related and Table 1 of TR-102323 indicates that testing of this equipment for RF susceptibility is optional and therefore was not specified to the vendor.

> The vendor performance testing was witnessed and documentation reviewed and accepted by Columbia. A demonstration of RF interference susceptibility was performed on May 21, 2015. The vendor reports and the demonstration results have been posted on the Columbia eportal as outlined in the Columbia Audit Plan dated January 16, 2015. Since no EMI, RFI, or RF susceptibility were identified, no mitigation is required.

The NRC staff reviewed Work Order 02046102 Task 16 Revision 1. In this order, the licensee's personnel on Reactor Building 606 ft. elevation key a plant radio in close proximity to the level probes SFP-LE-21A and SFP-LE-21B. The licensee confirmed with an individual in the Main Control Room who was monitoring the level indicators SFP-LIT-21A and SFP-LIT-21B that there was no radio interference indicated when the radio was keyed on the Reactor Building 606 ft. elevation. Based on this information, the staff finds that the SFPLI located at Reactor Building 606 ft. elevation is not susceptible to RF.

Based on the evaluations above, the NRC staff finds that the licensee's proposed instrument qualification process appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.5 <u>Design Features: Independence</u>

In its letter dated July 19, 2013 [Reference 26] the licensee stated, in part, that:

> The Columbia SFP level instrumentation will be designed to meet the requirements set forth in EA-12-051, NEI 12-02, and JLD-ISG-2012-03. The system will provide two completely independent channels of level instrumentation. The sensors will be located at opposite corners of the SFP and the electronics/processor will be located in the main control room. The location near the opposite corners of the SFP provides the maximum separation between sensors to maximize the protection against missile damage to both channels.
>
> The primary and the backup channels will use the same technology but will be redundant and independent. Both channels will be physically separated to the extent practical from each other by having the probes located at opposite corners of the SFP, separate cable routes, and separate electronics/display mounting locations. The power cables and signal cables for both channel will be routed separately and will maintain the Columbia station separation requirements. Both channels will be powered from independent and separate divisions.

During the onsite audit, the NRC staff performed a walkdown of the SFPLI channels. The staff noted that the primary instrument channel is independent of the backup instrument channel and consistent with recommendations for channel independence in NEI 12-02.

The NRC staff finds that the licensee's proposed design, with respect to instrument channel independence, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.6 Design Features: Power Supplies

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> Each SFP instrument channel is normally powered from a 120/240V 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 480V ac bus does not result in loss of normal 120V ac power for both instrument channels.
>
> On loss of normal 120V 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 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% 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.

During the onsite audit, the NRC staff reviewed drawings EWD-46E-242, "Electrical Wiring Diagram AC Electrical Distribution System – Power Panel E-PP-7AZ," Rev. 25; EWD-46E-254, "Electrical Wiring Diagram AC Electrical Distribution System – Power Panel E-PP-8AZ," Rev. 31; E503-8, "Auxiliary One Line Diagram," Rev. 97; and E503-7, "Auxiliary One Line Diagram," Rev. 88. The staff found that the power supply sources for each channel is independent and the loss of power supply from one channel will not result in a loss of ac power for both channels. The staff reviewed the battery duty cycle during the vendor audit and found it to be acceptable.

Based on the evaluation above, the NRC staff finds that the licensee's proposed power supply design appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.7 Design Features: Accuracy

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> 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.
>
> In general relative to normal operating conditions, an applicable calibration procedure tolerances or acceptance criterion have been established based on manufacturer 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.

The NRC staff finds that the instrument accuracy of ± 3 inches is within the accuracy of ± 1 foot as recommended in NEI 12-02. The licensee will use the manufacturer's design accuracy as acceptance criteria in procedures which were developed to take corrective action if the accuracy exceeds ± 3 inches.

Based on the evaluation above, the NRC staff finds that the licensee's proposed instrument accuracy appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.8 Design Features: Testing

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> 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

> 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.
>
> 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.

The NRC staff noted that the licensee adequately addresses the equipment testing including periodic testing, calibration, and preventive maintenance. These tasks appear to be consistent with the vendor recommendation.

Based on the evaluation above, the NRC staff finds that the licensee's proposed SFP instrumentation design that allows for testing appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.9 <u>Design Features: Display</u>

In its OIP, the licensee stated, in part, that:

> Both the primary and backup channel will be continuously displayed in the main control room. Both channels will have an alternate location for viewing level indication at the electronic control cabinet for the guided wave radar. The display supplied in the vendor electronics is a second display location that is designed

> not to impact the main control room loop indication. The main control room indicator is instrument loop powered.

The staff notes that the NEI 12-02 guidance for "Display" specifically mentions the control room as an acceptable location for SFP instrumentation displays as it is occupied or promptly accessible, outside the area surrounding the SFP, inside a structure providing protection against adverse weather and outside of any very high radiation areas or locked high radiation areas during normal operation.

Based on the evaluation above, the NRC staff finds that the licensee's proposed location and design of the SFP instrumentation displays appear to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

### 4.3 Evaluation of Programmatic Controls

Order EA-12-051 specified that the spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation programmatic controls, including training, procedures, and testing and calibration. Below is the NRC staff's assessment of the programmatic controls for the spent fuel pool instrumentation.

#### 4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated, in part, that:

> The Systematic Approach to Training (SAT) will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training. Training will be completed prior to placing the instrumentation in service. Training will be performed in accordance with station procedures, processes, and vendor recommendations.

The NRC staff finds that the licensee's plan to train personnel in the operation, maintenance, calibration, and surveillance of the SFPLI, including the approach to identify the population to be trained, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

#### 4.3.2 Programmatic Controls: Procedures

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

> 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 are 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 – CFT
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 - CC
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.

The NRC staff finds the licensee provided an adequate set of procedures addressing operation, calibration, test, maintenance, and inspection for the SFPLI. These procedures are consistent with the recommendation from the vendor.

Based on the evaluation above, the NRC staff finds that the licensee's procedure development appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.3 <u>Programmatic Controls: Testing and Calibration</u>

In its letter dated August 17, 2017 [Reference 18], the licensee stated that:

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. 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. 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.

The NRC staff finds that the licensee adequately addresses testing and calibration of the SFPLI. The staff also finds that the allowed outage time and compensatory actions for non-functional SFPLI channels are consistent with those recommended by NEI 12-02.

Based on the evaluation above, the NRC staff finds that the licensee's proposed testing and calibration plan appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

### 4.4 Conclusions for Order EA-12-051

In its letter dated August 12, 2015 [Reference 33], the licensee stated that they have met the requirements of Order EA-12-051 by following the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. In the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee has conformed to the guidance in NEI 12-02, as endorsed by JLD-ISG-2012-03. In addition, the NRC staff concludes that if the SFP level instrumentation is installed at Columbia according to the licensee's proposed design, it should adequately address the requirements of Order EA-12-051.

## 5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on Orders EA-12-049 and EA-12-051. The staff conducted an onsite audit in February 2015 [Reference 17]. The licensee reached its final compliance date on June 19, 2017, and has declared that the Columbia reactor is in compliance with the orders. The purpose of this safety evaluation is to document the strategies and implementation features that the licensee has committed to. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs that if implemented appropriately should adequately address the requirements of Orders EA-12-049 and EA-12-051. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders.

## 6.0 REFERENCES


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2. SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," February 17, 2012 (ADAMS Accession No. ML12039A103)

3. SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 - Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," March 9, 2012 (ADAMS Accession No. ML120690347)

4. Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," March 12, 2012 (ADAMS Accession No. ML12054A736)

5. Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (ADAMS Accession No. ML12054A679)

6. Nuclear Energy Institute document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 2, December 31, 2015 (ADAMS Accession No. ML16005A625)

7. 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," Revision 1, January 22, 2012 (ADAMS Accession No. ML15357A163)

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10. Letter, Energy Northwest to NRC, "Columbia Generating Station, Docket No. 50-397, Energy Northwest's Response to NRC Order EA-12-049 – Overall Integrated Plan for Mitigating Strategies," February 28, 2013 (ADAMS Accession No. ML130730477, non-public)

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12. Letter, Energy Northwest to NRC, "Columbia Generating Station, Docket No. 50-397, 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," February 27, 2014 (ADAMS Accession No. ML14073A122, non-public)

13. Letter, Energy Northwest to NRC, "Columbia Generating Station, Docket No. 50-397, 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," August 28, 2014 (ADAMS Accession No. ML14254A403)

14. Letter, Energy Northwest to NRC, "Columbia Generating Station, Docket No. 50-397, 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," March 2, 2015 (ADAMS Accession No. ML15083A086)

15. Letter from Jack R. Davis (NRC) to All Operating Reactor Licensees and Holders of Construction Permits, "Nuclear Regulatory Commission Audits of Licensee Responses to Mitigation Strategies Order EA-12-049," August 28, 2013 (ADAMS Accession No. ML13234A503)

16. Letter from Jeremy S. Bowen (NRC) to Mark Reddemann (Chief Executive Officer – Energy Northwest), "Columbia Generating Station – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Order EA-12-049 (Mitigation Strategies)," January 29, 2014 (ADAMS Accession No. ML13337A266)

17. Letter from Stephen Monarque (NRC) to Mark Reddemann (Chief Executive Officer – Energy Northwest), "Columbia Generating Station – 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," June 16, 2015 (ADAMS Accession No. ML15139A462)

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COLUMBIA GENERATING STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 DATED February 22, 2018

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