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

August 2, 2017

Mr. Adam C. Heflin
President, Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 – 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. MF0788 AND MF0781)

Dear Mr. Heflin:

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 to Modify 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 27, 2013 (ADAMS Accession No. ML13070A026), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted its OIP for Wolf Creek Nuclear Generating Station, Unit 1 (WCGS) in response to Order EA-12-049. At six month intervals following the submittal of its 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 February 6, 2014 (ADAMS Accession No. ML14002A190), and July 6, 2016 (ADAMS Accession No. ML16168A254), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated January 19, 2017 (ADAMS Accession No. ML17026A194), WCNOC submitted a compliance letter and Final Integrated Plan 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. ML13071A419), WCNOC submitted its OIP for WCGS 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 October 29, 2013 (ADAMS Accession No. ML13295A681), and July 6, 2016 (ADAMS Accession No. ML16168A254), the NRC staff issued an ISE and 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 June 30, 2015 (ADAMS Accession No. ML15190A337), WCNOC 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. By letter dated June 12, 2017 (ADAMS Accession No. ML17171A233), WCNOC provided additional information regarding the reliable spent fuel pool instrumentation at WCGS.

The enclosed safety evaluation provides the results of the NRC staff's review of WCNOC's strategies for WCGS. The intent of the safety evaluation is to inform WCNOC 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, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/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 Peter Bamford, Orders Management Branch, WCGS Project Manager, at 301-415-2833 or at Peter.Bamford@nrc.gov.

Sincerely,

Tony Brown, Acting Chief
Orders Management Branch
Japan Lessons-Learned Division
Office of Nuclear Reactor Regulation

Docket No.: 50-482

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

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 anticipated 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 applies 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 power reactor licensees and all holders of 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 the Mitigation Strategies order. The NRC staff reviewed NEI 12-06, Revision 2, and on January 22, 2016, issued Japan Lessons-Learned Directorate (JLD) 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 makeup water addition should no longer be deferred.
>
> 1. The spent fuel pool level instrumentation shall include the following design features:
>
> 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 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 <u>TECHNICAL EVALUATION OF ORDER EA-12-049</u>

By letter dated February 28, 2013 [Reference 10], Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted an Overall Integrated Plan (OIP) for Wolf Creek Generating Station, Unit 1 (WCGS, Wolf Creek) in response to Order EA-12-049. By letters dated August 28, 2013 [Reference 11], February 26, 2014 [Reference 12], August 28, 2014 [Reference 13], February 24, 2015 [Reference 14], August 25, 2015 [Reference 15], February 17, 2016 [Reference 16], and August 18, 2016 [Reference 17], the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 [Reference 18], 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 February 6, 2014 [Reference 19], and July 6, 2016 [Reference 20], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated January 19, 2017 [Reference 21], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved, and submitted a Final Integrated Plan (FIP).

3.1 <u>Overall Mitigation Strategy</u>

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. Thus, the ELAP with loss of normal access to the UHS 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.

Wolf Creek is a four loop Westinghouse pressurized-water reactor (PWR) with a dry ambient pressure 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 reactor coolant pumps (RCPs) coast down and flow in the reactor coolant system (RCS) transitions to natural circulation. Decay heat is removed by releasing steam from the steam generators (SGs) through the SG atmospheric relief valves (ARVs). Makeup to the SGs is initially provided by the turbine-driven auxiliary feedwater (TDAFW) pump taking suction from the condensate storage tank (CST). A transition to emergency operating procedure (EOP) EMG C-0, "Loss of All AC Power," will be made by the licensee upon the diagnosis of the total loss of ac power. This procedure directs isolation of RCS letdown pathways, verification of containment isolation, reduction of dc loads on the station Class 1E batteries, and establishes electrical equipment alignment in preparation for eventual power restoration. Within the first 8 hours after the event initiation, the operators will initiate a cooldown of the RCS at a maximum rate of 100 degrees Fahrenheit (°F) per hour until a SG pressure of 310 pounds per square inch gage (psig) is reached. This initial cooldown of the RCS minimizes inventory loss through the RCP seals and allows for eventual SG makeup from a portable pump, in the event that the TDAFW pump becomes unavailable. There are 4 safety injection (SI) accumulators that will begin to passively inject into the RCS when RCS pressure drops below approximately 600 psig. The accumulators provide borated RCS makeup to help ensure natural circulation is maintained within the RCS and also provide a negative reactivity insertion to the RCS. The target SG pressure for this cooldown is high enough to prevent nitrogen gas from the SI accumulators from entering the RCS. The CST's minimum usable volume should provide a suction source to the TDAFW pump for approximately 31 hours, assuming the operational profile described above.

The Phase 2 FLEX strategy for reactor core cooling and heat removal is to continue to utilize the TDAFW pump and ARVs. Before the complete depletion of the usable CST inventory, makeup to the CST from the UHS will be established. If necessary, SG water injection capability may eventually be provided using a portable FLEX core cooling pump. Borated RCS makeup, preferably supplied from a boric acid tank (BAT), will be initiated within 13 hours of the ELAP using an RCS FLEX makeup pump. The Phase 2 FLEX strategy also includes re-powering of certain 480 Volts-ac (Vac) buses within 8 hours using a portable 500 kilowatt (kW) FLEX diesel generator (DG).

In Phase 3, core cooling is continued to be maintained through natural circulation heat removal from the RCS via the SGs. RCS inventory and sub-criticality are maintained via the Phase 2 strategy. However, to supplement the borated water inventory at the site, the Phase 3 deployment from the National SAFER [Strategic Alliance for FLEX Emergency Response]

Response Center (NSRC) includes a mobile water filtration system and a skid capable of generating borated coolant to extend coping times with respect to RCS inventory management. Makeup to the SGs is continued to be maintained via either the TDAFW pump or the FLEX core cooling pump; however, if the UHS is providing the water source for SG makeup, use of this non-purified water source is not desired for long-term service. Therefore, deployment of a skid capable of generating clean water for use as makeup to the SGs will be used. Ultimately, once sufficient power sources and supplemental equipment is provided, RCS cooling can be transferred to the residual heat removal (RHR) system. To support this strategy, the Phase 3 deployment includes a large pump capable of providing water to the essential service water (ESW) system, to cool the component cooling water (CCW) system, and subsequently the RHR system.

Regarding SFP cooling, the licensee's calculations estimate that with no operator action following a loss of SFP cooling, assuming a normal SFP heat load and an initial SFP temperature of 140°F, the SFP will start to boil in approximately 6 hours. Once boiling starts, the SFP inventory will reach the top of the fuel racks in approximately 35 hours from initiation of the event if no makeup is provided. Under the maximum design SFP heat load (full core off-load) with initial SFP temperature of 140°F, the SFP will start to boil in approximately 2 hours and will reach the top of the fuel racks in approximately 15 hours from initiation of the event, if makeup is not provided. The Phase 1 coping strategy is to monitor SFP level using instrumentation installed as required by NRC Order EA-12-051, establish a vent path for the Fuel Building, and to pre-stage hoses and equipment that will eventually be used to provide SFP makeup.

The Phase 2 strategy is to initiate SFP makeup using a FLEX SFP pump. Makeup to the SFP would be from the refueling water storage tank (RWST) (if available), or the CST. The discharge of the FLEX SFP pump would be connected by a hose to a makeup connection inside of the Fuel Building, or to a backup connection, also in the Fuel Building. The licensee's FIP indicates that the capability to provide SFP spray is also included in the overall plan. For this option, the licensee utilizes hoses run from the FLEX SFP pump to portable spray nozzles placed on the operating floor of the Fuel Building.

The Phase 3 coping capabilities for SFP cooling continue to utilize the Phase 2 strategies. Additional low pressure/high flow pumps will be available from NSRC as a backup to the onsite FLEX SFP pump(s).

Regarding containment integrity, the licensee's evaluations have concluded that containment temperature and pressure will remain below the containment design limits and that key parameter instruments subject to the containment environment will remain functional for a minimum of 7 days. Therefore, actions to reduce containment temperature and pressure and to ensure continued functionality of the key parameters will not be required immediately. The licensee's procedures will ensure that actions to isolate containment without available ac power will be performed.

The Phase 2 coping strategy is to continue monitoring containment temperature and pressure using installed instrumentation. Phase 2 activities to repower key instrumentation are required to continue containment monitoring.

The Phase 3 strategy is to continue the monitoring of containment temperature and pressure using installed instrumentation. The licensee's plan states that reduction of the RCS heat load indirectly reduces containment temperature and pressure. Restoration of the CCW, ESW and RHR systems is performed in Phase 3 to accomplish this strategy.

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 equipment that is presumed lost due to the postulated event) that is robust in accordance with the guidance in NEI 12-06. In Phase 2, robust installed equipment is supplemented by onsite FLEX equipment, which is used to cool the core either directly (e.g., pumps and hoses) or indirectly (e.g., FLEX electrical generators and cables repowering robust installed equipment). The equipment available onsite for Phases 1 and 2 is further supplemented in Phase 3 by equipment transported from the NSRCs.

To adequately cool the reactor core under ELAP conditions, two fundamental physical requirements exist: (1) a heat sink is necessary to accept the heat transferred from the reactor core to coolant in the RCS, and (2) sufficient RCS inventory is necessary to transport heat from the reactor core to the heat sink via natural circulation. Furthermore, inasmuch as heat removal requirements for the ELAP event consider only residual heat, the RCS inventory should be replenished with borated coolant in order to maintain the reactor in a subcritical condition as the RCS is cooled and depressurized.

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 unit would have been operating at full power prior to the event. Therefore, the SGs may be credited as the heat sink for core cooling during the event. Maintenance of sufficient RCS inventory, despite 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 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 safety evaluation.

3.2.1 Core Cooling Strategy and RCS Makeup

3.2.1.1 Core Cooling Strategy

3.2.1.1.1 Phase 1

The licensee's FIP states that immediately following the loss of power, the reactor will trip and the plant will initially stabilize at no-load RCS temperature and pressure conditions, with reactor decay heat removal via steam release to the atmosphere through the SG ARVs. Natural circulation of the RCS will develop to provide core cooling, and the TDAFW pump, with suction from the CST, will provide water to the SGs to make up for steam release. The TDAFW pump starts automatically upon the loss of offsite power, and no operator action is required for the pump to supply water from the CST to the four SGs.

The CST has a minimum usable capacity of 385,646 gallons. According to the licensee this represents sufficient volume for at least the first 31.1 hours of the ELAP event, including the makeup required both for decay heat removal and the sensible heat associated with the maximum RCS cooldown rate of 100°F/hour. The CST has been upgraded to be robust relative to all applicable external hazards.

The licensee's Phase 1 strategy directs the initiation of a cooldown and depressurization of the RCS between 2 and 8 hours after the start of the ELAP event. Over a period of approximately 2 to 4 hours, operators will gradually cool down the RCS from post-trip conditions until SG pressure reaches 310 psig, which corresponds to an RCS cold-leg temperature of approximately 415°F. Cooldown and depressurization of the RCS significantly extends the expected coping time under ELAP/loss of normal access to the UHS conditions because it: (1) reduces the potential for damage to RCP seals, and (2) allows borated water stored in the nitrogen-pressurized SI accumulators to passively inject into the RCS to offset system leakage and add negative reactivity. Stabilizing the RCS at these plant conditions allows steam pressure to remain high enough to support continued use of the TDAFW pump. To remove heat, operators will discharge steam through the SG ARVs, which can be operated remotely or locally under ELAP conditions, with motive force provided by a safety-related backup nitrogen system. A FLEX air compressor is deployed to assist with continued ARV operation prior to the predicted depletion of the nitrogen backup system. In the event that these methods are unavailable, the ARVs associated with the "B" and "C" SGs can be operated locally using hand wheels.

3.2.1.1.2 Phase 2

The licensee's FIP states that the primary Phase 2 strategy for core cooling would be to continue using the SGs as a heat sink, with makeup water supplied by the CST. The turbine driver of the TDAFW pump is designed to operate with steam inlet pressures as low as 92 pounds per square inch absolute (psia); therefore, the licensee expects that the pump will remain in operation throughout Phase 2. If the TDAFW pump does become unavailable due to insufficient steam pressure, the licensee can use a portable, diesel-driven FLEX core cooling pump, with a capacity of 500 gallons per minute (gpm). This FLEX core cooling pump can take suction from either the CST (primary strategy) or from the discharge of the FLEX CST makeup pump (alternate strategy).

Before the volume of the CST is depleted, operators will establish makeup flow to the CST. A portable, submersible, electric-driven FLEX CST makeup pump will be deployed in either the "A" or "B" ESW pump room, taking suction from the UHS. The discharge of this pump would normally be aligned to the CST using flexible hoses, but could also be aligned to directly supply the suction of the FLEX core cooling pump. The FLEX CST makeup pumps (two total, one is stored in each FLEX Storage Building (FSB)) each have a capacity of 450 gpm.

Upon declaration of an ELAP event, operators will begin deploying a trailer-mounted 480 Vac FLEX DG, which can be placed into service no later than 8 hours after the start of the event. One 500-kW FLEX DG would support the necessary 480 Vac Class 1E safety-related buses, Class 1E battery chargers, a FLEX RCS makeup pump, and a FLEX electric air compressor.

The backup nitrogen supply to the ARVs is designed to support valve operation for the first 8 hours of the ELAP event. To ensure long-term operability of the ARVs, two electric FLEX air compressors (which represent "N+1" capability) are stored in the Auxiliary Building. Air from the receiver of a FLEX air compressor would be directed to the four ARV accumulators through the use of hoses and a manifold. Two diesel-powered FLEX air compressors are also stored in the FSBs to provide an alternate air supply for the ARV accumulators.

3.2.1.1.3 Phase 3

The licensee's core cooling strategy in Phase 3 is initially a continuation of the Phase 2 strategy, with either the TDAFW pump or FLEX core cooling pump supplying makeup water to the SGs. The licensee's FIP notes that injection of non-standard (UHS or cooling lake) water cannot be continued indefinitely, and a water treatment system from the NSRC would be deployed during Phase 3 to ensure the long-term quality of SG makeup water.

Eventually, a high-capacity pump, three smaller submersible pumps, three 4160 Vac combustion turbine generators (CTGs), and an electrical distribution bus from the NSRC could support the restoration of the ESW system, which would provide cooling water to the CCW heat exchangers. The CCW system in turn allows operation of the RHR system for long-term core cooling. The NSRC CTG could also power the electrical components of the CCW and RHR systems. According the WCGS Updated Safety Analysis Report (USAR) [Reference 50], Sections 3.2, 5.4.7 and 9.2.2, the relevant portions of the CCW and RHR systems are safety-related, primarily located in the Auxiliary Building or containment, and thus protected from the effects of natural phenomena, and are designed to remain functional following a Safe Shutdown earthquake (SSE). Thus the staff considers these systems to be robust.

3.2.1.2 RCS Makeup Strategy

3.2.1.2.1 Phase 1

Under ELAP conditions, RCS inventory will tend to diminish gradually due to leakage through RCP seals and other leakage points. Furthermore, the initial RCS cooldown would result in a contraction of the RCS inventory, to the extent that the pressurizer would drain and a vapor void would form in the upper head of the reactor vessel. As is typical of operating PWRs, active RCS makeup is not provided during Phase 1 and the beginning hours of Phase 2 under the postulated ELAP conditions. However, some passive injection from the nitrogen-pressurized accumulators would occur as the RCS is depressurized below the accumulator cover gas pressure, which would result in the addition of borated water to the RCS. As discussed further below, the licensee has determined that: (1) sufficient reactor coolant inventory would be available throughout Phase 1 to support heat transfer to the SGs via natural circulation without crediting the active injection of RCS makeup, and (2) according to the core operating history specified in NEI 12-06, a sufficient concentration of xenon-135 should exist in the reactor core to ensure sub-criticality throughout Phase 1, considering the planned cooldown profile.

3.2.1.2.2 Phase 2

In order to maintain sufficient borated RCS inventory in Phase 2, the licensee's FIP states that a portable, electric-powered, high-pressure FLEX RCS makeup pump would be deployed in the Auxiliary Building and aligned to take suction from one of the two BATs, or alternately from the RWST. The BATs are the preferred source for RCS injection; their minimum combined volume is 17,658 gallons of water, borated to between 7000-7700 parts per million (ppm). They are located within the Auxiliary Building and fully protected from all applicable external events. The RWST is robust to all external hazards with the exception of wind-borne missiles. It would contain a minimum volume of 394,000 gallons of water with a boron concentration of at least 2400 ppm. The licensee's shutdown margin evaluation concludes that a maximum of approximately 6000 gallons of highly concentrated BAT water, or approximately 20,000 gallons of the relatively less-concentrated RWST water, would be required to ensure long-term sub-criticality given the most restrictive core conditions. Both of these volumes are well bounded by the respective capacities of the borated makeup sources.

Two FLEX RCS makeup pumps are stored onsite. The pumps are stored in the Auxiliary Building, one on the 1974 foot (1974') elevation (basement), and the other on the 2000' elevation. The pump on the 2000' elevation can only take suction from the RWST (not from the preferred BATs); therefore, its use would be as a backup if the other pump is unavailable due to reasons such as postulated internal flooding resulting from a seismic event. The FLEX RCS makeup pumps each have a capacity of 36 gpm, which represents a significant margin over the licensee's calculated minimum required RCS injection capacity of 10 gpm from the BAT. During an ELAP event, a FLEX RCS makeup pump would be powered by the 480 Vac/500 kW FLEX DG. RCS injection is projected to begin no later than 13 hours after the initiating event. The primary injection path to the RCS would be to a primary high pressure coolant injection (EM) system isolation valve, via two flexible hoses and a hard FLEX intermediate pipe. The alternate flowpath would be from the FLEX RCS makeup pump discharge to an EM system boron injection tank (BIT) inlet isolation valve connection, via one flexible hose.

To prevent the injection of nitrogen cover gas from the SI accumulators into the RCS, operators will either isolate the accumulators by closing the motor-operated outlet valves, or vent them by opening the solenoid-operated vent valves. This will be done before RCS pressure drops low enough for the nitrogen gas to enter the RCS. During the audit process, the NRC staff confirmed both of these options are available by reviewing the licensee's FLEX Support Guideline (FSG) FSG-10, "Passive RCS Injection Isolation," Revision 0.

3.2.1.2.3 Phase 3

In Phase 3, the RCS makeup strategy is a continuation of the Phase 2 strategy, supplemented by equipment provided by the NSRC. In particular, a mobile boration unit, in conjunction with the water filtration system, would be deployed to establish a virtually unlimited supply of borated water for indefinite RCS inventory control. Also, a high-pressure injection pump will be supplied by the NSRC for additional backup as needed.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

There are no variations to the licensee's strategy to respond to an external flooding event.

3.2.3 Staff Evaluations

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

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

Core Cooling

The licensee provided descriptions in the FIP for the permanent plant SSCs to be used to support core cooling during Phase 1. The TDAFW pump provides feedwater to all four of the SGs. The TDAFW pump starts automatically on the loss of offsite power condition and is supplied by water from the CST. This pump is located in the Auxiliary Building, which is protected from all applicable external hazards. According to the licensee's FIP, the Auxiliary Feedwater (AFW) system is located within the Auxiliary Building (with the exception of a section of pump recirculation piping and the TDAFW pump exhaust pipe), and is thus protected from all applicable external hazards. The licensee's FIP also states that the section of the recirculation line piping not in the Auxiliary Building, but located in the CST pipe chase has been upgraded to be seismically qualified and also qualified for tornados and associated missiles. In addition, the licensee's USAR [Reference 50], Sections 10.4.9.2.2 and 10.4.9.3, describes how the TDAFW pump exhaust piping is not likely to crimp after a postulated missile strike to a degree that would impair the function of the pump. Based on the licensee's FIP statements and the USAR description, the NRC staff concludes that the TDAFW pump is robust.

The CST was upgraded for protection from tornado and seismic hazards to support the FLEX strategies and is also protected from the other applicable external hazards. The UHS is a fully protected water source for SG and CST makeup, with the cooling lake also available for all postulated external events except a seismic event. More detailed descriptions of the CST, UHS, and cooling lake are provided in Section 3.10.1 of this safety evaluation.

The licensee's FIP describes how the SG ARVs will be used to support reactor core cooling and decay heat removal. The SG ARVs are air-operated globe valves, which are supplied by a safety-related nitrogen system. The SG ARVs are controlled remotely from the Main Control Room (MCR) or locally. The licensee's plan indicates that FLEX air compressors will be deployed to support long-term operation of the SG ARVs. The SG ARVs are located in the Auxiliary Building, which is protected from all applicable external hazards.

During the audit, the NRC staff reviewed the locations of the permanent SSCs described above to confirm that the SSCs will be protected from all applicable external hazards and have supporting procedures or guidelines that will direct operators to take action as needed to support the FLEX strategy. Based on the design and location of the protected water sources, as well as the permanent and newly installed plant SSCs as described in the FIP, the NRC staff finds that the licensee's strategy should support core cooling during an ELAP caused by a BDBEE, consistent with Condition 4 of NEI 12-06, Section 3.2.1.3.

RCS Inventory Control

According to the licensee's FIP, the FLEX RCS makeup pumps draw suction from borated water sources. The BATs are the preferred borated water source for RCS makeup and are protected from all applicable external hazards. The RWST is a stainless steel, seismically-qualified tank located outside of the Control Building. The RWST is protected from all applicable external hazards, except for high wind (tornado) missiles. Both the BATs and RWST are described in more detail in Section 3.10.2 of this safety evaluation. The licensee also described in its FIP that four SI accumulators are available to begin injecting borated water into the RCS when system pressure drops below approximately 600 psig. The accumulators are Seismic Category I tanks located in containment, which is protected for all applicable external hazards.

Based on the potential for flooding of the 1974' elevation of the Auxiliary Building due to the potential rupture of non-seismic piping, the licensee's plan locates the second FLEX RCS makeup pump on the 2000' elevation, where it would not be susceptible to this postulated flooding mechanism during a seismic event. A pump on the 2000' level would only have access to the RWST as a suction source (not the BATs). Since the RWST is seismically robust, this meets the NEI 12-06 provision for having "N" pathways available after a beyond-design-basis (BDB) seismic event, and is thus acceptable. However since there is not a robust "N+1" capability (pump or pathway) for a BDB seismic event, the licensee must follow the provisions of Section 11.5.4.b of NEI 12-06, Revision 2, which states that: "The required FLEX equipment may be unavailable for 90 days provided that the site FLEX capability ("N") is met. If the site "N" capability is met but not protected for all of the site's applicable hazards, then the allowed unavailability is reduced to 45 days." Thus, if the FLEX RCS makeup pump stored on the 2000' elevation of the Auxiliary Building is unavailable for any reason, the licensee would not have "N" sets of equipment protected from a potential seismic event. In this case, the allowed unavailability for this pump would be 45 days instead of 90 days. A similar logic would be applied to the connections for the FLEX RCS makeup pump on the 2000' elevation (electrical and mechanical). Specifically, if only a singular connection is postulated to be available after a seismic event with the induced flooding, than that connection's allowed out-of-service time would be 45 days instead of 90 days. Similarly, since the RWST is not tornado missile protected and the 2000' elevation RCS makeup pump can only use the RWST as a suction source, if the 1974' elevation RCS makeup pump is out-of-service, the licensee would not have

"N" sets of equipment protected from a postulated tornado missile, and the 1974' elevation RCS makeup pump would thus be restricted to a 45 day out-of-service time. During the onsite audit, the NRC staff walked down the RCS makeup provisions to confirm that the licensee will have at least one fully protected pathway for all applicable external events.

3.2.3.1.2 Plant Instrumentation

According to the licensee's FIP, the following instrumentation would be relied upon to support its core cooling and RCS inventory control strategy:

- RCS hot leg temperature
- RCS cold leg temperature
- Core exit thermocouple temperature
- RCS wide range pressure
- pressurizer level
- reactor vessel level indication system
- SG pressure
- SG water level (wide range and narrow range)
- AFW flow rate
- CST level
- excore nuclear instruments

The staff reviewed the licensee's available instruments and notes that the instrumentation available to support the strategies for core cooling and RCS inventory during the ELAP event is consistent with, and for some variables exceeds, the recommendations specified in the endorsed guidance of NEI 12-06.

These instruments are protected from all applicable external events, and supported by Class 1E power. In Phase 2, long-term power is established for these essential instruments by deploying a FLEX DG to power the Class 1E safety-related buses and Class 1E battery chargers. The primary monitoring strategy for all of these parameters is to obtain readings from the MCR. Alternatively, FSG-07, "Loss of Vital Instrumentation or Control Power," provides guidance for operators to locally read instruments using portable FLEX equipment. Based on this information provided by the licensee, the NRC staff concludes that indication for the above parameters would be available and accessible continuously throughout the ELAP event.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee's mitigating strategy for reactor core cooling is based, in part, on a generic thermal-hydraulic analysis performed for a reference Westinghouse four-loop reactor using the NOTRUMP computer code. The NOTRUMP code and corresponding evaluation model were originally submitted in the early 1980s as a method for performing licensing-basis safety analyses of small-break loss-of-coolant accidents (LOCAs) for Westinghouse PWRs. Although NOTRUMP has been approved for performing small-break LOCA analysis under the conservative 10 CFR Appendix K paradigm and constitutes the current evaluation model of record for many operating PWRs, the NRC staff had not previously examined its technical adequacy for performing best-estimate simulations of the ELAP event. Therefore, in support of mitigating strategy reviews to assess compliance with Order EA-12-049, the NRC staff evaluated licensees' thermal-hydraulic analyses, including a limited review of the significant assumptions and modeling capabilities of NOTRUMP and other thermal-hydraulic codes used for these analyses. The NRC staff's review included performing confirmatory analyses with the NRC's TRACE code to obtain an independent assessment of the duration that reference reactor designs could cope with an ELAP event prior to providing makeup to the RCS.

Based on its review, the NRC staff questioned whether NOTRUMP and other codes used to analyze ELAP scenarios for PWRs would provide reliable coping time predictions in the reflux or boiler-condenser cooling phase of the event because of challenges associated with modeling complex phenomena that could occur in this phase, including boric acid dilution in the intermediate leg loop seals, two-phase leakage through RCP seals, and primary-to-secondary heat transfer with two-phase flow in the RCS. Due to the challenge of resolving these issues within the compliance schedule specified in Order EA-12-049, the NRC staff requested that industry provide makeup to the RCS prior to entering the reflux or boiler-condenser cooling phase of an ELAP, such that reliance on thermal-hydraulic code predictions during this phase of the event would not be necessary.

Accordingly, the ELAP coping time prior to providing makeup to the RCS is limited to the duration over which the flow in the RCS remains in natural circulation, prior to the point where continued inventory loss results in a transition to the reflux or boiler-condenser cooling mode. In particular, for PWRs with inverted U-tube SGs (such as Wolf Creek), the reflux cooling mode is said to exist when vapor boiled off from the reactor core flows out the saturated, stratified RCS hot legs and condenses in the SG tubes, with the majority of the condensate subsequently draining back into the reactor vessel through the hot legs in countercurrent fashion. Quantitatively, as reflected in documents such as the PWR Owners Group (PWROG) report PWROG-14064-P, "Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances," Revision 0, [Reference 40] the commercial nuclear power industry has proposed defining this coping time as the point at which the 1-hour centered time-average of the flow quality passing over the SG tubes' U-bend exceeds one-tenth (0.1). As discussed further in Section 3.2.3.4 of this evaluation, a second metric for ensuring adequate coping time is associated with maintaining sufficient natural circulation flow in the RCS to support adequate mixing of boric acid.

With specific regard to NOTRUMP, preliminary results from the NRC staff's independent confirmatory analysis performed with the TRACE code indicated that the coping time for Westinghouse PWRs under ELAP conditions could be shorter than predicted in Westinghouse Commercial Atomic Power (WCAP)-17601-P, "Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering and Babcock & Wilcox NSSS Designs" [Reference 41]. Subsequently, a series of additional simulations performed by the staff and Westinghouse identified that the discrepancy in predicted coping time could be attributed largely to differences in the modeling of RCP seal leakage. These comparative simulations showed that when similar RCP seal leakage boundary conditions were applied, the coping time predictions of TRACE and NOTRUMP were in adequate agreement. From these simulations, as supplemented by review of key code models, the NRC staff obtained sufficient confidence that the NOTRUMP code may be used in conjunction with the WCAP-17601-P evaluation model for performing best-estimate simulations of ELAP coping time prior to reaching the reflux cooling mode.

Although the NRC staff obtained confidence that the NOTRUMP code is capable of performing best-estimate ELAP simulations prior to the initiation of reflux cooling using the one-tenth flow-quality criterion discussed above, the staff was unable to conclude that the generic analysis performed in WCAP-17601-P could be directly applied to all Westinghouse PWRs, as the vendor originally intended. In PWROG-14064-P, the industry subsequently recognized that the generic analysis would need to be scaled to account for plant-specific variation in RCP seal leakage. However, the staff's review, supported by sensitivity analysis performed with the TRACE code, further identified that plant-to-plant variations in additional parameters, such as RCS cooldown terminus, accumulator pressure and liquid fraction, and initial RCS mass, could also result in substantial differences between the generically predicted reference coping time and the actual coping time that would exist for specific plants.

As identified in the licensee's FIP, WCNOC relies on a plant-specific model for WCGS, based on the generic analysis of WCAP-17601. The licensee's analysis incorporates plant specific values for partial volumes of the RCS and assumes significantly lower RCP seal leakage due to the installation of low-leakage seals. During the audit process, the NRC staff reviewed the licensee's analysis, documented in calculation CN-SEE-I-12-34-P, "Wolf Creek Reactor Coolant System (RCS) Inventory, Shutdown Margin and Mode 5/6 Boric Acid Precipitation Control Analysis to Support the Diverse and Flexible Coping Strategy (FLEX)," Revision 0, which concludes that single-phase natural circulation would continue for up to 48.3 hours after the start of the ELAP without any active RCS injection. Based on this review, the staff concludes that licensee's strategy to commence FLEX RCS injection no later than 13 hours into the event would prevent the onset of reflux cooling, with considerable margin. The time requirement to borate the RCS to maintain adequate shutdown margin is significantly more limiting than the requirement to prevent reflux cooling, due to the presence of low-leakage RCP seals.

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 Reactor Coolant Pump (RCP) Seals

Leakage from RCP seals is among the most significant factors in determining the duration that a PWR can cope with an ELAP event prior to initiating RCS makeup. An ELAP event would interrupt cooling to the RCP seals, resulting in the potential for increased seal leakage and the failure of elastomeric O-rings and other components, which could further increase the leakage rate. As discussed above, as long as adequate inventory is maintained in the RCS, natural circulation can effectively transfer residual heat from the reactor core to the SGs and limit local variations in boric acid concentration. Along with cooldown-induced contraction of the RCS inventory, cumulative leakage from RCP seals governs the duration over which natural circulation can be maintained in the RCS. Furthermore, the seal leakage rate at the depressurized condition can be a controlling factor in determining the flow capacity requirement for FLEX pumps to offset ongoing RCS leakage and recover adequate system inventory.

Westinghouse Generation III SHIELD low-leakage RCP seals have been installed on all four RCPs at Wolf Creek. The SHIELD seal incorporates a thermally driven actuator mechanism that is designed to initiate automatically upon a loss of seal cooling. Upon activation, the seals are designed to allow a leakage rate of less than 1 gpm per RCP.

The NRC staff's audit review considered whether the SHIELD low-leakage seals have been credited in Wolf Creek's FLEX strategy in accordance with the four conditions identified in the NRC's endorsement letter of Westinghouse's white paper TR-FSE-14-1-P, "Use of Westinghouse SHIELD Passive Shutdown Seal for FLEX Strategies," dated May 28, 2014 [Reference 42]. The staff's audit review concluded that the licensee conforms to each condition from the NRC staff's endorsement letter as follows:

> Condition 1: Credit for the SHIELD seals is only endorsed for Westinghouse RCP Models 93, 93A, and 93A-1.

This condition is satisfied because the RCPs at Wolf Creek are Westinghouse Model 93A-1 RCPs.

> Condition 2: The maximum steady-state reactor coolant system (RCS) cold-leg temperature is limited to 571°F during the ELAP (i.e., the applicable main steam safety valve setpoints result in an RCS cold-leg temperature of 571°F or less after a brief post-trip transient).

The maximum steady-state RCP seal temperature during an ELAP event is expected to be the RCS cold leg temperature corresponding to the lowest SG safety relief valve setting. Per the WCGS technical specifications, Table 3.7.1-2, the nominal lift setpoint is 1185 psig, with a ±3 percent tolerance. Therefore, this condition is satisfied, since the saturation temperature at this pressure (with ±3 percent tolerance applied) corresponds to an RCS cold leg temperature of approximately 571°F.

> Condition 3: The maximum RCS pressure during the ELAP (notwithstanding the brief pressure transient directly following the reactor trip comparable to that predicted in the applicable analysis case from WCAP-17601-P) is as follows: For Westinghouse Models 93 and 93A-1 RCPs, RCS pressure is limited to 2250 psia; for Westinghouse Model 93A RCPs, RCS pressure is to remain bounded by Figure 7.1-2 of TR-FSE-14-1-P, Revision 1.

Normal operating pressure at Wolf Creek is 2235 psig, per the plant's USAR. Allowing for the possibility of a brief pressure transient directly following the reactor trip, the NRC staff concludes that the licensee's mitigating strategy of cooling the reactor core via the main steam safety valves and/or SG ARVs will otherwise maintain reactor pressure below 2250 psia.

> Condition 4: Nuclear power plants that credit the SHIELD seal in an ELAP analysis shall assume the normal seal leakage rate before SHIELD seal actuation, and a constant seal leakage rate of 1.0 gallon per minute for the leakage after SHIELD seal actuation.

The licensee's FIP and supporting calculations assume a constant Westinghouse SHIELD RCP seal package leakage rate of 1 gpm per RCP, plus 1 gpm of unidentified RCS leakage, for a total RCS leakage of 5 gpm. The actual seal leakage rate expected during an ELAP event would exceed this value for a brief period prior to actuation of the SHIELD seal (according to the actuation range specified in TR-FSE-14-1-P, actuation of the SHIELD seal would occur well within 1 hour of ELAP event initiation). As noted previously, the licensee has calculated that reflux cooling would not be entered for over 48 hours into the event, even if FLEX RCS makeup flow were not provided as planned. Since the licensee's mitigating strategy directs RCS makeup to begin approximately 13 hours after event initiation, ample margin exists to accommodate the small additional volume of leakage that is expected to occur prior to actuation of the SHIELD seal.

The seal leakoff analysis assumes no failure of the seal design, including the elastomeric O-rings. In order to confirm the validity of the licensee's FIP Table 4 assumption of a 1 gpm leak rate per pump, during the audit review the staff questioned whether the licensee has installed high-temperature-qualified RCP seal O-rings at Wolf Creek. The licensee indicated that high temperature qualified 7228-C or 7228-D type O-rings were installed and that only equivalent or better O-rings would be used in the future. Based on these factors, the staff's audit review concluded that O-ring failure for Wolf Creek during a beyond-design-basis ELAP event should not occur, and the 1 gpm leak rate assumed per pump is appropriate.

Based upon the discussion above, the NRC staff concludes that the RCP seal leakage rates assumed in the licensee's thermal-hydraulic analysis are appropriate.

3.2.3.4 Shutdown Margin Analyses

In the analyzed ELAP event, the loss of electrical power to control rod drive mechanisms is assumed to result in an immediate reactor trip with the full insertion of all control rods into the core. The insertion of the control rods provides sufficient negative reactivity to achieve sub-criticality at post-trip conditions. However, as the ELAP event progresses, the shutdown margin for PWRs is typically affected by several primary factors:

- the cooldown of the RCS and fuel rods adds positive reactivity

- the concentration of xenon-135, which (according to the core operating history assumed in NEI 12-06) would

  - initially increase above its equilibrium value following reactor trip, thereby adding negative reactivity
  - peak at roughly 12 hours post-trip and subsequently decay away gradually, thereby adding positive reactivity

- the passive injection of borated makeup from nitrogen-pressurized accumulators due to the depressurization of the RCS, which adds negative reactivity

At some point following the cooldown of the RCS, PWR licensees' mitigating strategies generally require active injection of borated coolant via FLEX equipment. In many cases, boration would become necessary to offset the gradual positive reactivity addition associated with the decay of xenon-135; but, in any event, borated makeup would eventually be required to offset ongoing RCS leakage. The necessary timing and volume of borated makeup depend on the particular magnitudes of the above factors for individual reactors.

The specific values for these and other factors that could influence the core reactivity balance that are assumed in the licensee's current calculations could be affected by future changes to the core design. However, NEI 12-06, Section 11.8 states that "[e]xisting plant configuration control procedures will be modified to ensure that changes to the plant design ... will not adversely impact the approved FLEX strategies." Inasmuch as changes to the core design are changes to the plant design, the NRC staff expects that any core design changes, such as those considered in a core reload analysis, will be evaluated to determine that they do not adversely impact the approved FLEX strategies, especially the analyses which demonstrate that re-criticality will not occur during a FLEX RCS cooldown.

During the audit, the NRC staff reviewed the licensee's shutdown margin calculation, LTR-SEE-17-87, "Evaluation of the Wolf Creek Reactor Coolant System (RCS) Boration Capability during an Extended Loss of AC Power Event," Revision 0. The licensee's analysis was performed for a range of core times-in-life, including beginning-of-life, most reactive time-of-life, middle-of-life, and end-of-full-power-capability (EOFPC). The key assumptions in this analysis included: initial and final RCS temperatures, core cooldown profile (100°F/hour beginning at 8 hours following the loss of ac power), minimum BAT/RWST boron concentrations, and high BAT/RWST liquid temperatures. Notably, the licensee's analysis assumes a 10 gpm injection rate from the BAT, and a 20 gpm injection rate from the RWST; these volumetric rates are conservative relative to the actual FLEX RCS makeup pump design flow rate of 36 gpm.

The licensee's calculations credited the transient reactivity effects due to core xenon, but the final required RCS boron concentrations were based on the shutdown margin boron concentrations at 100 hours following shutdown, i.e., with no negative reactivity added by xenon, which is conservative. The licensee also assumed that no borated water from the accumulators passively injects into the RCS during cooldown and depressurization, and the NRC staff concurs that this assumption is acceptable for the Wolf Creek shutdown margin analysis. Based on a review of the licensee's procedures and guidelines, the staff concludes that any passively injected volume from accumulators (which can be at a lower boron concentration than the RWST or BATs) will not prevent operators from actively borating the RCS to the required boron concentration.

According to the FIP, borated water from either the BAT or the RWST will be injected into the RCS no later than 13 hours into the event. Based on the NRC staff's audit review of the shutdown margin calculation and associated figures, the licensee determined that the reactor will remain subcritical, with k_{eff} [effective neutron multiplication factor] ≤ 0.99, at 415°F (i.e., the saturation temperature corresponding to the target SG pressure of the RCS cooldown) for at least 20 hours following reactor shutdown without any active boron injection. The required borated injection volumes were based on achieving a final RCS boron concentration that meets or exceeds the 200°F, 100-hour (xenon free), 1 percent shutdown margin boron concentration for the applicable core time-in-life. The licensee's procedures/guidelines provide the necessary administrative control to ensure that RCS cooldown below 415°F will not be performed until the RCS boration to a cold, xenon-free value is complete.

If one of the BATs is used for RCS boration, which is the licensee's preferred strategy, the licensee concluded that a maximum volume of 5460 gallons of water at the minimum technical specification boron concentration of 7000 ppm would need to be injected to maintain 1 percent shutdown margin (EOFPC proved to be the most limiting core time-in-life, regardless of the injection source). Assuming makeup initiation at 13 hours after the ELAP at a rate of 10 gpm, this injection would be concluded no later than 23.1 hours into the event. This includes a 1 hour delay to account for uniform boron mixing in the RCS. Similarly, if the RWST is the boration source, a maximum volume of 18,330 gallons of water at the minimum technical specification boron concentration of 2400 ppm would need to be injected to maintain 1 percent shutdown margin. Assuming an injection rate of 20 gpm, this injection would be completed by 34.8 hours into the event (again, including 1 hour to account for boron mixing).

For either injection source and for all core times-in-life, initiating RCS injection at 13 hours into the event effectively maintains RCS boric acid concentration above the level required to maintain shutdown margin at 415°F, which varies over time as xenon decays. Therefore, the licensee's analysis demonstrates that commencing active RCS injection at 13 hours into the event will maintain shutdown margin throughout the duration of the ELAP response, at all core times-in-life.

If the RWST is used as the boration source, depending on the core time-in-life, operators may need to vent the RCS to ensure that they can inject a volume of borated coolant that is sufficient to satisfy shutdown margin requirements. During the audit process, the staff confirmed that the licensee's guideline FSG-8, "Alternate RCS Boration" contains provisions to accomplish the necessary venting, via the head vent valves (preferred) or a RCS power operated relief valve.

The NRC staff's audit review of the licensee's shutdown margin calculation further determined that credit was taken for uniform mixing of boric acid during the ELAP event. The NRC staff had previously requested that the industry provide additional information to justify that borated makeup would adequately mix with the RCS volume under natural circulation conditions potentially involving two-phase flow. In response, the PWROG submitted a position paper, dated August 15, 2013 (withheld from public disclosure due to proprietary content), which provided test data regarding boric acid mixing under single-phase natural circulation conditions and outlined applicability conditions intended to ensure that boric acid addition and mixing during an ELAP would occur under conditions similar to those for which boric acid mixing data is available. By letter dated January 8, 2014 [Reference 43], the NRC staff endorsed the above position paper with three conditions:

- The required timing and quantity of borated makeup should consider conditions with no RCS leakage and with the highest applicable leakage rate.

- Adequate borated makeup should be provided either: (1) prior to the RCS natural circulation flow decreasing below the flow rate corresponding to single-phase natural circulation, or (2) if provided later, then the negative reactivity from the injected boric acid should not be credited until 1 hour after the flow rate in the RCS has been restored and maintained above the flow rate corresponding to single-phase natural circulation.

- A delay period adequate to allow the injected boric acid solution to mix with the RCS inventory should be accounted for when determining the required timing for borated makeup. Provided that the flow in all loops is greater than or equal to the corresponding single-phase natural circulation flow rate, a mixing delay period of 1 hour is considered appropriate.

Based on the FIP, and the audit review of the licensee's calculations, procedures and guidelines, the NRC staff concludes that the licensee will comply with the PWROG position paper on boric acid mixing, including the conditions imposed in the staff's corresponding endorsement letter, with one exception. The staff notes that the licensee's analyses considered only the case that RCS leakage through the RCP seals is zero; however, the staff also notes that this is the most conservative assumption for the purposes of shutdown margin and RCS boron concentration. The NRC staff further confirmed that the licensee would provide RCS makeup prior to RCS loop flow decreasing below the single-phase natural circulation flow rate. Finally, the NRC staff also confirmed that the licensee's calculations adequately incorporated a 1-hour delay to account for the mixing time of boric acid in the RCS, as noted earlier in this section.

Finally, the NRC staff's audit review notes that while the licensee's analyses demonstrate adequate shutdown margin for an RCS average temperature as low as 200°F, the licensee may further cool the RCS below this temperature as Phase 3 progresses. However, considering the time duration available prior to completing this cooldown, the NRC staff concludes that the licensee's strategy has provisions that would accomplish the following: (1) obtain sufficient equipment (i.e., including both onsite and NSRC-supplied equipment, such as mobile boration units) and supplies of powdered boric acid to prepare the required borated water, (2) allow sufficient time to vent the RCS, if necessary, to allow injection of the volume of borated water required to ensure adequate shutdown margin, and (3) allow sufficient time for the licensee's technical support personnel to determine, if necessary, the boron concentration required to ensure adequate shutdown margin. The staff also confirmed that the licensee's procedures and guidelines include steps to consult with the plant engineering staff and/or the Technical Support Center when evaluating long-term plant status.

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

For SG makeup, the licensee's FIP describes two FLEX core cooling pumps, one pump stored in each of two FSBs. Having two FLEX core cooling pumps is intended to satisfy the "N+1" provision of NEI 12-06. The FLEX core cooling pump is a trailer-mounted, diesel engine-driven centrifugal pump that is deployed and staged for SG makeup in the event that the TDAFW pump can no longer perform its function. Each of these pumps has a design flow rate of 500 gpm and a head of 1155 feet. The minimum required flow rate for SG makeup is 360 gpm. The CST and UHS are the protected water sources that can supply the FLEX core cooling pump. The cooling lake is also available for all postulated external events except for a seismic event. The licensee also indicated that other preferred water sources may be used if they are available after initiation of the ELAP event prior to using the cooling lake or UHS. These additional water sources are described in Section 3.10.1 of this safety evaluation.

For RCS makeup, the licensee's FIP describes two FLEX RCS makeup pumps, both permanently staged in the Auxiliary Building. One RCS makeup pump is stored at the 1974' elevation in the Auxiliary Building and the second pump is stored at the 2000' elevation of the Auxiliary Building. According to the FIP, the FLEX RCS makeup pumps are both capable of delivering 36 gpm. The BATs and RWST are the borated water sources used for supplying the FLEX RCS pumps.

The licensee's FIP (Section 2.3.5.3, Section 2.3.5.4, Table 4, and Figures 3a through 3d) describes the feature of a CST makeup pump. In the FIP, Section 2.3.5.3 also clarifies that there will be one CST makeup pump stored in each FLEX building (total of two pumps). This pump provides water from the UHS or cooling lake to the CST once the available CST volume is depleted. During the audit process, the NRC staff confirmed that the licensee's program plan, AP-21A-002, "Diverse and Flexible Coping Mitigation Strategies (FLEX) Program," Revision 0, contains provisions for the two pumps. According to the licensee's program plan, one CST makeup pump is deployed to provide makeup to the CST. Thus the second pump is intended to provide a backup capability ("N+1").

During the audit process, the NRC staff reviewed calculations FD-13-007, "AFW FLEX Hydraulic Evaluation," Revision 0, and FD-13-008, "RCS FLEX Hydraulic Evaluation," Revision 0. These calculations evaluated the FLEX pumps providing SG and RCS makeup. The NRC staff review concludes that based upon the capability of these FLEX pumps and the respective FLEX connections being made as directed by the FSGs that sufficient makeup capability exists in the licensee's plan. In addition, the hydraulic evaluation for CST makeup, FD-13-006, "FLEX CST Hydraulic Evaluation," Revision 1, was also reviewed and the staff confirmed that one pump has the required capability to supply up to 450 gpm of water makeup, accounting for the needed flow to support long-term core cooling and SFP makeup.

During the onsite audit, the NRC staff also conducted a walk down of the hose deployment routes for the above FLEX pumps to confirm the evaluations of the pump staging locations, hose distance runs, and connection points as described in the above hydraulic analyses and the FIP.

3.2.3.6 Electrical Analyses

The licensee's electrical strategies provide power to the equipment and instrumentation used to mitigate the ELAP and loss of normal access to the UHS. 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 safety evaluation. During the audit process, the NRC staff reviewed the licensee's FIP, electrical single-line diagrams, and summaries of calculations for sizing the FLEX generators and station batteries.

During the first phase of an ELAP event, the licensee plans to rely on the Class 1E station batteries and inverters to provide power to key instrumentation for monitoring parameters and controls for SSCs used to maintain the key safety functions. The Class 1E station batteries, inverters, and associated dc distribution systems are located within a safety-related building which is a Seismic Category I structure. As such, the staff concludes that the Phase 1 electrical equipment is protected from a postulated BDEEE, consistent with the provisions of NEI 12-06.

The Class 1E batteries will be used initially to power required key instrumentation and other required loads. The licensee's procedures EMG-C-0, "Loss of All AC Power," Revision 32, and FSG-04, "ELAP DC Bus Load Shed and Management," Revision 0, were reviewed during the audit process. The NRC staff confirmed that plant operators are directed to prolong dc power availability during the event by stripping non-essential loads on the Class 1E batteries. Plant operators would do this to extend battery life to at least 8 hours or until backup power (Phase 2) is available. Plant operators would commence shedding loads within 45 minutes and complete load shedding within 60 minutes from the onset of the postulated event. The staff review of FSG-04 also confirmed that it provides operational guidance to shed non-essential loads within the time assumed in the licensee's analysis.

Wolf Creek has four independent Class 1E 125 Volt direct current (Vdc) batteries (NK011, NK012, NK013, and NK014). Calculation CN-PEUS-12-12, "Wolf Creek FLEX Battery Coping Analysis," Revision 1, was reviewed during the audit process and it showed that the WCGS Class 1E batteries were manufactured by AT&T [American Telephone and Telegraph] Round Cell Technologies. The NK011 and NK014 batteries are type LIST-1S-48 μΩ, 1600 ampere – hour (A-H) and the NK012 and NK013 batteries are type LIST-2S-48 μΩ, 864 A-H.

Calculation CN-PEUS-12-12 evaluated the capacity of the Class 1E batteries to supply dc power to the required loads during the first phase of the WCGS FLEX mitigation strategies for an ELAP event. The licensee's evaluation identified the required loads and their associated ratings (ampere (A) and minimum required voltage) and the non-essential loads on the batteries that would be shed within 60 minutes of the start of the event to ensure battery operation for at least 8 hours. This strategy provides sufficient margin to transition to Phase 2 as the licensee expects the onsite portable FLEX DG to be deployed, staged, connected to repower the Class 1E battery chargers within 8 hours after onset of an ELAP event.

Based on the NRC staff's review of the licensee's analysis and procedures, the battery vendor's capacity and discharge rates for the Class 1E station batteries, the NRC staff finds that the WCGS dc systems should have adequate capacity and capability to power the loads required to mitigate the consequences during Phase 1 of an ELAP, provided that the necessary load shedding is completed within the times assumed in the licensee's analysis.

During Phase 2, the licensee's strategy includes transition from installed equipment to the onsite portable FLEX equipment. The licensee's FIP describes the use of one of two 480 Vac, 500 kW, trailer mounted, portable FLEX DGs to power the Phase 2 loads. The FLEX 480 Vac DGs are stored in two separate FSBs. The licensee also discussed primary and alternate strategies for supplying power to Phase 2 equipment using a combination of portable FLEX and permanently installed, seismically robust components.

The NRC staff reviewed the calculation DAR-PEUS-12-6, "FLEX Electrical Conceptual Design for the Wolf Creek Nuclear Operating Corporation," Revision 1, and the licensee's evaluation of Phase 2 loading and the capacity of the FLEX DGs during the audit process. The licensee's evaluation showed that 1 FLEX DG will power 4 vital battery chargers, a FLEX RCS makeup pump, an air compressor, an accumulator isolation valve, 12 back up fans, 4 lighting panels, and a BAT tank room/area heater for an extreme cold event. The licensee estimated that the total Phase 2 loads would be 340 kW, which is within the rated capacity of a FLEX 500 kW DG. In assessing this capability, the NRC staff reviewed conceptual single line electrical diagrams, the separation and isolation of the FLEX DGs from the Class 1E emergency diesel generators (EDGs), and guideline FSG-05, "Initial Assessment and FLEX Equipment Staging," Revision 0, that provides guidance to the plant operators to deploy and stage the FLEX 500 kW DG.

The WCGS strategy also includes use of one of the two 15 kW 120/240 Vac portable DGs to power temporary ventilation fans. One 15 kW DG would be stored in each FSB. Each fan is rated for 0.7 kW. As such, a 15 kW DG should have adequate capacity to support four fans. The licensee's guideline, FSG-05, provides instructions to deploy and stage the FLEX 15 kW DG near the southwest door of the Control Building.

Based on its review, the NRC staff concludes that the Phase 2 FLEX DGs should have adequate capability and capacity to power the loads needed during an ELAP event.

The licensee plans to use one FLEX 500 kW DG as the 'N' DG and other one as the "N+1" DG. If the "N" FLEX DG becomes unavailable or is out-of-service for maintenance, the other ("N+1") FLEX DG would be deployed to support the required loads. The "N+1" FLEX DG is identical to the "N" FLEX DG, thus ensuring electrical compatibility and sufficient electrical capacity in an instance where substitution is required. Since the "N+1" FLEX DG is identical and interchangeable with the "N" FLEX DG, the NRC staff finds that the licensee has met the provisions of NEI 12-06, for spare equipment capability regarding the Phase 2 FLEX DGs. Similarly, the licensee has two 15 kW DGs ("N" and"N+1") to support certain portable ventilation fans included in the overall strategy.

For Phase 3, the licensee plans to continue the Phase 2 coping strategy with additional assistance provided from offsite equipment/resources as needed. The offsite resources that will be provided by an NSRC include three 1.0 megawatt (MW), 4160 Vac CTGs, one 480 Vac, 1.1 MW CTG, a distribution center, and cables. According to the licensee's FIP, the three NSRC-supplied 4160 Vac CTGs, connected to the distribution panel and operating in parallel, will supply power to one of the two Class 1E 4160 Vac buses. The licensee's FIP states that the three CTGs are adequate to supply the Phase 3 loads. Additionally, by restoring a Class 1E 4160 Vac bus, power can be restored to the Class 1E 480 Vac system via the 4160/480 Vac step down transformers to power selected 480 Vac loads. The NRC staff reviewed the licensee's evaluation DAR-PEUS-12-6, "FLEX Electrical Conceptual Design for the Wolf Creek Nuclear Operating Corporation," Revision 1. The licensee's evaluation showed that during Phase 3, one RHR Pump and one CCW Pump can be powered to cool the RCS and containment. Therefore, the total loads on the three NSRC 4160 Vac CTGs would include one 700 horsepower (hp) CCW pump, one 500 hp RHR pump, and the Phase 2 480 Vac loads (340 kW). Based on this, the total Phase 3 loads would be approximately 1,235 kW indicating that the 3.0 MW capacity of three CTGs operating in parallel will support starting and operating the Phase 3 loads. While the licensee's strategy does not rely on the NSRC-supplied 480 Vac CTG, it would be available for use, if necessary. Based on the FIP and analysis review, the NRC staff concludes that the electrical equipment the licensee plans to have delivered to WCGS from an NSRC should have adequate capacity to supply power to the Phase 3 loads.

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 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 3-2 and Appendix D 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 as described in Section 3.5.1 of this safety evaluation if it is higher than the site's SSE. During the event, the licensee selects the appropriate SFP makeup method to use based on plant conditions. The selected approach will also require 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 BDB, 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 effect of an ELAP with full core offload to the SFP is addressed in Section 3.11 of this safety evaluation. The licensee's plan includes provisions for providing spray flow to the SFP.

3.3.1 Phase 1

The licensee stated in its FIP that Phase 1 will consist of pre-staging FLEX equipment that will be used to perform SFP cooling in the Fuel Building. Operators will also create a vent path in the Fuel Building by opening up the roll-up door. These actions are performed within approximately 5 hours of the initiating ELAP event. The operators will also monitor SFP water level using reliable SFP level instrumentation installed per Order EA-12-051.

3.3.2 Phase 2

The licensee stated in its FIP that the SFP makeup will be initiated about 35 hours after declaration of ELAP. The FLEX SFP pump is trailer-mounted and will be towed from the FSB and staged either near the RWST or CST for SFP makeup. The mechanical connections and hose connections from the FLEX SFP pump to the designated areas in the Fuel Building for SFP cooling are described in Section 3.7.3.1 of this safety evaluation.

3.3.3 Phase 3

The licensee's FIP also states that the Phase 2 SFP makeup strategy is continued until NSRC equipment arrives on site. Additional low pressure/high flow pumps will be available from NSRC as a backup to the onsite FLEX SFP pumps.

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 permanent plant equipment contained in structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles, are 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) SFP cooling system is intact, including the attached piping.

The staff reviewed calculation CN-SEE-II-12-35, "Determination of the Time to Boil in the Wolf Creek Spent Fuel Pool[s] after an Earthquake," Revision 1, which provided the licensee's time to boil calculation for the SFP. This calculation indicates that boiling begins at approximately 5.59 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 pre-stage FLEX equipment needed for SFP makeup after ELAP event initiation while the SFP area remains habitable for personnel entry.

As described in the licensee's FIP, the licensee's Phase 1 SFP cooling strategy does not require any anticipated actions other than pre-staging hoses needed for SFP makeup. 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 by licensee guideline FSG-11, "Alternate SFP Makeup and Cooling," to open the rollup door on the 2000' elevation of the Fuel Building. Airflow through this door provides a vent pathway through which steam generated by SFP boiling can exit the Fuel Building.

The licensee's SFP cooling strategy involves the use of one of the two FLEX SFP pumps and associated hoses and fittings with suction from the RWST or the CST. The staff's evaluation of the robustness and availability of FLEX connections points for the FLEX SFP pump is discussed in Section 3.7.3.1 of this safety evaluation. Furthermore, the staff's evaluation of the robustness and availability of the RWST and CST is discussed in Section 3.10.3 of this safety evaluation.

3.3.4.1.2 Plant Instrumentation

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 portable generators. The NRC staff's review of the SFP level instrumentation is discussed in Section 4 of this safety evaluation.

3.3.4.2 Thermal-Hydraulic Analyses

In NEI 12-06, Section 3.2.1.6 states that SFP heat load assumes the maximum design-basis heat load for the site. In accordance with NEI 12-06, the licensee performed a thermal-hydraulic analysis of the SFP (CN-SEE-12-35) as a basis for the inputs and assumptions used in its FLEX equipment design. The calculation concluded that the SFP heat load will reach maximum boiling temperature in approximately 5 hours under normal, no load conditions. The time to boil off to 10 feet from the top of the active fuel is approximately 35 hours without any operator actions. For core off-load conditions, the calculation concluded that the SFP will reach maximum boiling temperature in approximately 2.4 hours. In this case the SFP boils off to reach the top of the fuel racks in 15.43 hours from initiation of the ELAP event. The licensee also indicated in the FIP that the SFP makeup will begin around 35 hours into the ELAP event at a flow rate of up to 250 gpm, which is higher than the minimum requirements of approximately 56 gpm (for normal conditions) and 132 gpm (for core off-load conditions), to replenish the SFP water being boiled off.

The NRC staff reviewed the calculation CN-SEE-II-12-35 and concludes that the licensee's projected SFP makeup flow rate capability will be able to maintain an adequate SFP level above the top of the active fuel for an ELAP occurring during normal power operation and shutdown conditions. Further, based on the information contained in the FIP, confirmed by the thermal-hydraulic calculation, the NRC staff finds that the licensee has considered the maximum design-basis SFP heat load, consistent with NEI 12-06 Section 3.2.1.6.

3.3.4.3 FLEX Pumps and Water Supplies

As described in the FIP, the SFP cooling strategy relies on one of two FLEX SFP pumps to provide SFP makeup during Phase 2 and 3. The second FLEX SFP pump satisfies the "N+1" provision of NEI 12-06. Each FSB on site will include one FLEX SFP pump and associated equipment for storage. Section 2.4.8.1 of the FIP references calculation FD-13-009, "SFP FLEX Hydraulic Evaluation," Revision 0, which describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX SFP pump. The FLEX SFP pump can provide SFP flow rate of 250 gpm at 92.2 psig discharge pressure, which exceeds the maximum SFP makeup requirements. The FLEX SFP pump is also capable of providing 125 gpm makeup flow to the SFP through the spray nozzles. The NRC staff notes that the performance criteria of a FLEX pump supplied from an NSRC for Phase 3 is equal to or better than the onsite FLEX SFP pump. Thus, if the onsite FLEX SFP pumps were to fail, further backup capability would be available. The NRC staff review of the SFP analysis described above concludes that it is consistent with

NEI 12-06, Section 11.2 and the FLEX equipment should be capable of supporting the SFP cooling strategy.

In NEI 12-06, Table 3-2 and Table D-3 summarize an acceptable 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 direct to pool 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. Additionally, JLD-ISG-2012-01 specifies that unless the site performs a seismic evaluation of their spent fuel pool, spray must be provided via portable nozzles from the refueling floor using a portable pump capable of providing a minimum of 200 gpm per unit (250 gpm if overspray occurs). The licensee has performed a seismic evaluation [Reference 44] for the SFP as part of its response to an NRC Request for Information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.54(f) (hereafter referred to as the 50.54(f) letter) [Reference 22]. The NRC has reviewed this SFP evaluation and concluded that Wolf Creek met the specified criteria and therefore, the licensee responded appropriately to the NRC's 50.54(f) letter regarding SFP integrity [Reference 45]. Thus, the NRC concludes that the licensee is not required to provide 250 gpm (assuming overspray) of SFP spray flow via portable nozzles to meet to meet NEI 12-06, Revision 2, as endorsed by the NRC. The staff further concludes that the licensee's provision for spray flow of 125 gpm through the portable nozzles provides additional capability.

3.3.4.4 Electrical Analyses

The NRC staff reviewed the licensee's electrical strategies for SFP cooling. The SFP will be monitored by instrumentation installed in response to NRC Order EA-12-051. This equipment has battery capacity for 72 hours. According to the licensee's FIP, onsite portable FLEX generators will provide alternative power to the SFP instrumentation display panels and to recharge the backup battery within 72 hours of an ELAP event. This would correspond to the 500 kW FLEX generator in the primary electrical strategy or the NSRC-supplied 4160 Vac CTGs in the backup electrical strategy. The NRC staff reviewed the licensee's Phase 2 480 Vac FLEX DG capability and Phase 3 4160 Vac CTGs, as described in Section 3.2.3.6 of this safety evaluation and concludes that the power sources should be adequate to supply alternate power to the SFP instrumentation display panels and to recharge the backup battery. The licensee did not credit any other electrical equipment for its SFP cooling strategy.

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

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 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

In industry guidance document, NEI 12-06, Table 3-2, 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. Wolf Creek has a dry ambient pressure containment.

During the audit process the NRC staff reviewed the licensee's containment evaluation, CN-OA-13-7, "Wolf Creek ELAP Containment Environment Analysis," Revision 1, which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy of automatic containment isolation and monitoring containment parameters and concluded that the containment parameters of pressure and temperature remain well below the respective USAR Section 6.2 design limits of 60 psig and 320°F for more than 72 hours. From its review of the evaluation, the NRC staff noted that actions to maintain containment integrity and also maintain the instrumentation credited in the strategy have been developed, and are summarized below.

3.4.1 Phase 1

The Phase 1 coping strategy involves verifying containment isolation per Emergency Management Guideline EMG C-0, "Loss of All AC Power" and monitoring containment temperature and pressure using installed instrumentation. According to the licensee's FIP, containment pressure and temperature indication will be available in the MCR.

3.4.2 Phase 2

The Phase 2 coping strategy is to continue the Phase 1 strategy. Phase 2 activities to repower key instrumentation using the FLEX DG are required to continue containment monitoring.

3.4.3 Phase 3

The Phase 3 coping strategy is to initially maintain the Phase 1 strategy. Eventually, the licensee plans to use offsite equipment to reduce containment temperature indirectly through the reduction of RCS heat load. The licensee's FIP provides a discussion of the use of NSRC-supplied equipment to accomplish this feature of the plan. The strategy will result in a restoration of the CCW, ESW, and RHR systems to support core cooling. Based on this strategy the licensee has determined that conditions in containment will improve with no further actions. Electrical needs for the equipment, including powering of the CCW and RHR systems involves using 4160 VAC CTGs and distribution bus supplied from the NSRC and tied into one of the 4160 VAC Class 1E buses. The NRC evaluation of the Phase 3 electrical loading is contained in Section 3.2.3.6 of this safety evaluation and includes these components.

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

In the WCGS USAR, Section 6.2 describes the reactor containment as a post-tensioned, pre-stressed, reinforced concrete, cylindrical structure with a hemispherical dome 205 feet tall and 140 feet in diameter and a conventionally reinforced concrete base slab with a central cavity and instrumentation tunnel to house the reactor vessel. The base slab, cylinder, and dome are reinforced by bonded reinforcing steel, as required by the design loading conditions. Additional reinforcing is provided at discontinuities in the structure and at major penetrations in the shell. The interior of the structure is lined with carbon steel plates welded together to form a barrier which is essentially leak tight. The containment internal free volume is 2,500,000 cubic feet. The internal design pressure is 60 psig and the design temperature is 320°F.

The Containment Building is a Seismic Category I structure. Seismic Category I structures, components, and systems are designed to withstand the SSE. According to the WCGS USAR, Section 3.2.1, Seismic Category I structures are sufficiently isolated or protected from the other structures to ensure that their integrity is maintained.

Essential Service Water (ESW) System

Portions of the ESW system are credited in the licensee's strategy to respond to the postulated ELAP and loss of normal access to the UHS. Specifically, the ESW system provides cooling water for the CCW system, which in turn cools the RHR system. The RHR system will eventually be used in Phase 3 to cool the RCS, and indirectly cool the containment, according to the licensee's FIP. The RHR and CCW systems are evaluated in Sections 3.2.1.1.3 and 3.2.3.6 of this safety evaluation. According to USAR Section 9.2.1.2, the ESW system consists of two redundant cooling water trains. The ESW system is safety-related, and is required to function following a design-basis accident to achieve and maintain the plant in a post-accident safe shutdown condition. The ESW system is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles and is designed to remain functional after an SSE. In addition, according the WCGS USAR, failure of any adjacent non-Seismic Category I structure will not constitute a hazard to the ESW system.

3.4.4.1.2 Plant Instrumentation

In NEI 12-06, Table 3-2 specifies that containment pressure is a key containment parameter which should be monitored by repowering the appropriate instruments. The licensee's FIP states that MCR 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 was available, the FIP states that key credited plant parameters, including containment pressure, would be available using alternate methods.

The licensee identified the following instrumentation as providing key parameters credited for all phases of the containment integrity strategy:

- Containment Pressure: Containment pressure indication is available in the MCR throughout the event.
- Containment Wide Range Temperature: Containment wide range temperature indication is available in the MCR throughout the event.
- Containment Radiation: Containment radiation indication in MCR is not available until the FLEX 480 VAC/500 kW DG is placed in service approximately eight hours after the declaration of an ELAP.

Containment radiation monitors provide containment radiation levels for use in FSG-12, "Alternate Containment Cooling in Mode 5 and 6," to support the containment venting strategy, if needed. Containment radiation instruments are repowered concurrently when the FLEX 480 VAC/500 kW DG is placed in service, well before containment venting is required.

In the unlikely event that 125 Vdc and 120 Vac Vital Bus infrastructure is damaged, FLEX strategy guidelines for alternately obtaining the critical parameters locally is provided in FSG-07, "Loss of Vital Instrumentation or Control Power".

3.4.4.2 Thermal-Hydraulic Analyses

The licensee completed containment evaluation, CN-OA-13-7, "Wolf Creek ELAP Containment Environment Analysis," Revision 1, to evaluate transient containment environment during an ELAP. The evaluation used the Generation of Thermal Hydraulic Information in Containment (GOTHIC) thermal hydraulic computer program. The model was modified to disable active containment cooling features. The model was run out to 7 days.

The analysis determined the peak containment pressure during the 7 day transient is 16.5 psig and occurs at the end of day 7. The peak bulk containment temperature is 222°F also occurring at 7 days.

The NRC staff noted the analysis demonstrates the mitigating strategies maintain containment temperate and pressure remain below the design limits of 60 psig and 320°F, with margin.

3.4.4.3 FLEX Pumps and Water Supplies

The licensee's coping strategies do not require the use of FLEX pumps or water supplies to directly maintain containment integrity. Instead, they rely on the reduction of the RCS temperature to reduce the heat load into the containment thereby leading to a reduction of temperature and pressure. Section 3.2.3.5 of this safety evaluation evaluates the FLEX pumps and water supplies used for RCS heat removal.

3.4.4.4 Electrical Analyses

The licensee's Phase 1 coping strategy for containment includes monitoring containment temperature and pressure using installed instrumentation powered by the Class 1E station batteries. The licensee's strategy to repower instrumentation using the Class 1E station batteries is described in Section 3.2.3.6 of this safety evaluation and is adequate to ensure continued containment monitoring.

The licensee's Phase 2 coping strategy is to continue monitoring containment temperature and pressure using installed instrumentation. The licensee's electrical strategy is to repower instrumentation using a 480 Vac, 500 kW FLEX DG is described in Section 3.2.3.6 of this safety evaluation and is adequately sized to ensure continued containment monitoring.

The licensee's Phase 3 coping strategy is to continue the Phase 2 strategy using equipment supplied by an NSRC, if necessary. During Phase 3, the three NSRC supplied 1-MW, 4160 Vac CTGs would provide power to the required Phase 2 loads (instruments) plus Phase 3 loads (e.g., an RHR pump, CCW pump, etc.) for restoring containment cooling and to maintain containment pressure and temperature within the equipment design limits, if necessary. The licensee's electrical strategy to repower Phase 3 loads using three 1.0 MW 4160 Vac CTGs is described in Section 3.2.3.6 of this safety evaluation and the equipment is adequately sized to ensure continued containment monitoring and cooling.

Based on above, the NRC staff concludes that the licensee's electrical strategy is acceptable to restore or maintain containment indefinitely during an ELAP event.

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 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; severe storms with high winds; snow, ice and extreme cold; and extreme high temperatures.

References to external hazards within the licensee's mitigating strategies and this safety evaluation 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 pursuant to 10 CFR Part 50, Section 50.54(f) [Reference 22], 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 proposed rule, titled "Mitigation of Beyond-Design-Basis Events," hereafter called the MBDBE rule, which was published for comment in the Federal Register on November 13, 2015 [Reference 49]. The proposed 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 Reevaluation of Flooding Hazards" [Reference 46]. The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 23]. 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 37], 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 safety evaluation using the revised seismic and flooding hazards information and, if necessary, make changes to the strategies or equipment. Licensees will submit the MSAs for NRC staff review.

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 safety evaluation 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 seismic criteria for WCGS, which includes the SSE as specified in the USAR [Reference 50], Section 2.5. The maximum horizontal acceleration for the SSE is 0.20g. In its FIP, the licensee described that WCGS is part of the Standard Nuclear Unit Power Plant System (SNUPPS), which was developed with enveloping seismic loads for several sites; all the sites utilized Regulatory Guide 1.60 response spectra anchored at 0.20g Zero Period Acceleration (ZPA). All standard plant, safety-related, power block structures were designed and analyzed using this original value. These structures include:

- Reactor Building
- Auxiliary Building
- Control Building
- EDG Building
- Fuel Building
- safety-related tanks
- ESW vertical loop chase
- refueling water storage tank
- buried ESW piping
- safety-related duct banks

In its FIP, the licensee also described that for the remainder of the structures and structural components in the plant, WCGS evaluated a new free-field response spectrum that is enveloped by a Regulatory Guide 1.60 spectrum anchored at 0.15 ZPA. All the safety-related structures were deemed acceptable with regard to the new free-field response spectrum anchored at 0.15 ZPA. These structures include:

- ESW Pumphouse
- electrical manholes
- circulating and warming water pipe encasements
- UHS dam
- ESW caissons

The WCGS USAR, Section 2.5, states that an SSE corresponding to a horizontal acceleration anchored at 0.12g was selected as the design criteria for the facilities. For the purposes of this safety evaluation, the structures listed in this section of the safety evaluation are considered by the NRC staff to be robust for seismic considerations.

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 OIP, the licensee stated that the WCGS site is considered a "dry" site and that flooding hazards are not applicable to the WCGS site. However, the OIP also states that the site plans to develop FLEX strategies that address probable maximum precipitation flooding hazards. In its FIP, the licensee stated that the maximum design basis power block flood level from any cause corresponds to an elevation of 1099.92 feet mean sea level (MSL). This flood elevation is postulated to occur based upon an analysis of the LIP event. According to the licensee's FIP, the site has an elevation of 1099.5 feet MSL and the floor level of all safety-related structures is 1,100.0 feet MSL (USAR Section 2.4.2.3.2). In its FIP, the licensee determined that specific analysis of flood levels resulting from ocean storm front surges and tsunamis is not required because of the inland location of the plant. Seiches pose no flood threats because of the size and configuration of the cooling lake and that its topographic and geologic features are not conducive to landslide formation. The licensee's USAR, Section 3.4.1.1.1, indicates that safety-related systems located below grade are protected from groundwater in-leakage by a combination of a waterproofing system for the structures and other features such as watertight compartments, sump pumps, alarms and other water level indications and administrative controls. The waterproofing system is applied up to grade level and will serve to minimize groundwater in-leakage. Since the installed sump pumps described in the USAR may not be powered during an ELAP, the NRC staff notes that the licensee's plan includes provisions to deal with potential flooding of the Auxiliary Building 1974' level by keeping an RCS makeup pump stored at the 2000' elevation (ground level) to mitigate this possibility. This strategy would also protect against any groundwater intrusion.

The NRC staff concludes that the licensee has appropriately considered this external hazard in its FLEX strategies and has properly identified the hazard levels to be evaluated.

3.5.3 High Winds

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 miles per hour (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. Based on a review of NEI 12-06, figure 7-1, the staff concludes that hurricane hazards screen out for WCGS.

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, Revision 2, February 2007; if the recommended tornado design wind speed for a 1E-6/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, Revision 1.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee described that per NEI 12-06, the site has the potential for damaging winds caused by a tornado exceeding 130 mph. In addition, NEI 12-06 indicates a maximum wind speed of 200 mph for Region 1 plants, including WCGS.

Therefore, high-wind hazards are applicable to the plant site. The licensee has appropriately screened in the high wind hazard and characterized the hazard in terms of wind velocities and wind-borne missiles.

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 should address the impact of ice storms.

In its FIP, the licensee stated that the region averages about 10 and 20 inches of snow a year. The extreme 24 hour snowfall was 26 inches and the snow usually only remains on the ground for a few weeks. An accumulation of 0.25 inches of ice due to ice storms will occur once a year and an accumulation of 0.50 inches of ice every 2 years. The mean duration of glaze ice on utility wires, if an ice storm occurs, is 53 hours for the state of Kansas.

In its FIP, the licensee also stated that temperatures in the site region are below freezing approximately 120 days per year. The lowest temperature recorded in the site region (Burlington, Kansas) was minus 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 low temperature 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 temperatures in the site region (Burlington Kansas) exceed 90°F approximately 60 to 70 days per year (USAR Section 2.3.2.1.1). The peak temperature recorded in Burlington was 117°F.

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 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 described that FLEX equipment, with the exception of the FLEX RCS makeup pumps, is stored in two 7,200 square foot FSBs constructed to American Society of Civil Engineers (ASCE) standard ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures," to meet the requirements of NEI 12-06 for seismic, flooding, high winds, snow, ice, extreme cold and high temperature for the protection of the FLEX equipment. According to the licensee, each of the FSBs has a common set of equipment capable of supporting each FLEX function/strategy such that the "N+1" provision of NEI 12-06 is met. In its FIP, the licensee indicated that the FLEX RCS makeup pumps are stored in the Auxiliary Building and are protected against all BDBEE hazards.

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

3.6.1.1 Seismic

In its FIP, the licensee stated that the SSE was utilized as the input for the FSBs' seismic design requirements. The FSBs were evaluated for the effect of local seismic ground motions consistent with the WCGS ground motion response spectra developed for the site as a result of the seismic hazard reevaluation and found to have adequate structural margin to remain functional (i.e., collapse is not expected and access to the interior retained).

In its FIP, the licensee indicated that analyses of components stored in the FSBs have been performed to determine appropriate measures to prevent seismic interaction. All equipment stored in the buildings is restrained. The fire protection and heating and ventilation systems in the FSBs are seismically installed. The lighting, conduits, electrical, and fire detection components are not seismically installed, but are considered insignificant and not able to damage FLEX equipment.

3.6.1.2 Flooding

In its FIP, the licensee stated that the primary FSB is located in in the southeast part of the site. The alternate FSB is located to the west of the station blackout (SBO) DG building. These locations are above the flood elevation from the most recent site flood analysis.

3.6.1.3 High Winds

In its FIP, the licensee described that the FLEX equipment is stored in two FSBs constructed to ASCE 7-10, to meet the requirements of NEI 12-06 for high winds for the protection of the FLEX equipment. The FIP does not address whether the FSBs can withstand the impact of potential missiles generated by a high wind event. During the audit process, the NRC staff reviewed the protection provided to the licensee's FLEX equipment with regard to high wind missiles. Specifically, the staff reviewed the licensee's document "Study of the Adequacy of the FLEX Buildings' Separation Considering Severe Storms and High Wind Conditions," Revision 1. The staff noted that the licensee's study considered site-specific building separation considerations, as well as the predominant tornado path in establishing the axis of separation for the buildings. In addition, the licensee's study considered intervening missile-protected structures that could provide a degree of protection. Thus, the staff concludes that provisions of NEI 12-06, Section 7.3.1 are met regarding tornado missile protection via diverse locations for the two WCGS FSBs. Therefore, since the licensee has two full sets of equipment in each building, at least one set ("N") should be deployable after a tornado event.

During the audit process, the staff also noted that the debris removal equipment was an exception to the licensee's plan for protected storage of two separated sets of equipment. Specifically, the licensee had provisions for storing only one set of debris removal equipment in the primary FSB. Thus, if a tornado were to disable the primary FSB, along with its stored debris removal equipment, the licensee would not be able to deploy "N" sets of equipment from the other building, contrary to the provisions of NEI 12-06. Therefore, the licensee implemented a procedure change such that if a high wind warning is posted for the site, the licensee would move additional debris removal equipment into the alternate FSB, thus restoring capability to deploy "N" sets of equipment, consistent with the diverse location high wind missile provisions of NEI 12-06. The NRC staff confirmed this procedural direction had been implemented by reviewing procedure AI 14-006, "Severe Weather," Revision 17, during the audit process. Based on the licensee's building design and procedural provisions for debris removal equipment, the staff finds the storage plan acceptable for high winds and associated missiles.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

In its FIP, the licensee stated that the FSBs are designed to withstand extreme temperatures by using an installed ventilation system. The ventilation system and individual space heaters will maintain temperatures in the buildings about 50°F.

3.6.1.5 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 a single resource that is sized to support the required functions for multiple units at a site (e.g., a single pump capable of all water supply functions for a dual unit site). In this case, the "N+1" could simply involve a second pump of equivalent capability. In addition, 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.

In its FIP, the licensee stated that sufficient equipment has been purchased to address all functions on-site, plus one additional spare, i.e., an "N+1" capability, where "N" is the number of equipment required by FLEX strategies. Therefore, where a single resource is sized to support the required function, a second resource has been purchased to meet the "N+1" capability. In addition, where multiple strategies to accomplish a function have been developed, (e.g., two separate means to repower instrumentation) the equipment associated with each strategy does not require "N+1" capability. In its FIP, the licensee provided Table 1, "FLEX Portable Equipment Stored On-Site," that lists some of the portable equipment stored on-site.

In its FIP, the licensee described its strategy for spare hoses and cables. The licensee states that hoses and cables are passive components being stored in a protected facility. It is postulated the most probable cause for degradation/damage of these components would occur during deployment of the equipment. Therefore the "N+1" capability is accomplished by having sufficient hoses and cables to satisfy the "N" capability plus an additional 10 percent length of spares or at least 1 length of hose and cable. The licensee states that this margin capability ensures that failure of any one of these passive components would not prevent the successful deployment of a FLEX strategy. The licensee's FIP describes this storage feature as an "alternate approach".

After the issuance of NEI 12-06, Revision 0, NEI, on behalf of the commercial nuclear power industry, submitted a letter to the NRC [Reference 47] proposing an alternative regarding the quantity of spare hoses and cables to be stored on site. The alternative proposed was that either: (a) 10 percent additional lengths of each type and size of hoses and cabling necessary for the "N" capability plus at least one spare of the longest single section/length of hose and cable be provided, or (b) that spare cabling and hose of sufficient length and sizing to replace the single longest run needed to support any FLEX strategy. By letter dated May 18, 2015 [Reference 48], the NRC agreed that the alternative approach is reasonable, but that the licensees may need to provide additional justification regarding the acceptability of various cable and hose lengths with respect to voltage drops, and fluid flow resistance. The NEI alternative for spare hoses and cables was later incorporated into the NRC-endorsed NEI 12-06, Revision 2. The licensee has elected to store the majority of their FLEX equipment in two separate storage buildings. For non-hurricane sites such as WCGS, this concept provides protection from high wind missiles via separation, in that the equipment in at least one building should remain deployable because a tornado is not postulated to impact both structures. Thus, in order to ensure that sufficient spare hoses and cables are available, after accounting for the potential of damage during deployment, there would need to be appropriate spares in each storage building. According to the licensee's FIP, Table 1, each FSB contains the "associated" hoses and cables.

Since the licensee's description of spare hoses and cables appears to be consistent with the provisions of NEI 12-06, Revision 2, and the FIP states that the licensee is following NEI 12-06, Revision 2, the NRC staff concludes that the quantity of spare hoses and cables for FLEX at WCGS is acceptable, as long as there are sufficient spares stored in each of the two FSBs, or in another fully protected structure at the site.

Based on the FIP description and audit review, the NRC staff finds that, if implemented appropriately, the licensee's supply of portable FLEX pumps, FLEX DGs, and support equipment are consistent with the "N+1" provision in Section 3.2.2.16 of NEI 12-06. The staff notes that the RCS makeup pumps and connections have event-based clarifications to this provision, as described in Section 3.2.3.1.1 of this safety evaluation.

3.7 Planned Deployment of FLEX Equipment

In its FIP, the licensee described that pre-determined, preferred haul paths have been identified and documented in FSG-05. The licensee also described that the deployment of onsite FLEX equipment in Phase 2 requires that pathways between the FSBs and various deployment locations be clear of debris resulting from BDB seismic, high wind (tornado), or flooding events. A front end loader is available to deal with more significant debris conditions and is stored inside the primary FSB. According to the licensee, this is protected from the severe storm and high wind hazards such that it remains functional and deployable to clear obstructions from the pathway between the FSB and its deployment location(s). In its FIP, the licensee described that the deployment of the Phase 2 FLEX equipment will not require external power because the building equipment doors can be opened manually. The NRC staff concludes that deployment should be feasible, given the FIP description and the licensee's procedural direction for pre-positioning supplemental debris removal equipment as described in Section 3.6.1.3 of this safety evaluation.

3.7.1 Means of Deployment

In its FIP, Table 1, the licensee listed two tow vehicles, with one being stored in each FSB. In addition, the debris removal vehicle stored in the primary FSB could also be used as a tow vehicle. In addition to debris removal, the NRC staff reviewed the strategy that will be implemented by the licensee to clear the roads of snow and ice to allow for the movement of FLEX equipment during the audit process. The licensee has included provisions for snow and ice removal of the FLEX-associated outdoor areas and haul paths into guideline FSG-05. The staff observed during the audit process that the debris removal equipment could be employed for snow and ice removal.

3.7.2 Deployment Strategies

In its FIP, the licensee described that the haul paths have been reviewed for potential soil liquefaction and have been determined to be stable following a seismic event. Additionally, the preferred haul paths minimize travel through areas with trees, power lines, narrow passages, etc. to the extent practical.

In its FIP, the licensee indicated that severe low temperatures could adversely affect access to and the flowpath from the cooling lake. Ice could form on the surface of the cooling lake and impact FLEX strategies. However, capabilities are available to break through the ice, if needed, to provide access and a flowpath.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

Core Cooling (SG) Primary and Alternate Connections

In its FIP, the licensee described the primary connections for SG makeup. The suction of the FLEX core cooling pump is connected to a CST isolation valve using two sections of non-collapsible hoses and gated wye adapters. The discharge from the FLEX core cooling pump is connected to a flexible hose, which is routed to an intermediate hard pipe, which is in turn connected by a second section of hose to the "Emergency Water to TDAFWP Discharge Line Hose" connection on the 2000' elevation of the Auxiliary Building. The alternate flowpath on the suction side of the FLEX core cooling pump utilizes a direct connection to the CST makeup pump discharge. The CST makeup pump is placed into one of the ESW bays inside of the ESW Pumphouse and the attached hose is connected into a wye adapter outside of the ESW Pumphouse. Additional hoses are routed from the wye adapter to either a CST isolation valve (to fill the CST) or directly to the suction of the FLEX core cooling pump. The alternate discharge path is described as a flexible hose connected from the discharge of the FLEX core cooling pump to the same intermediate hard pipe as the primary strategy, with a second flexible hose connected to the "Emergency Water to Motor Driven Auxiliary Feedwater (MDAFW) Pump "B" Discharge Line Hose" connection on the 2000' elevation of the Auxiliary Building. The CST suction connection and the upstream intermediate pipe connection are located in the CST pipe chase which, according to the licensee's FIP, has been upgraded to be seismically robust and high wind missile protected. The discharge FLEX connections for the second hoses in both primary and alternate strategies are inside the Auxiliary Building and are thus protected from all applicable external hazards.

RCS Inventory Control Primary and Alternate Connections

The primary suction flow path for the FLEX RCS makeup pump on the 1974' elevation of the Auxiliary Building is from one of the two BAT tanks located on the 1974' elevation of the Auxiliary Building. A non-collapsible suction hose is connected between a BAT isolation valve connection and the pump suction. The primary discharge flow path for this pump connects to a high pressure coolant injection system isolation valve on the 2000' elevation of the Auxiliary Building via a two flexible hoses and a FLEX intermediate pipe (hard pipe from the 1974' elevation to 2000' elevation of the Auxiliary Building).

The alternate suction flow path for the FLEX RCS makeup pump on the 1974' elevation of the Auxiliary Building is from the RWST. A non-collapsible suction hose is connected between a minimum flow line connection on RHR train "A" and the pump suction. The alternate discharge flow path uses a hose from the pump discharge to a high pressure coolant injection system BIT inlet isolation valve connection on the 1974' elevation of the Auxiliary Building.

The suction flow path for the FLEX RCS makeup pump on the 2000' elevation of the Auxiliary Building is from the RWST. A non-collapsible suction hose is connected between a RWST isolation valve connection in the RWST Building and a gated wye. A second hose is connected between the gated wye and the pump suction. The discharge flow path uses a hose from the pump discharge to a high pressure coolant injection system isolation valve on the 2000' elevation of the Auxiliary Building.

All of the RCS inventory connections except for the RWST suction to the 2000' elevation RCS makeup pump are located in the Auxiliary Building are thus protected from all hazards.

<u>SFP Makeup Primary and Alternate Connections</u>

In its FIP, the licensee described the primary strategy for SFP makeup. The RWST is the primary makeup source. Two series non-collapsible hoses are connected from a RWST isolation valve to the suction of the FLEX SFP pump, which is staged near the RWST. A gated wye adapter is connected between the two suction hoses. A hose is then connected from the pump discharge and routed to a connection point located near the Fuel Building rollup door. This connection point is hard-piped to spray nozzles on the 2047' elevation of the Fuel Building. The alternate SFP strategy uses the CST as a suction source. In this case the FLEX SFP pump is staged near the CST. Two non-collapsible hoses and a gated wye adapter are routed between a CST isolation valve and the pump suction. Flexible hose is then connected to an alternate SFP supply connection just inside the Fuel Building rollup door. This connection is hard-piped to the 2026' elevation of the Fuel Building. A second flexible discharge hose is routed from the hard pipe to a valve connection in the fuel pool cooling and cleanup system. This valve connection will allow manual valves to be opened for discharging the CST water directly into the "A" SFP cooling train. The SFP connections in the Fuel Building are protected from all applicable external hazards.

<u>ARV Accumulators Primary and Alternate Connections</u>

In its FIP, the licensee described the primary connection for the SG ARV accumulators. Two electric FLEX air compressors are stored on the Auxiliary Building 1974' elevation. A hose is routed from one of the electrical FLEX air compressor receiver outlet connections to a manifold located on the 2000' elevation of the Auxiliary Building. Hoses are then run from the manifold to an adapter on the ARV accumulators. The alternate strategy for the ARV accumulators utilizes FLEX diesel-powered air compressors stored in the FSBs. One of these compressors would be staged at the east Turbine Building rollup door. A section of hose is connected between a compressor outlet connection and the ARV accumulators through a manifold in the Turbine Building. The air supply connections for both the primary and alternate methods are located in the AFW pump vestibule on the Auxiliary Building 2000' elevation, and are protected from all applicable external hazards.

3.7.3.2 Electrical Connection Points

The licensee's electrical strategy uses connection points to support Phases 2 and 3 of the overall strategy. For Phase 2, the licensee has developed primary and alternate strategies for supplying power to the necessary equipment using a combination of permanently installed and portable components. The primary strategy for repowering the Class 1E battery chargers is to deploy one of the two 480 Vac FLEX DGs and the cable trailer, stored in the FSB, to the staging area near the Control Building 2000' elevation southwest door. Operators would then run color-coded cables from the 480 Vac FLEX DG to the electrical connection panel located on the west wall of Engineered Safety Feature (ESF) Switchgear Room NB02. After the cable connections are complete, operators would align breakers and disconnect switches from the FLEX DG to the selected Class 1E 480 Vac load center to restore power to the Class 1E battery chargers. The electrical connection points for this strategy are located in the Control Building, which is a safety-related Seismic Category I structure that is designed to protect equipment during design-basis events. As such, electrical connections are protected from all site applicable external hazards, and are therefore robust.

The licensee's alternate Phase 2 strategy for supplying power to the Class 1E battery chargers is to deploy a 480 Vac FLEX DG and cable trailer to the staging area near the Auxiliary Building southwest corridor door. Operators would then run color-coded cables to a connection panel located in the Auxiliary Building southwest corridor. Operators would align breakers to restore power to the selected 480 Vac Class 1E load centers to provide power to the Class 1E battery chargers. The electrical connection points for this strategy are located in the Auxiliary Building which is a safety-related Seismic Category I structure that will protect electrical equipment and connections from design basis events. As such, electrical connections for the alternate strategy are protected from all site applicable external hazards.

During the audit, the staff questioned how proper phase rotation for the Phase 2 480 Vac FLEX DGs would be ensured. The staff confirmed that the proper phase rotation was determined during the work order process that established the connection points. Therefore, by following the color coding scheme referred to above, proper FLEX equipment rotation would thus be ensured.

For Phase 3, three 4160 Vac CTGs will be deployed outside the west wall of the Control Building and will be connected to a distribution panel (provided from the NSRC). Primary and alternate strategies will allow the NSRC supplied 4160 Vac CTGs to supply power to one of the Class 1E 4160 Vac buses (NB01 or NB02) to power the RHR and CCW pumps. Operators would remove cables from a breaker on one of the Class 1E 4160 Vac buses and then connect cables from the distribution panel to the selected Class 1E 4160 Vac breaker. The staff reviewed guideline FSG-15, "Deployment of NSRC FLEX Generators," Revision 0, which provides guidance for connecting the Phase 3 primary and alternate electrical strategies and includes guidance for verifying proper phase rotation.

Based on the licensee's FIP, and confirmed by a review of single line electrical diagrams and station procedures, the NRC staff concludes that the licensee's approach is acceptable given the protection and diversity of the power supply pathways, the separation and isolation of the FLEX DGs from the Class 1E EDGs, and availability of procedures to direct operators how to align, connect, and protect associated systems and components.

3.7.4 Accessibility and Lighting

In its FIP, the licensee described that following a BDBEE and subsequent ELAP event, FLEX coping strategies require the routing of hoses and cables through various doors and gates in order to connect FLEX equipment to station fluid and electric systems. The ability to open doors for ingress and egress, ventilation, or temporary cables/hoses routing is necessary to implement the FLEX coping strategies. According to the licensee, operators responsible for implementing FLEX strategies have immediate access to keys that provide access to all required areas.

In its FIP, the licensee described that MCR lighting is powered by plant batteries and adequate portable lighting is provided to support activities outside of the MCR. The FIP also indicates that lighting is included on the 500 kW DG trailers to assist with illumination at the DG staging area. The licensee's FIP includes provisions for eight diesel-powered light towers (four per storage building) that could be available to support FLEX deployment. In addition, the FSBs include a stock of flashlights and headband lights to further assist the staff responding to an ELAP event during low light conditions. During the audit process, the NRC staff reviewed the licensee's program document, AP-21A-002, "Diverse and Flexible Coping Mitigating Strategies (FLEX) Program," Revision 0. This document says that small portable generators with an attached lighting kit and/or trailer mounted lights poles will be set up. Further, FSG-05 contains provisions for setting up portable lighting to illuminate the staging areas and the pathway to the FLEX connections. In addition, the NRC staff notes that operations procedure AP 21-001, "Conduct of Operations," Revision 77, states that operators should carry flashlights.

According to the licensee's FIP, most areas important to the FLEX strategy contain fire protection program ("Appendix R") lighting as well. These emergency lights will provide a minimum of 8 hours of lighting with no external ac power sources, thus providing lighting for pathways to implement the Phase 1 FLEX strategies. According to the licensee's USAR, Section 9.5.3.2.3, this lighting is installed to survive a design-basis event, which for WCGS would include a seismic or high wind event. During the audit process, the staff confirmed the seismic capability of the Appendix R lighting by reviewing licensee specification E-168900, "Lighting Notes, Symbols, and Details," Revision 78. Based on this review, the staff concludes that the "Appendix R" lighting, when located in protected structures, would be considered to be "robust" in accordance with the provisions of NEI 12-06.

Given these available options, the NRC staff concludes that the licensee has made adequate provisions for lighting under the postulated ELAP conditions.

3.7.5 Access to Protected and Vital Areas

In its FIP, the licensee provided information describing that access to the owner controlled area, site protected area, and areas within plant structures will not be hindered under conditions where there is a loss of power. The licensee has contingencies in place that include provisions such as keyed access, manual controls, and administrative measures to provide access to areas required for the ELAP response if the normal access control systems are without power. During the audit process, the NRC staff confirmed that the licensee's procedure, AP 21-001, includes provisions for operators to have the ability to obtain the necessary keys that allow access to applicable operational areas.

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee described that FLEX equipment is stored in the fueled condition. Refueling of FLEX equipment commences in the first 24 hours or sooner if needed. The general coping strategy for supplying fuel oil to diesel-driven portable equipment being used to cope with an ELAP/loss of normal access to the UHS event is to draw fuel oil out of any available existing diesel fuel oil tanks on site. The following sources of fuel could be utilized to refuel the FLEX equipment via the fuel transfer truck (or portable containers) as required.

- Four (4) 10,000-gallon underground tanks (3 diesel fuel and 1 gasoline) at the vehicle maintenance shop (non-safety)
- Two (2) 600-gallon, seismically qualified EDG day tanks
- Two (2) 100,000-gallon, seismically qualified EDG 7-day storage tanks (Technical Specification minimum 74,200 gallons/tank)
- One (1) 469,000-gallon auxiliary boiler fuel tank (above ground and non-seismic)

In addition, the licensee described in its FIP that they have four dc fuel transfer pumps that can be used to move fuel from any of their on-site sources and into either jerry cans or their existing fuel truck. Each pump can provide 8-10 gpm with a 30-minute on/off duty cycle, which means that the pumps would need to be rotated into and out of service after providing 240 to 300 gallons of fuel to the jerry cans or fuel truck. The fuel can then be transported to the FLEX equipment requiring refueling.

The licensee also stated in its FIP, that diesel fuel in the fuel oil storage tanks is routinely sampled and tested to assure fuel oil quality is maintained to ASTM [American Society for Testing and Materials] standards. This sampling and testing surveillance program also assures the fuel oil quality is maintained for operation of the station EDGs. Portable equipment powered by diesel fuel is designed to use the same low sulfur diesel fuel oil as the installed EDGs.

The licensee's FIP describes a preventative maintenance (PM) program for the FLEX equipment, including fluid analysis for portable diesel pumps and generators. During the audit process, the licensee clarified that guidance for the PM program will include the fuel quality sampling based on Electric Power Research Institute (EPRI) templates. Specifically, the fuel oil will be sampled annually for water, viscosity, and cloud point for functionality. The NRC staff found that this approach was acceptable.

In its FIP, the licensee described that they calculated the maximum fuel usage to be approximately 80 gallons/hour and determined that even with a fuel consumption rate of 880 gallons/hour, the site could cope for at least 7-days without the need for offsite fuel using only the volume in the EDG day tanks and EDG 7-day tanks.

The licensee also indicated in its FIP that provisions for receipt of diesel fuel from offsite sources are in place to facilitate the Phase 3 re-powering strategy with the portable 4160 Vac CTGs.

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 WCGS SAFER Plan

The commercial nuclear power industry has collectively established the needed off-site capabilities to support FLEX Phase 3 equipment needs via the SAFER Team. The SAFER team consists of the Pooled Equipment Inventory Company (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 NSRCs, located near Memphis, Tennessee and Phoenix, Arizona, established 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 the NSRC.

By letter dated September 26, 2014 [Reference 24], 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 reviewed AREVA Document Number 38-9247053-00, "SAFER Response Plan for Wolf Creek Generating Station," Revision 2, dated September 10, 2015, during the audit process. During this review the 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. The staff also noted that the plan contains provisions for helicopter support, if it is required.

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 25-35 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. The WCGS SAFER Response Plan does not have provisions for Staging Area "D". Staging Area "C" is identified as the "BETO" [Burlington, Emporia, Topeka, Ottawa], which is a location near the intersection of U.S. Interstate 35 and U.S. Highway 75, about 16 miles from the WCGS site. Staging Area "B" is on the north side of the site, within the owner controlled area. Staging Area "A" (multiple locations) is the point-of-use location for the FLEX response equipment. Even though the licensee's Staging Area "C" is less than 25 miles from the site, based on the general topography in the area the NRC staff concludes that the licensee's designation of a Staging Area "C" without an alternate Staging Area "D" is acceptable.

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 event with a subsequent loss of power, ventilation that provides cooling to occupied areas and areas containing mitigation equipment will be lost. The licensee performed and the staff reviewed WCGS calculation XX-M-090, "Post-ELAP Room Temperature Evaluation," Revision 0, to quantify maximum steady state temperatures following a loss of ventilation during an ELAP for certain key areas to ensure the environmental conditions remain acceptable for personnel habitability and within equipment qualification limits.

The key areas identified for all phases of execution of the FLEX strategy activities are the:

- MCR
- NK Battery Rooms
- DC Switchboard Rooms
- ESF Switchgear Rooms
- Auxiliary Building 1974' and 2000' Elevations
- Fuel Building general area
- TDAFW Pump Room
- TDAFW Pump Discharge Valve Nitrogen Accumulator Rooms
- Boric Acid Tank and Transfer Rooms.
- Containment

The licensee considered the results to be conservative, as many of the temperatures are calculated utilizing internal room heat loads that represent equipment operating either normally or in accident conditions. During an ELAP, the majority of this equipment will be non-functional and therefore will not generate heat.

Main Control Room

For the MCR, the licensee performed evaluation DE-17-001, "Post-ELAP Main Control Room Temperature," Revision 0. In this evaluation, the licensee determined that the MCR temperature would reach a steady-state temperature of 115°F assuming certain Control Building doors were opened, but with no forced ventilation. The evaluation assumes a constant outdoor temperature of 97°F and 10 persons in the MCR. In its FIP, the licensee stated that procedure EMG C-0 directs all MCR equipment cabinets and several Control Building doors to be opened within 30 minutes after a loss of all ac power to assure vital instrument cooling. In addition, guideline FSG-04 provides direction to implement a ventilation flow path through the MCR. Guideline FSG-05 provides instructions to the plant operators to install a 15 kW portable DG at Staging Area "A3" to energize four high flow electrical fans inside the Control Building to create a forced air flow path through the MCR with a projected completion time of 8 hours. The licensee's calculation concludes that when using the FLEX fans along with the opening of doors, steady state MCR temperature would be below 102°F.

The NRC staff reviewed the licensee's calculation during the audit process and determined that its methodology, assumptions, and results were reasonable. Based on the projected MCR 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 equipment to be able to survive indefinitely), the NRC staff finds that the electronic equipment and components in the MCR should not be adversely impacted by the loss of ventilation as a result of an ELAP.

NK Battery Rooms

The licensee's calculation XX-M-090 evaluated the battery rooms. The method chosen only allows for the energy to be absorbed by the air and is therefore conservative because energy absorbed by equipment cabinets and walls has been ignored. The calculation assumes a constant outdoor ambient temperature of 97°F. Compensatory actions such as opening doors were not modeled. A computer program was used to model the room heat-up for a duration of 4 hours. The calculation indicated room temperatures reached a steady state temperature by 1 hour. The calculation shows Battery Room 4 (Room 3405) reaches a temperature of 108.4°F and Battery Room 2 (Room 3411) reaches a temperature of 108.3°F.

The battery manufacturer (AT&T) recommends that the battery room temperature should not exceed 122°F (for short durations) and that the batteries should not exceed 110°F without periodic monitoring of the electrolyte level. Guideline FSG-04 provides instruction to the plant operators to monitor battery room temperature. If battery room temperature is greater than 60°F, plant operators would open selected battery room doors (to release hydrogen buildup, as discussed in Section 3.9.1.3 of this safety evaluation) that will allow hot air to flow out of the battery rooms, resulting in lower temperature than predicted in the calculation XX-M-090. The licensee will also establish forced ventilation using FLEX fans as directed by FSG-05. The mitigating actions in procedures FSG-04 and FSG-05 should maintain room temperature at or below the analyzed temperature (108°F).

Based on the evaluation result of a steady state 108°F temperature, combined with provisions for establishing portable ventilation, the NRC staff concludes that the licensee's ventilation strategy should maintain the battery room temperature below the maximum temperature limit specified by the manufacturer, and therefore the batteries should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

DC Switchboard Rooms and ESF Switchgear Rooms

Similar to the battery rooms, calculation XX-M-090 also evaluates the dc switchboard rooms and the ESF switchgear rooms. The calculation indicated room temperatures reached a steady state temperature by 1 hour. The calculation predicts Switchboard Room 4 (Room 3404) reaches a steady state temperature of 121°F. Switchboard Room 2 (Room 3410) reaches a steady state temperature of 123.8°F. Similarly, the calculation indicates the Switchgear Room 2 (Room 3302) reaches a steady state temperature of 113.9°F. The staff notes that FSG-04 (opening selected Control Building doors) and FSG-05 (using four temporary fans within 8 hours into the event and powered by a 15 kW DG to blow outside fresh air into the rooms), the expected room temperatures would be lower than predicted in the calculation.

NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, specifies 120°F as the temperature below which electronic equipment is likely to survive indefinitely. Even though the steady state temperatures in some of these rooms are calculated to exceed 120°F by a small amount, the NRC staff concludes that the mitigating actions in guidelines FSG-04 and FSG-05 which are not reflected in the calculation are likely to lower temperatures such that the equipment in the vital dc switchboard rooms and ESF switchgear rooms should remain available during an ELAP.

Auxiliary Building 1974' and 2000' Elevations

During the audit process, the NRC staff reviewed WCGS Basic Engineering Disposition, "Plant Room Temperature Post-ELAP," dated 09/20/2016. According to this document, which references design input EQSD-1, "Equipment Qualification Design Basis Document," Revision 10, Attachment B, Note 2, the temperature in the Auxiliary Building general areas is assessed as follows:

> With the exception of the RHR heat exchanger rooms, the auxiliary feedwater turbine-driven pump room, and the main steam/main feedwater isolation valve rooms, the ambient temperature outside of the containment in rooms and corridors which do not have ESF coolers will not exceed 120°F during loss of normal ventilation conditions, because of the lack of heat sources.

According to the licensee's FIP, the Auxiliary Building general areas do not have ESF coolers; therefore, the 120°F maximum temperature applies. The licensee further concludes that no compensatory measures would be needed in these areas. Based on the licensee's FIP statements, the NRC staff concludes that any electronic equipment in the Auxiliary Building general areas should remain available during an ELAP.

Fuel Building General Area

During the audit process, the NRC staff reviewed WCGS Basic Engineering Disposition, "Plant Room Temperature Post-ELAP," dated September 20, 2016. According to this document, the temperature in the Fuel Building general area post-ELAP is calculated in FD-13-002, "Fuel Building Habitability Analysis," Revision 0. According to the licensee's FIP, this considers the scenarios of normal and full core offload in the SFP, in conjunction with two scenarios of when the rollup door may be opened during the FLEX Integrated Plan.

Under normal offload (plant operating) conditions, Fuel Building temperature remains below 105°F until 5.59 hours post-ELAP, at which time the SFP boils, causing ambient temperature to jump over 140°F and slowly climb to 152°F at 35 hours post-ELAP. Under full core offload conditions, temperature remains under 105°F for approximately 2.4 hours, at which point the SFP boils, the ambient temperature jumps to over 160°F and continues to climb. In conjunction with full core offload, if the Fuel Building rollup door is not opened until 5.59 hours post-ELAP, the internal Fuel Building temperature may reach 212°F around 5 hours post-ELAP.

To compensate, FSG-05 directs operators to perform time-sensitive actions as early as possible during the ELAP but no later than 5 hours and 35 minutes during Modes 1-5, and 2 hours and 23 minutes if defueled, due to the likelihood of high temperature and humidity in the Fuel Building. These actions include directing operators to block open selected doors to establish a vent pathway, and perform deployment of the SFP spray equipment in the Fuel Building. The performance of this guideline within these time periods will ensure the SFP spray strategy is ready for initiation before conditions threaten habitability. The cooling effects of opening of the Fuel Building doors per FSG-05 are not considered in the temperature analysis, therefore its predicted temperatures for the Fuel Building will likely remain below the predicted temperature response calculated in FD-13-002.

TDAFW Pump Room

The licensee performed calculation GF-M-002, "Room 1331 Temperature Profile (Turbine Driven Aux Feed Pump room)," and determined that the TDAFW pump room has a steady-state temperature of 146.4°F that remains applicable for a 7-day ELAP, assuming no ventilation, doors closed, no lighting, and the TDAFW pump operating. The staff reviewed this calculation during the audit process. The staff also noted that procedure EMG C-0 directs doors for the TDAFW pump room and AFW corridor to be opened early during a loss of all ac power. This action will provide a path for natural convection through the TDAFW pump room, and is not accounted for within calculation GF-M-002.

Wolf Creek USAR, Table 3.11(B)2 shows that the design-basis temperature limit is 150°F for the TDAFW Pump room, which bounds the expected room temperature. Calculation GF-M-002 also shows that the maximum equipment qualification temperature is 150°F. Procedure EMG-C-0 provides guidance to block open doors for the TDAFW Pump room and Auxiliary Feedwater corridor at approximately 1.5 hours from the onset of an ELAP. This action will provide an air flow path for natural convection through the TDAFW Pump room. Furthermore, the licensee relies on the TDAFW pump only for the first 10 hours after initiation of an ELAP event. After that, the back-up FLEX diesel driven core cooling pump (to be staged at approximately 8 hours) can provide makeup water to the SGs if necessary.

Based on the expected steady-state room temperature (146.4 °F) remaining below the TDAFW Pump room design temperature limit (150°F), the NRC staff finds that the electrical equipment should remain functional during an ELAP event.

Main Steam/Feedwater Tunnel

The FIP states that calculation XX-M-090 determined the Main Steam Tunnel will reach a temperature of 220°F during the 7-day ELAP. In the event of a loss of dc power and nitrogen gas to the ARVs, operators may need to momentarily enter the Main Steam Tunnel to manually adjust the ARVs. During the audit process, the staff noted that the licensee has provisions established for entering hot areas such as this and also notes that the licensee's controlling procedure, EMG C-0, has a caution addressing the necessary personnel protection if a high temperature entry is necessary to adjust the ARVs. Given the capability to manually operate the valves and the licensee's administrative controls for entering a hot environment to make adjustments, if needed, the staff concludes that the equipment in this area will support the licensee's strategy.

Containment

The NRC staff reviewed calculation CN-OA-13-7, "Wolf Creek ELAP Containment Heat-Up," Revision 1, which the licensee performed to model the transient temperature response in the containment following an ELAP event. The calculation analyzed the containment pressure and temperature response for 7 days following an ELAP. The results of this analysis determined that the containment design limits for pressure (60 psig) and temperature (306.1°F) are not challenged during this period. Based on the licensee's evaluation, both pressure and temperature will remain below the limits for electrical components being credited as part of the licensee's mitigating strategies during an ELAP. Thus, actions to reduce containment temperature and pressure are not predicted to be required in Phase 1 and 2. However, in the FIP, the licensee stated that containment temperature will be procedurally monitored and, if necessary, the temperature will be reduced to ensure that key containment instruments will remain within their analyzed limits for equipment qualification. During the audit process, the staff confirmed that licensee procedure EMG C-0 has steps to monitor containment temperature and pressure. Procedure EMG C-0 also has provisions to implement alternate monitoring of these parameters if they are not available in the MCR. During Phase 3, the licensee will have the ability to restore the CCW, ESW, and RHR systems to support containment cooling using NSRC supplied 4160 Vac CTGs, as discussed in Section 3.4.4.4 of this safety evaluation, to reduce or maintain temperatures and support equipment functionality. During the audit, the staff confirmed that the licensee's procedures and guidelines have steps to consult plant staff regarding further RCS cooldown, once the necessary support equipment is available. Based on containment temperature and pressure remaining below their respective design limit, the licensee's strategy to monitor containment temperature, and the availability of Phase 3 resources, the NRC staff expects that the necessary equipment, including credited instruments, located inside containment should remain functional throughout an ELAP event.

Based on its review of the essential electrical equipment required to support the licensee's FLEX mitigation strategy, which are located in the MCR, NK battery rooms, vital dc switchboard rooms and ESF switchgear rooms, TDAFW Pump room and containment, the NRC staff concludes that the electrical equipment relied on to mitigate the analyzed ELAP event should remain functional indefinitely.

3.9.1.2 Loss of Heating

The licensee indicated in its FIP that strategies involving FLEX equipment or existing plant equipment would not require additional heat tracing systems. FLEX connections used for hoses or piping that will involve water will be located inside buildings. Portable heaters will be stored in the FSBs and deployed with temporary enclosures to provide freeze protections. Hoses routed outside from FLEX pumps will consist of steady flow to prevent freezing inside of the exposed portion of the hoses. Cold weather packages will be deployed with FLEX equipment to be deployed outdoors. The CST is insulated and adequately protected from extremes of hot and cold weather. The UHS would remain available as a water source during extreme cold conditions. The operators will be equipped with tools to break through the top layer of ice of the UHS to allow deployment of the CST makeup submersible pump into the ESW forebay.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern applicable to Phase 2 and Phase 3 is the potential buildup of hydrogen in the NK battery rooms. Off-gassing of hydrogen from batteries is only a concern when batteries are charging. Once the 480 Vac FLEX DG or NSRC CTGs restore power to the battery chargers to begin recharging the Class 1E batteries, the FLEX forced ventilation arrangement described in guideline FSG-05 will preclude any significant hydrogen accumulation. Guideline FSG-05 directs operators to stage one portable 15 kW FLEX DG and four FLEX high flow electric fans. Concurrently, FSG-04 directs operators to block open various Class 1E equipment room doors and establish a flow path for the dispersion of hydrogen gas. FSG-05 specifies that this action should occur within 8 hours after initiation of an ELAP (i.e., prior to recharging the batteries).

Licensee calculation change notice GK-370-004-CN002, "Battery Rooms Hydrogen Concentration," Revision 4, dated 9/29/16, showed that with no ventilation the time required for the hydrogen concentration to reach 2 percent by volume would be 272 hours. Thus, the room may be isolated for 11 days before outside air ventilation must be utilized. Therefore, depending on the battery room temperature, the licensee's actions to implement guidance in FSG-04 and FSG-05 to establish outside air ventilation would ensure that hydrogen concentration remains within acceptable limits.

Based on its review of the licensee's evaluations and mitigating actions in FSG-04 and FSG-05, the NRC staff concludes that hydrogen accumulation in the WCGS safety-related battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

During the audit process, the NRC staff reviewed the licensee's calculation DE-17-001, "Post-ELAP Main Control Room Temperature," Revision 0. This calculation evaluates the projected MCR temperature profile and accounts for the opening of doors and establishment of a ventilation flow path in the MCR. With these provisions, the MCR temperature is projected to be less than 102°F. During the audit process, the staff reviewed licensee FSGs, and confirmed that they provide guidance to establish temporary ventilation, if necessary, in the MCR.

Based on the licensee's calculations and administrative guidance for deployment of temporary ventilation which should maintain the MCR temperature below 110°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 personnel performing light work for a 4 hour period while dressed in conventional clothing), the NRC staff finds that the MCR will not be adversely impacted by the loss of ventilation as a result of an ELAP.

3.9.2.2 Spent Fuel Pool Area

See Section 3.3.4.1.1 above for the detailed discussion of ventilation and habitability considerations in the SFP area. In general, the licensee plans to establish Fuel Building ventilation and deploy any necessary hoses before the SFP boiling affects habitability. The licensee also has the ability to add water to the SFP without accessing the refueling floor. Thus, the staff concludes that the licensee's strategy has adequate provisions for habitability in this area.

3.9.2.3 Other Plant Areas

TDAFW Pump Room

Although the TDAFW pump room gets as hot as 146°F, the licensee's strategy does not involve prolonged manual operation in the TDAFW pump room. In event that the TDAFW pump trips offline, operators may have to enter the room temporarily to restart the pump. Since this is a short duration activity the NRC staff concludes that the calculated room temperatures should not inhibit a successful reset. Furthermore, as stated earlier, the licensee has procedures in place to establish temporary ventilation to reduce the temperature in the TDAFW pump room. Eventually, the TDAFW pump will not be required once the site transitions to the FLEX SG makeup pump after approximately 10 hours into the event.

Main Steam/Feedwater Tunnel

The FIP states that calculation XX-M-090 determined the Main Steam Tunnel will reach a temperature of 220°F during the 7-day ELAP. In the event of a loss of dc power and nitrogen gas to the ARVs, operators may need to momentarily enter the Main Steam Tunnel to manually adjust the ARVs. During the audit, the staff noted that the licensee's procedure EMG C-0, has a caution addressing the use of personnel protective equipment and stay times when accessing local SG ARV controls.

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 Steam Generator Makeup

In its FIP, the licensee indicated that the CST is the preferred source of SG makeup and is protected against all applicable BDBEEs. The licensee's FIP also indicates that the CST contains a usable volume of 385,646 gallons that would provide approximately 31 hours of RCS decay heat removal concurrent with a 100°F/hour RCS cooldown to a minimum SG pressure of 310 psig.

In its FIP, the licensee also described that the UHS is assumed to be the only large water source protected against all applicable BDBEE and consists of a cooling pond that provides a water volume of 455 acre-feet (approximately 148 million gallons) behind a Seismic Category I dam built in one finger of the cooling lake. The licensee also indicated that the UHS can be used as an available water source for RCS and SFP Cooling.

In its FIP, the licensee described that the cooling lake would be available as a source of water for the SG under all applicable BDBEE with the exception of an earthquake. The licensee further indicated that the cooling lake is the largest volume of water on site with a nominal volume of 111,280 acre-feet. In addition, the cooling lake also serves as an available water source for RCS and SFP cooling.

Table 3 of the licensee's FIP provides a list of potential SG water sources in order of preferred use, including their capacities, and an assessment of their availability following various postulated BDBEEs applicable to the site. In addition to the three sources listed above, the licensee may also be able to utilize the demineralized water storage tank containing 50,000 gallons, reactor makeup water storage tank containing 126,000 gallons, the portable water storage tank containing 70,000 gallons, and the RWST containing 394,000 gallons.

3.10.2 Reactor Coolant System Makeup

In its FIP, the licensee described that four SI accumulators, each with a capacity between 6,122 gallons and 6,594 gallons, will start to inject borated water into the RCS when pressure drops below approximately 600 psig. The accumulators provide RCS makeup capability to ensure natural circulation is maintained within the RCS as well as negative reactivity insertion to the RCS during Phase 1.

In its FIP, the licensee described that the preferred borated water source for RCS injection is the two BATs located in the Auxiliary Building that are protected against all applicable BDBEEs. The licensee also indicated that the combined minimum BAT volume is 17,658 gallons with a boron concentration between 7,000 and 7,700 ppm.

In its FIP, the licensee also described that the RWST is another borated water source for RCS injection that is protected against all applicable BDBEEs, with the exception of high wind missiles. The licensee also indicated that the RWST borated volume is maintained greater than 394,000 gallons at a boron concentration no less than 2,400 ppm.

In its FIP, the licensee stated that a mobile boration skid will be available from NSRC for the Phase 3 strategy.

3.10.3 Spent Fuel Pool Makeup

In its FIP, the licensee described that any water source available is acceptable for use as makeup to the SFP; however, the primary water source would be from the RWST through the FLEX SFP pump. The CST would serve as the alternate preferred backup water source for SFP makeup. The licensee also stated that water quality, or boration, is not a significant concern for makeup to the SFP.

3.10.4 Containment Cooling

In its FIP, the licensee described that with the installation of the low leakage RCP shutdown seals, the containment pressure and temperature are not expected to rise to levels which could challenge the containment structure. The licensee also described in its FIP that during Phase 3, specific actions to reduce containment temperature are not required to ensure continued functionality of the key parameters as a reduction of the RCS heat load indirectly reduces containment temperature and pressure. The licensee further indicated in its FIP that restoration of the CCW, ESW, and RHR systems is performed in Phase 3 to support core cooling and conditions in containment will improve with no further actions.

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. When the ELAP occurs with the plant at power, the mitigation strategy initially focuses on the use of the steam-driven TDAFW 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 reactor vessel 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's analysis shows that following a full core offload to the SFP, about 15 hours are available to implement makeup before boil-off results in the water level in the SFP dropping far enough to reach the top of the fuel racks, and the licensee has stated that they have the ability to implement makeup to the SFP within that time.

When a plant is in a shutdown mode in which steam is not available to operate the TDAFW pump and allow operators to release steam from the SGs (which typically occurs when the RCS has been cooled below about 300°F), another strategy must be used for decay heat removal. The NRC-endorsed strategy was originally described in an NEI position paper that has subsequently been incorporated into Section 3.2.3 of NEI 12-06, Revision 2. This strategy 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. In its FIP, the licensee stated that it was abiding by the position paper, and the licensee's FIP also states that the WCGS FLEX program has been developed in accordance with NEI 12-06, Revision 2.

Based on the licensee's incorporation of 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

In its FIP, the licensee described that the inability to predict actual plant conditions that require the use of FLEX equipment makes it impossible to provide specific procedural guidance. As such, FLEX procedures provide guidance that can be employed for a variety of conditions. Clear criteria for entry into FSGs ensures that FLEX strategies are used only as directed for BDBEE conditions, and are not used inappropriately in lieu of existing procedures. When FLEX equipment is needed to supplement EOP or off-normal operating procedure (OFN) strategies, the EOP or OFN directs entry into the appropriate FSG. The FSGs have been developed per PWROG guidelines. The FSGs provide instructions for implementing available, pre-planned FLEX strategies to accomplish specific tasks in the EOPs or OFN's. The FSGs are used to supplement (not replace) the existing procedure structure that establishes command and control for the event.

In addition, the FIP describes that procedural interfaces have been incorporated into procedure EMG C-0, and procedure OFN NB-034, "Loss of All AC Power-Shutdown Conditions," to the extent necessary to include appropriate reference to FSGs and provide command and control for the ELAP.

As described in the FIP, the FSGs have been reviewed and validated by the involved groups to the extent necessary to ensure that implementation of the associated FLEX strategy is feasible. Specific FSG validation was accomplished via table top evaluations and walk-throughs of the FSGs when appropriate.

In its FIP, the licensee described that FSG maintenance is performed by the operations procedures group per procedure AP 15C-004, "Preparation, Review and Approval of Procedures, Instructions, and Forms."

3.12.2 Training

In its FIP, the licensee described that the training program has been revised to assure personnel proficiency in utilizing FSGs and associated FLEX equipment for the mitigation of BDBEEs is adequate and maintained. These programs and controls were developed and have been implemented per the Systematic Approach to Training (SAT) Process.

In its FIP, the licensee also indicated that initial training has been provided and periodic training will be provided to emergency response organization leaders on BDBEE emergency response strategies and implementing guidelines. Personnel assigned to direct the execution of the FLEX mitigation strategies for BDBEEs have received the necessary training to ensure familiarity with the associated tasks, considering available job aids, instructions, and mitigation strategy time constraints. Care has been taken to not give undue weight (in comparison with other training requirements) to operator training for BDBEE accident mitigation. The testing/evaluation of operator knowledge and skills in this area have been similarly weighted.

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 the mitigating strategies. The procedures have been issued in accordance with NEI 12-06, Section 11.4, and a training program has been established and will be maintained in accordance with NEI 12-06, Section 11.6.

3.13 Maintenance and Testing of FLEX Equipment

As a generic issue, NEI submitted a letter to the NRC dated October 3, 2013 [Reference 38], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 39], the NRC endorsed the use of the EPRI report and the EPRI database as providing a useful input for licensees to use in developing their maintenance and testing programs.

In its FIP, the licensee described that initial component level testing, consisting of factory acceptance testing and site functional testing, was conducted to ensure the portable FLEX equipment can perform its required FLEX strategy design functions. Factory acceptance testing verified that the portable equipment performance conformed to the manufacturers rating for the equipment as specified in the purchase order. Verification of the vendor test documentation was performed as part of the receipt inspection process for each of the affected pieces of equipment and included in the applicable vendor technical manuals. Site acceptance testing confirmed factory acceptance testing to ensure portable FLEX equipment delivered to the site performed per the FLEX strategy functional design requirements.

The FIP also described that the portable FLEX equipment that directly performs a FLEX mitigation strategy for the core cooling, containment, or SFP cooling is subject to periodic maintenance and testing per NEI 12-06 and the Institute of Nuclear Power Operations (INPO) AP-913, "Equipment Reliability Process," to verify proper function. Additional FLEX support equipment that requires maintenance and testing will have preventive maintenance to ensure it will perform its required functions during a BDBEE.

In addition, the FIP described that EPRI has completed and has issued a document titled "Preventive Maintenance Basis for FLEX Equipment - Project Overview Report." The PM templates for the major FLEX equipment including the portable diesel pumps and generators have also been issued. The PM templates include activities such as:

- Periodic Static Inspections - Monthly walkdown
- Fluid analysis - Annually
- Periodic operational verifications
- Periodic performance tests

In addition, the FIP described that PM procedures and test procedures are based on the templates contained within the EPRI PM Basis Database, or from manufacturer provided information/recommendations when templates were not available from EPRI. The corresponding maintenance strategies were developed and documented. The performance of the PMs and test procedures are controlled through the site work order process. FLEX support equipment not falling under the scope of INPO AP-913 will be maintained as necessary to ensure continued reliability. Performance verification testing of FLEX equipment is scheduled and performed as part of the WCGS PM process.

In its FIP, the licensee described that the unavailability of equipment and applicable connections that directly perform a FLEX mitigation strategy for core cooling, containment, and SFP cooling will be managed such that risk to mitigation strategy capability is minimized. Maintenance/risk guidance conforms to the guidance of NEI 12-06 as follows:

- Portable FLEX equipment may be unavailable for 90 days provided that the site FLEX capability (N) is available.

- If portable equipment becomes unavailable such that the site FLEX capability (N) is not maintained, initiate: actions within 24 hours to restore the site FLEX capability (N) and implement compensatory measures (e.g., use of alternate suitable equipment or supplemental personnel) within 72 hours.

The licensee's program, as described in the FIP, utilizes NEI 12-06, Revision 2, as its basis. Thus, though not described explicitly in the equipment maintenance and testing portion of the FIP, under certain circumstances regarding storage protection from external events, the licensee would use a 45-day allowed out-of-service time, in accordance with provision 11.5.4.b of NEI 12-06, Revision 2. The staff notes that this NEI 12-06 provision merits careful consideration with regard to the RCS makeup pumps, as described in Section 3.2.3.1 of this safety evaluation.

Based on the FIP description, 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's FIP lists the hoses and cables as an alternative; however, as discussed in Section 3.6.2 of this safety evaluation, the NRC staff concludes that the licensee meets the provisions of NEI 12-06 in this area. Thus, no alternatives to NEI 12-06, Revision 2 are evaluated for WCGS.

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 25], the licensee submitted its OIP for WCGS in response to Order EA-12-051. By letter dated July 17, 2013 [Reference 26], as corrected by letter dated August 1, 2013 [Reference 27], the NRC staff sent a request for additional information (RAI) to the licensee. The licensee provided a response to the RAI by letter dated August 15, 2013 [Reference 28]. By letter dated October 29, 2013 [Reference 29], the NRC staff issued an ISE and RAI to the licensee.

By letters dated August 28, 2013 [Reference 30], February 26, 2014 [Reference 31], August 21, 2014 [Reference 32], and February 24, 2015 [Reference 33], 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 June 30, 2015 [Reference 35], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved. In addition, by letter dated June 12, 2017 [Reference 51], the licensee provided additional information regarding the WCGS reliable SFP instrumentation.

The licensee has installed a SFP level instrumentation system designed by Westinghouse. The NRC staff reviewed the SFP level instrumentation system design specifications, calculations and analyses, test plans, and test reports for the Westinghouse system during a vendor audit. The staff issued an audit report regarding the Westinghouse system on August 18, 2014 [Reference 34].

4.1 Levels of Required Monitoring

In its RAI response letter dated August 15, 2013 [Reference 28], the licensee identified the SFP levels of monitoring as follows:

- Level 1 corresponds to a plant elevation of 2046'-0"
- Level 2 corresponds to a plant elevation of 2031'-1.25"
- Level 3 corresponds to a plant elevation of 2022'-1.25"

With regard to the Level 1 designation, in its compliance letter dated June 30, 2015 [Reference 35], the licensee stated that calculation EC-48, "Minimum Safety Limit for LSL- 57 & 58," concludes that elevation 2043'-6" is the SFP water elevation sufficient for the [SFP cooling] pump's required net positive suction head (NPSH). According to the licensee, the Level 1 indication is at the normal SFP water operating level of elevation 2046'-0" and sufficiently covers the calculated level to maintain the pump's required NPSH.

In its RAI response letter dated August 15, 2013, the licensee provided a sketch depicting the SFP levels of monitoring and the measurement range for the instrument channels as illustrated below.

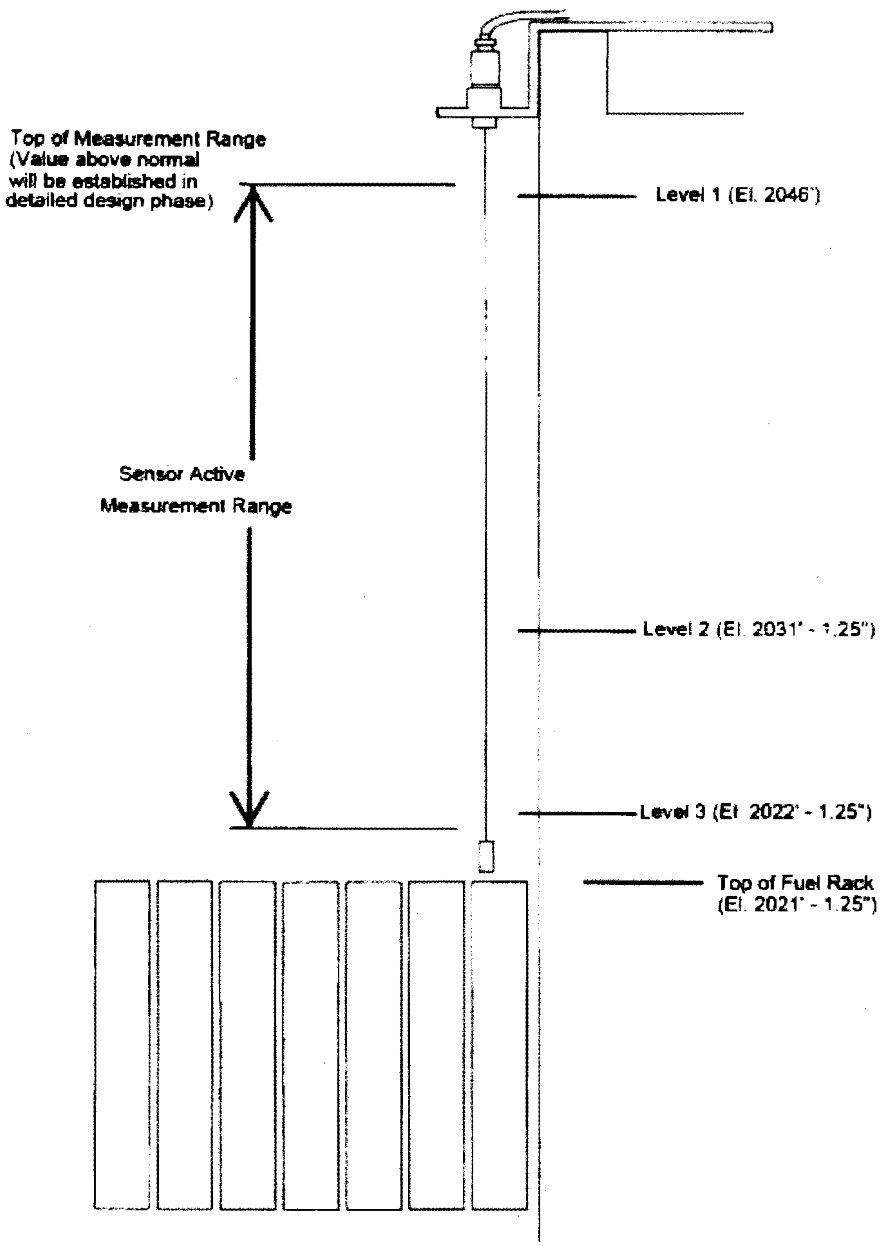

Figure 1 - Wolf Creek SFP Levels of Monitoring

The NRC staff's assessment of the licensee's selection of the SFP levels of monitoring is as follows:

- Level 1: Per NEI 12-02, Level 1 represents the HIGHER of two points. The first point is the water level at which suction loss occurs due to uncovering of the coolant inlet pipe, weir or vacuum breaker. The licensee's calculation EC-48, "Minimum Safety Limit for LSL-57 & 58," Revision 0, identifies 2040' - 0" as the plant elevation where suction loss occurs due to uncovering of the spent fuel coolant intake pipe. The second point is the water level at which loss of SFP pump NPSH occurs under saturated conditions. This is identified in EC-48, as 2043' - 6" plant elevation. Wolf Creek's Level 1 (2046' - 0") is higher than both of the above two points and is therefore consistent with NEI 12-02.

- Level 2: Level 2 was identified by the licensee as 2031' - 1.25" plant elevation. This level is consistent with the first of the two options described in NEI 12-02 for Level 2, which is 10 feet (+/- 1 foot) above the highest point of any fuel rack seated in the SFP.

- Level 3: Level 3 was identified by the licensee as 2022' - 1.25" plant elevation. This level is one foot above the highest point of any fuel rack seated in the SFP, which is consistent with NEI 12-02 guidance for Level 3.

Based on the evaluation above, the NRC staff finds that the licensee's selection of Levels 1, 2, and 3 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 Evaluation of Design Features

Order EA-12-051 requires that the SFP level instrumentation shall include specific design features, including specifications on the instruments, arrangement, mounting, qualification, independence, power supplies, accuracy, testing, and display. Below is the NRC staff's assessment of the design features of the SFP level instrumentation.

4.2.1 Design Features: Instruments

In its OIP dated February 28, 2013 [Reference 25], the licensee stated that the Wolf Creek SFP Instrumentation System (SFPIS) will utilize fixed primary and backup guided wave radar (GWR) sensors. The GWR technology provides the capability to reliably monitor the SFP water level under adverse environmental conditions. The GWR technology uses the principle of time domain reflectometry to detect the SFP water level. A microwave signal is sent down the cable probe sensor, and when it reaches the water, it is reflected back to the sensor electronics. This is due to the difference between the dielectric constants of air and water. Using the total signal travel time, the sensor electronics embedded firmware computes the level of the water in the SFP.

Related to the SFP level instrumentation's measurement range, the licensee stated in its OIP that the primary and backup instrument channels will provide continuous level indication over a range of 23' - 10.75", from 12 inches above the top of the fuel storage racks (plant elevation of 2022' - 1.25") to the normal pool level elevation (plant elevation of 2046' - 0"). The NRC staff notes that the instrument's measurement range covers Levels 1, 2, and 3, as described in Section 4.1 above.

Based on the licensee's OIP description and RAI response, the NRC staff finds that the licensee's design, with respect to the number of SFP instrument channels and measurement range, 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 compliance letter, the licensee stated that:

> The primary element [level probe] is located at the north end of the pool at the east corner. The element terminates into a 90 degree metal coupler. The coupler is connected into liquid tight EMI/RFI [electromagnetic interference/radiofrequency interference] resistant flex conduit, transitioning into rigid metal conduit. The rigid metal conduit is first routed horizontally close to the fuel pool floor and then routed vertically, up the north wall before turning west and running horizontally before exiting the Fuel [Handling] Building into the Auxiliary Building. The backup element is located at the north end of the pool at the west corner. Like the primary element, the backup element terminates into a 90 degree metal coupler. The coupler is connected into liquid tight EMI/RFI resistant flex conduit, transitioning into rigid metal conduit. The rigid metal conduit is first routed horizontally close to the fuel pool floor and maintains this above floor level route before turning west and running horizontally before exiting the Fuel [Handling] Building into the Auxiliary Building.
>
> A separation distance of greater than 16 feet is maintained between primary and backup conduits routed in the Fuel [Handling] Building.
>
> By routing primary and backup cables in flex and rigid conduit and providing significant physical separation between primary and backup cables in the Fuel [Handling] Building, missiles will not disable both primary and backup systems of the SFPIS.

Based on the licensee's description of the instrument channel arrangement, the NRC staff's review concludes that there is sufficient channel separation between the primary and backup level instrument channels, sensor electronics, and routing cables to provide reasonable protection against loss of SFP level indication due to missiles that may result from damage to the structure over the SFP. The NRC staff finds that the licensee's arrangement for the SFP level instrumentation, if implemented as described, 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 compliance letter, the licensee stated that:

> The mounting bracket for the sensing probe was designed according to the plant design basis for SSE seismic hazard curve at the appropriate plant elevation. Loads that were considered in the evaluation of the bracket and its mounting are: (1) Static loads including the dead weight of the mounting bracket in addition to the weight of the level sensing instruments and cabling, and (2) Dynamic loads including the seismic load due to excitation of the instruments dead weight in addition to the hydrodynamic effects resulting from the excitation of the SFP water.

The licensee's compliance letter also indicates that a response spectra analysis was performed for the seismic evaluation of the mounting bracket using GTSTRUDL [structure design and analysis software]. Hydrodynamic effects on the mounting bracket were evaluated using TID-7024, "Nuclear Reactors and Earthquakes," 1963, and added to the GTSTRUDL model. Plant acceptance criteria and applicable codes were used for the design of the bracket and its anchorage.

In addition, the compliance letter states the following regarding the mounting bracket analysis:

> All members' results were shown to be adequate for the loads and load combinations used in the analysis. Welded and bolted connections were evaluated and were shown to be adequate with significant margin. Base plate of the mounting bracket and anchorage to concrete were evaluated using Plate Wizard in GTSTRUDL and designed to meet the plant criteria for base plates and anchors.

By letter dated June 12, 2017 [Reference 51], the licensee provided supplemental information regarding the SFP level instrumentation. In this letter, the licensee stated that per Design Change Package (DCP) 014403, "Spent Fuel Instrument System," the sensor bracket is designed according to the plant's SSE design basis at the 2047'-6" Fuel Building elevation, with additional conservatism of two times the acceleration values. The instructions in the design package specify that the conduit supports be installed consistent with safety-related, Seismic Category I standards. As for the SFP level instrumentation electronics mounting, the licensee stated that the sensor electronics module is mounted using a seismic Class-1 bracket. The indicator enclosures are also seismically mounted (Class-1) such that they would withstand a SSE.

The NRC staff noted that the licensee's site-specific seismic analyses demonstrated that the SFP level instrumentation's mounting design allows the instrument to perform its design function during and following the maximum seismic ground motion. The staff's audit review concludes that the assumptions and modeling used in the sloshing analysis for the sensor mounting bracket are adequate. Further, the staff concludes that the design criteria and methodology used to estimate and test the total loading on the mounting devices, including the design-basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing were appropriate.

Based on the evaluation above, the NRC staff finds the licensee's mounting design for the SFP level instrumentation 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 RAI response letter, the licensee stated that appropriate quality assurance measures will be selected for the SFPIS, consistent with Appendix A-1 of NEI 12-02 and similar to those imposed by Regulatory Guide 1.155.

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 Equipment 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 components used 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.

During the Westinghouse system vendor audit, as described in Reference 34, the NRC staff reviewed the SFP level instrumentation's qualifications and testing for temperature, humidity, radiation, shock and vibration, seismic, and electromagnetic compatibility (EMC). During the WCGS audit process, the staff further reviewed the anticipated seismic, radiation, and environmental conditions to verify that the vendor qualification and testing were appropriately bounded by the site-specific parameters. Below is the staff's assessment of the reliability of the WCGS SFP level instrumentation.

4.2.4.2.1 Radiation, Temperature, and Humidity

4.2.4.2.1.1 Radiation

In its supplement dated June 12, 2017, the licensee stated that with the water level in the SFP at the top of the fuel racks, a dose rate of slightly less than 7E+03 Rad/hour was projected. For the purposes of calculating a maximum reasonable dose, this rate was increased to 1E+04 Rad/hour. For seven days (168 hours), the total integrated dose (TID) is 1.68E+06 Rad, which was rounded up to 2E+06 Rad, and then finally multiplied by 5 to allow for some margin in regards to the distribution of the spent fuel assemblies in the pool. Based on NRC staff questions during the audit process, the licensee also stated that the Wolf Creek FSGs initiate actions that would prevent the water level in the SFP from falling to Level 3. If these guidelines are implemented properly, the water level in the pool will always be at least 10 feet above the top of the racks. According to the licensee, this additional shielding will reduce the projected dose rate by five orders of magnitude.

The SFPIS transmitters are located in the personnel hatch area (Room 1507) of the Auxiliary Building. In its supplement dated June 12, 2017, the licensee stated that during a BDBEE, the expected radiological conditions in the Auxiliary Building would be consistent with the normal operating conditions of the Wolf Creek environmental qualification design-basis document, EQSD-I. This determination is based on: (1) an assumption that the RCS remains intact and core cooling is maintained by the FLEX mitigating strategies, and (2) the SFP makeup capabilities exceed the calculated SFP boil off rate, thus maintaining the integrity of the fuel stored in the SFP. The estimated dose rate in Room 1507 for these conditions is less than 0.0025 Rad/hour and the TID for 7 days is 0.42 Rad. For the BDBEE radiological conditions in the HVAC [Heating, Ventilating and Air Conditioning] areas (Rooms 1501 and 1512), where the electronics enclosures are located, the dose rate is predicted to be less than 0.0005 Rad/hour and the TID for 7 days is 0.084 Rad.

Based on the expected conditions presented in the licensee's supplement, the NRC staff review concludes that the design limits of the Westinghouse equipment envelop the anticipated Wolf Creek's BDBEE radiological conditions.

4.2.4.2.1.2 Temperature and Humidity

In its supplement dated June 12, 2017, the licensee stated that the level instrument systems were tested for reliability at temperature and humidity levels consistent with the SFP water at saturated conditions for an extended period. More specifically, the portions of the system in the SFP area are expected to withstand the radiological conditions for a normal refueling of freshly discharged (100 hours) fuel with the SFP water at approximately 1 foot above the fuel racks. The instrument is designed to operate within the SFP at a temperature of 212°F and with 100 percent relative humidity (boiling water and/or steam environment), and in a concentrated borated water environment.

As for the environmental conditions outside of the SFP area, in its compliance letter, the licensee stated that there are no high energy lines located on the 2047' elevation of the Auxiliary Building, where the sensor electronics are located. The normal temperature for the applicable Auxiliary Building Rooms (1501, 1507, and 1512) is 104°F. Without ventilation and no significant heat sources, the licensee does not expect the temperature to exceed 140°F, the specified operating temperature for the equipment. The sensor electronics are rated for humidity of 0-95 percent (non-condensing). The Fuel Building is connected to the Auxiliary Building on the 2047' elevation by a pressure door, which is designed to limit leakage. Therefore, according to the licensee, environmental conditions in Rooms 1501, 1507, and 1512 are expected to remain within equipment qualification parameters.

Based on the licensee's expected temperature and humidity projections, the NRC staff concludes that the Westinghouse equipment's design limits envelop the anticipated Wolf Creek's BDBEE conditions.

In summary, the staff finds that the licensee's testing and equipment qualifications envelop the anticipated Wolf Creek's radiation, temperature, and humidity conditions both during a postulated BDBEE, as well as after the event. Thus the staff concludes that the SFP instrumentation should maintain its functionality under projected BDBEE conditions.

4.2.4.2.2 Shock and Vibration

In its supplement dated June 12, 2017, the licensee stated that the effects of shock and vibration on the SFPIS components during and following any applicable event were considered. All components located within the SFP are passive and are inherently resistant to shock and vibration loadings. According to the licensee, active electronics components located outside the SFP building are permanently and rigidly attached to seismic racks or structural walls, and are not subject to significant shock and vibration loadings.

Based on the licensee's description, the NRC staff concludes that if implemented appropriately, the SFP level instrumentation should provide its design functions with respect to shock and vibration.

4.2.4.2.3 Seismic

In its supplement dated June 12, 2017, the licensee stated that the SFPIS equipment underwent seismic testing by Westinghouse prior to installation at WCGS. According to the licensee, this testing was performed in accordance with IEEE [Institute of Electrical and Electronics Engineers] Standard 344-2004, which is endorsed by NRC Regulatory Guide 1.100. The required response spectra included the 10 percent margin recommended by IEEE Standard 323-2003. The licensee further stated that the WCGS design change package specified that a response spectra analysis has been performed for the seismic evaluation of the sensor bracket. The analysis was performed with finite element analysis software using the SNUPPS plant floor response spectra at the operating deck elevation (2047'-6") in the Fuel Building (i.e. the mounting floor elevation). Damping values were chosen according Regulatory Guide 1.61, Table-1 for the SSE, which is consistent with the design basis of the station.

Based on the licensee's description, the NRC staff review concludes that the SFP level instrument was designed and tested to seismic conditions that envelop the SSE seismic criteria at Wolf Creek, and is therefore acceptable.

4.2.4.2.4 Electromagnetic Compatibility

As a result of the NRC staff's evaluation of the EMC testing results during the vendor audit, the staff identified a generic open item applicable to all licensees using the Westinghouse SFP level instrument to identify whether any additional measures or site-specific installation instructions are required to address the potential for an EMC event to impact the SFPI equipment. The NRC staff asked the licensee whether an assessment of the potential susceptibilities of EMI/RFI in the areas where the SFP instruments are located had been performed, and how to mitigate those susceptibilities, if identified. The licensee provided a response to the NRC staff's concern in its supplement dated June 12, 2017, in which it stated that the SFPIS has been designed and tested to EMI/RFI standards. In addition, the licensee stated that the SFPIS design package specified several features to reduce system susceptibility to EMI/RFI, including the following:

- Low impedance grounding of all SFPIS components.

- The use of EMI/RFI resistant flexible conduits.
- The use of rigid conduit.
- Special EMI/RFI gasketing including sealing compounds for condulet covers.
- The use of glands to ground signal cable at the entry points of panels.

The licensee further stated that risk of EMI/RFI is further reduced by locating the transmitter boxes and the electronics enclosures away from any sources of EMI/RFI such as large induction motors, variable frequency drives, uninterruptible power supplies (UPSs) and other sources of power switching.

In addition, in its supplement dated June 12, 2017, the licensee stated that EMI/RFI testing was not performed after the SFP level instrumentation was installed and no administrative barriers prohibiting the use of radios adjacent to the transmitter or electronic enclosures were put in place. During the audit process, the staff raised a concern regarding this lack of administrative barriers. In response, the licensee generated Condition Report 00111177 and performed a detailed review of the pre-installation EMI/RFI testing. The licensee stated that this testing included a walkie-talkie test. The test started with the radio greater than 10 feet from all three electronic devices, transmitter enclosure/cabinet, electronics cabinet display and launch plate. The distance was decreased down to less than 1 foot with output current monitored for change. No changes were observed and the testing was considered to be acceptable.

The NRC staff reviewed the licensee's EMI/RFI testing, design features and location plans, and concluded that the concern with respect to potential susceptibilities of EMI/RFI had been appropriately addressed. Given the lack of administrative controls at the locations of the transmitter or the level indicator enclosures, the staff expects that any future operational experience regarding adverse EMI/RFI impacts to the SFPIS would be appropriately entered and dispositioned in accordance with the licensee's Corrective Action Program.

Based on the information provided by the licensee regarding equipment reliability, as described above, the NRC staff finds 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 Design Features: Independence

In its RAI response letter, the licensee stated that within the SFP area, the brackets will be mounted as close to the northwest (primary sensor) and northeast (backup sensor) corners of the SFP as permanent plant structures allow. Each system will be installed using completely independent cabling structures, including routing of the interconnecting cable within the SFP area in separate hard-pipe conduits. Primary and backup systems will be completely independent of each other, having no shared components.

For the SFP level instrument channel's electrical independence, in its compliance letter, the licensee stated that the primary and backup SFPIS channels are powered by separate ac buses. One of the channels is powered by Class 1E electrical bus NG01A via non-Class 1 electrical panel PN07, while the other channel is powered by Class 1E electrical bus NG02A via non-Class 1 electrical panel PN08.

The NRC staff noted that with the licensee's design, the loss of one level instrument channel would not affect the operation of the other channel under BDBEE conditions. The instrument channels' physical separation was evaluated in Section 4.2.2 of this safety evaluation. Based on the information provided by the licensee, the staff finds the licensee's 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 compliance letter, the licensee stated that the primary and backup SFPIS channels are powered by separate ac buses. Each channel of equipment has an independent UPS with a 24 Vdc battery backup that ensures at least 72 hours of battery power without ac power. Additionally, an interface is provided for an alternate power supply, such as a FLEX generator. The SFP level can be functional for at least 3 days under station blackout conditions with battery power only, and the staff notes that both channels have the capability to function over the long-term with an alternate power supply.

Guidance document NEI 12-02 stipulates that electrical power supply for each channel shall be provided by different sources and that all channels shall provide the capability of connecting the channel to a source of power independent of the normal plant ac and dc systems. The NRC staff finds the licensee's power supply design appears to meet these provisions and is thus 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 compliance letter, the licensee stated that channel accuracy for each SFPIS instrument channel is ± 3 inches for the full level measurement span. This span extends from the normal SFP surface level (or higher) to within six inches of the fuel assembly under both normal and BDBEE conditions. In addition, both SFP primary and backup 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, an electronic output calibration will be required. The calibration adjustment is performed to restore level measurement accuracy to within the acceptance criteria at multiple points of the full span.

The NRC staff review finds that if the licensee's design and calibration protocol is implemented properly, the instrument channels should maintain the designed accuracy following a power source change or interruption without the need of recalibration. Further, the expected instrument channel accuracy performance under both normal and BDBEE conditions is within the specified provisions of NEI 12-02. Thus, the staff review concludes that the 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 OIP, the licensee stated that the instrument channel design will provide for routine testing and calibration consistent with NEI 12-02. The NRC staff vendor audit of the Westinghouse system had found that the SFP level instrumentation is capable of routine testing and calibration.

Based on the audit review of the licensee's site-specific implementation of the Westinghouse design, the staff concludes that the licensee's SFP instrumentation design allows for adequate testing, consistent with the vendor audit conclusion. Thus, the testing features of the WCGS system appear to be consistent with the NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.9 Design Features: Display

Regarding the SFP level instrumentation display location, in its supplement dated June 12, 2017, the licensee stated that the primary and backup level electronics cabinets (instrument displays) are located in Rooms 1501 and Room 1512 of the Auxiliary Building. The radiological and environmental conditions of these rooms are discussed in Section 4.2.4.2 of this safety evaluation. Based on the information the licensee provided, these rooms are considered to be a mild environment and would be habitable during a BDBEE. In addition, in its supplement dated June 12, 2017, the licensee stated that the displays are located a short walk from the MCR.

Guidance document NEI 12-02 specifies that the SFP level displays should be promptly available to the plant staff and decision makers. It further states that they should be located in an appropriate and accessible location. Based on the descriptions provided, the NRC staff finds that the licensee's location of the SFP level 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

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that the 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.

Guidance document NEI 12-02 specifies that the SAT process can be used to identify the population to be trained, and also to determine both the initial and continuing elements of the required training. The NRC staff finds that the licensee's plan to train personnel in the operation, maintenance, calibration, and surveillance of the SFP level instrumentation, 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 RAI response letter, the licensee stated that site procedures will be developed for system inspection, calibration and test, maintenance, repair, operation and normal and abnormal responses, in accordance with WCNOC procedure controls. Technical objectives to be achieved in each of the respective procedures are described below.

- System Inspection

  To verify that system components are in place, complete, and in the correct configuration, and that the sensor probe is free of significant deposits of crystallized boric acid.

- Calibration and Test

  To verify that the system is within the specified accuracy, is functioning as designed, and is appropriately indicating SFP water level.

- Maintenance

  To establish and define scheduled and preventive maintenance requirements and activities necessary to minimize the possibility of system interruption.

- Repair

  To specify troubleshooting steps and component repair and replacement activities in the event of system malfunction.

- Operation

  To provide sufficient instructions for operation and use of the system by plant operation staff.

- Responses

  To define the actions to be taken upon observation of system level indications, including actions to be taken at the levels defined in NEI 12-02.

Guidance document NEI 12-02 states that procedures will be developed using guidelines and vendor instructions to address maintenance, operation and abnormal response issues. Based on the RAI response, the staff finds that the licensee's procedure development appears to be consistent with NEI 12-02, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.3 Programmatic Controls: Testing and Calibration

In its compliance letter, the licensee described testing and calibration programs for the SFP level instrumentation as follows:

> Channel comparisons, comparing level indication on one channel against the level indication on the other channel, will be performed on a daily basis.... The periodic calibration tests and functional checks will be performed within 60 days

of a planned refueling outage considering normal testing scheduling allowances (e.g., 25 percent). This calibration check is not required to be performed more than once per 12 months.

In its supplement dated June 12, 2017, the licensee provided a listing of the PM tasks related to the SFP level instrumentation and associated frequencies. In addition, the licensee described the compensatory measures for the SFP level instrument channel(s) out-of-service in its supplement dated June 12, 2017. Specifically, the licensee stated that Wolf Creek Procedure AP 21A-002, "Diverse and Flexible Coping Mitigation Strategies (FLEX) Program," provides programmatic guidance for managing the availability of both channels of the SFP level instrumentation consistent with the requirements of NEI 12-02. Step 6.17.5.1 of this procedure states that, "One channel of FLEX SFP Wide Range Level Instrumentation can be out-of-service for 90 days if the other channel is functional. If one channel is out-of-service for greater than or equal to 90 days, then compensatory actions must be implemented." Step 6.17.5.2 states that, "If both channels of FLEX SFP Wide Range Level Instrumentation become non-functioning, then initiate action within 24 hours to restore one channel to service and initiate compensatory actions within 72 hours."

In its supplement dated June 12, 2017, the licensee also stated that Wolf Creek performs daily checks on the SFP level instrumentation to verify they are operational. The checks compare the level variance between the SFP level instrumentation channels and each SFP level instrumentation channel is checked against actual SFP level. Discrepancies are recorded in the reading sheet and reported to the control room resulting in generation of a condition report and an entry into the Equipment Out-Of-Service Log (EOL). The allowed out-of-service (OOS) time for the SFP level instrumentation channel is tracked by the operations department in the EOL. The condition report is entered into Corrective Action Program where it is reviewed by plant staff and assigned to the appropriate department to repair/restore the equipment within the allowed OOS time.

The NRC staff's review confirmed that the licensee's testing and calibration plan, as well as the maintenance program, appears to be consistent with the vendor recommendations. Additionally, compensatory actions for instrument channel(s) out-of-service appear to be consistent with guidance in NEI 12-02.

Based on the evaluation above, the staff finds that the licensee's testing and calibration program appears to be consistent with NEI 12-02, 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 compliance letter dated June 30, 2015 [Reference 35], the licensee stated that it 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's plans conform to the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. Based on the evaluation above, the NRC staff concludes that if the SFP level instrumentation is installed at Wolf Creek according to the licensee's 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 for Order EA-12-049 in May of 2016 [Reference 20]. The licensee reached its final compliance date on November 21, 2016, for Order EA-12-049 and May 3, 2015, for Order EA-12-051 [References 21 and 35, respectively], and has declared that the 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 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.

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Date: August 2, 2017

WOLF CREEK GENERATING STATION, UNIT 1 – 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. MF0788 AND MF0781) DATED AUGUST 2, 2017

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