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December 16, 2014

AEP-NRC-2014-89
10 CFR 50.4

Docket No.: 50-315

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

Donald C. Cook Nuclear Plant Units 1
Compliance with March 12, 2012, NRC Order Regarding Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)

Reference:

Letter from E. J. Leeds and M. R. Johnson, U. S. Nuclear Regulatory Commission, to All Power Reactor Licensees and Holders of Construction Permits in Active or Deferred Status, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012, Agencywide Documents Access Management System Accession No. ML12054A736.

In response to events at the Fukushima Dai-ichi nuclear power plant, the U. S. Nuclear Regulatory Commission (NRC) issued the referenced order, EA-12-049, to all power reactor licensees, including Indiana Michigan Power Company (I&M), the licensee for the Donald C. Cook Nuclear Plant (CNP) Unit 1. The order directed licensees to develop, implement, and maintain guidance and strategies to restore or maintain core cooling, containment, and spent fuel pool cooling capabilities in the event of a beyond-design-basis external event. The order also directed licensees to report when full compliance with the requirements stated in the order was achieved. This letter reports compliance with the requirements of the order for CNP Unit 1.

Enclosure 1 to this letter provides an affirmation regarding the information contained herein. Enclosure 2 provides a description of CNP Unit 1 compliance with the order. Enclosure 3 provides responses to issues that have been addressed by I&M and are pending NRC closure.

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This letter contains no new or revised regulatory commitments. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

JRW/amp

Enclosures:

1. Affirmation
2. Donald C. Cook Nuclear Plant Unit 1 Compliance with NRC Order EA-12-049
3. Responses to Items from the U. S. Nuclear Regulatory Commission (NRC) Interim Staff Evaluation, NRC Audit, and 3rd Overall Implementation Plan Update, Which Have Been Previously Addressed by Indiana Michigan Power Company and are Pending NRC Closure.

c: M. L. Chawla, NRC Washington, DC
J. T. King, MPSC, w/o enclosures
E. J. Leeds, NRR, NRC
MDEQ – RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
A. J. Williamson, AEP Ft. Wayne, w/o enclosures

AFFIRMATION

I, Joel P. Gebbie, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Joel P. Gebbie
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 16th DAY OF December, 2014

Notary Public

My Commission Expires 01/21/2018

Enclosure 2 to AEP-NRC-2014-89

Donald C. Cook Nuclear Plant Unit 1 Compliance with NRC Order EA-12-049

References for this enclosure are identified in Section 5.

1. Introduction

In response to U.S. Nuclear Regulatory Commission (NRC) Order EA-12-049 (Reference 1), Indiana Michigan Power Company (I&M) developed an Overall Integrated Plan (OIP) (Reference 2) describing diverse and flexible mitigation strategies (FLEX) for responding to beyond-design-basis external events at the Donald C. Cook Nuclear Plant (CNP). As required by the order, I&M has submitted OIP status updates at six month intervals. The current Unit 1 strategies are described in a CNP FLEX Program document.

The order requires that licensees complete full implementation of the strategies no later than the second refueling outage after submittal of the OIP. The order also requires that licensees report when full compliance has been achieved. The NRC staff has requested that the compliance report be submitted within 60 days of commencing unit startup from the outage in which implementation of the strategies is required. I&M is hereby reporting that full compliance with the order was achieved prior to commencing the CNP Unit 1 startup, on October 23, 2014, from the second refueling outage after submittal of the OIP.

2. Open Item Resolution

The issues from the NRC Interim Staff Evaluation (ISE) (Reference 3), NRC Audit Report (Reference 4), and most recent OIP status update (Reference 5), which have been previously addressed by I&M and are pending NRC closure are as follows:

<u>ISE Open Items (OI)</u> – ISE OI 3.2.4.10, "Battery Duty Cycle."

<u>ISE Confirmatory Items (CI)</u> – ISE CI 3.1.1.2.A, "Deployment of FLEX Equipment," ISE CI 3.2.1.2.A, "Reactor Coolant Pump Seals," ISE CI 3.2.1.2.B, "Reactor Coolant Pump Seals," ISE CI 3.2.1.2.C, "Reactor Coolant Pump Seals," ISE CI 3.2.1.6.B, "SOE Timeline," Refueling," ISE CI 3.2.4.6.A, "Personnel Habitability," ISE CI 3.2.4.9.A, Fuel Consumption," ISE CI 3.2.4.10.A, Load Shedding."

<u>NRC Audit Questions (AQ)</u> – AQ 36 (Control Room temperature), AQ 53 (Steam Generator Power Operated Relief Valve – habitability).

<u>Audit Report SE Items</u> – SE #1 (Reactor Coolant System venting), SE #2 (Westinghouse standard seals), SE #5 (NOTRUMP code), SE #8 (procedure validation)..

<u>OIP Open Items</u> – No. 27 (time to establish flow to the Reactor Coolant System in Mode 6), and No. 29 (tool for water to Essential Service Water in Phase 3).

Responses to the above identified issues are provided in Enclosure 3 to this letter. The NRC staff has indicated that I&M's responses to all other issues identified in the NRC ISE, NRC Audit Report, and Open Items in the OIP status update, are acceptable. I&M intends to submit correspondence documenting the responses to these issues within 60 days of achieving compliance with NRC Order EA-12-049 for CNP Unit 2.

3. Milestone Schedule Status

The following table lists the milestones identified in the most recent OIP status update that are applicable to Unit 1, and reflects the status of the milestone on the required Unit 1 compliance date, i.e. when the unit startup was commenced on October 23, 2014.

Milestone Schedule	
Milestone	**Activity Status**
Submit 60-Day Status Report	Complete
Submit OIP	Complete
Submit Six-Month Updates:	
Update 1	Complete
Update 2	Complete
Update 3	Complete
Walk-throughs or Demonstrations	Complete
Perform Staffing Analysis	Complete
Modifications:	
Modifications Evaluation	Complete
Unit 1 Design Engineering	Complete
Unit 1 Implementation Outage	Complete
Storage:	
Storage Design Engineering	Complete
Storage Implementation	Complete
FLEX Equipment:	
Procure On-Site Equipment	Complete
Develop Strategies with Regional Response Center	Complete

Procedures:	
Pressurized Water Reactor Owners Group (PWROG) issues nuclear steam system supply-specific guidelines. (Modes 1-4)	Complete
PWROG issues nuclear steam system supply-specific guidelines. (Modes 5 & 6)	Site specific Mode 5 and 6 FLEX Support Guidelines (FSG) were issued and credited for CNP Unit 1 compliance with EA-12-049. I&M plans to modify these FSGs, if needed, following issuance of PWROG guidelines for Mode 5 & 6, which is expected in December 2014.
Create Site-Specific FSGs – Unit 1	Complete
Create Maintenance Procedures	Maintenance program requirements were established and initial maintenance of FLEX equipment was completed by the Unit 1 compliance date. Procedures for subsequent maintenance activities are to be issued as needed to support performance of the maintenance activity within its specified interval.
Training:	
Develop Training Plan	Complete
Training Complete	Complete
Unit 1 FLEX Implementation	Complete
Submit Completion Report	Complete with this Submittal

4. Order EA-12-049 Compliance Elements Summary

CNP Unit 1 compliance with Order EA-12-049 was achieved using the guidance in Nuclear Energy Institute (NEI) document NEI 12-06 (Reference 6) which has been endorsed by the NRC (Reference 7). The significant compliance elements have been addressed as described below.

STRATEGIES – COMPLETE

CNP Unit 1 strategies are in compliance with Order EA-12-049. The strategies are documented in the CNP FLEX Program FSG (PMP-4027-FSG-003). Enclosure 3 to this letter documents responses to all strategy-related ISE Open Items, ISE Confirmatory Items, NRC Audit Questions, Audit Report SE Items, and OIP Open Items which have been addressed by I&M and are pending NRC closure.

MODIFICATIONS – COMPLETE

The plant modifications required to support the FLEX strategies for CNP Unit 1 were implemented in accordance with the station design control process such that the associated systems and components are fully capable of supporting the FLEX strategies.

EQUIPMENT – PROCURED AND MAINTENANCE & TESTING – COMPLETE

The equipment required to implement the FLEX strategies for CNP Unit 1 was procured, received at CNP, initially tested and/or performance verified, and is available for use. The continued availability of the FLEX equipment and connection points is controlled by the CNP Technical Requirements Manual. In accordance with the CNP FLEX Equipment Program, periodic maintenance and testing is to be conducted through the use of the CNP Preventative Maintenance program.

PROTECTED STORAGE – COMPLETE

All equipment required to implement the FLEX strategies for CNP Unit 1 is stored in locations within the Owner Controlled Area. Equipment is located both inside and outside the Protected Area (PA). A storage building was constructed outside the PA in accordance with a CNP Engineering Change, and provides protection from the applicable site hazards. Storage facilities were placed within the PA in accordance with a CNP Engineering Change, and provide protection from the applicable site hazards. Debris removal equipment was staged outdoors outside the PA, with redundant equipment stored in separate locations to assure the required equipment will survive the applicable site hazards. The equipment required to implement the FLEX strategies for CNP Unit 1 was verified to be stored in its required locations by the CNP FSG for FLEX Equipment Inventory and Checks (12-OHP-5030-FSG-523) which requires periodic checks and inventory of equipment used in the FLEX strategies to ensure the equipment is available.

PROCEDURES – COMPLETE

FSGs for CNP Unit 1 were developed, and where appropriate, were integrated with existing procedures. The FSGs and affected existing procedures were verified and are available for use in accordance with the site procedure control program.

TRAINING – COMPLETE

Initial training for CNP Unit 1 was completed in accordance with an approved FLEX training program prepared in accordance with the Systematic Approach to Training process.

STAFFING – COMPLETE

The staffing assessment for CNP (Reference 8) was completed in accordance with the alternative to the 10 CFR 50.54(f) request for information regarding Near-Term Task

Force Recommendation 9.3. The alternative was approved by the NRC (Reference 9). The CNP staffing assessment did not identify any changes to be made to the CNP Emergency Plan. The CNP Staffing Assessment was approved by the NRC by Reference 10.

NATIONAL SAFER RESPONSE CENTERS – COMPLETE

I&M established a contract with Pooled Equipment Inventory Company (PEICo) and has joined the Strategic Alliance for FLEX Emergency Response (SAFER) Team Equipment Committee for off-site facility coordination. PEICo was confirmed to be ready to support CNP with Phase 3 equipment stored in the National SAFER Response Centers in accordance with the site specific SAFER Response Plan.

VALIDATION – COMPLETE

I&M completed a review which documented consistency of the CNP Unit 1 FLEX procedure validation actions with those prescribed in the NEI document titled "FLEX Validation Process." The review determined that there are adequate resources necessary to implement all required Unit 1 FLEX strategies within required the constraints identified for Phases 1 and 2.

FLEX PROGRAM DOCUMENT – ESTABLISHED

CNP FLEX program documents were developed, reviewed, and issued.

5. References

1. Letter from E. J. Leeds and M. R. Johnson, U. S. Nuclear Regulatory Commission (NRC), to all power reactor licensees and holders of construction permits in active or deferred status, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012, Agencywide Documents Access Management System (ADAMS) Accession No. ML12054A736.

2. Letter from J. P. Gebbie, Indiana Michigan Power Company (I&M), to NRC, "Donald C. Cook Nuclear Plant - Unit 1 and Unit 2, Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," AEP-NRC-2013-13, dated February 27, 2013, ADAMS Accession No. ML13101A381.

3. Letter from J. S. Bowen, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant - Units 1 and 2 - Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Order EA-12-049 (Mitigation Strategies) (TAC Nos. MF0766 and MF0767)," dated January 24, 2014, ADAMS Accession No. ML13337A366.

4. Letter from J. Boska, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Report for the Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051

(TAC Nos. MF0766, MF0767, MF0761, and MF0762)," dated August 13, 2014, ADAMS Accession No. ML14209A122.

5. Letter from J. P Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2 - Third Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated August 27, 2014, ADAMS Accession No. ML14241A236.

6. Nuclear Energy Institute Document 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, Revision 0, dated August 2012, ADAMS Accession No. ML12242A378.

7. NRC JLD-ISG-2012-01, Compliance with Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events Interim Staff Guidance," Revision 0, dated August 29, 2012, ADAMS Accession No. ML12229A174.

8. Letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Phase 2 On-Shift Staffing Assessment Report Requested by U. S. Nuclear Regulatory Commission Letter, 'Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident' dated March 12, 2012," AEP-NRC-2014-40, dated May 23, 2014.

9. Letter from P. S. Tam, NRC, to L. J. Weber, I&M, "Review of 60-Day Response to Request for Information Regarding Recommendation 9.3 of the Near-Term Task Force Related to the Fukushima Dai-ichi Nuclear Power Plant Accident (TAC Nos. ME8683 and ME8684)," dated June 8, 2012, ADAMS Accession Number ML12145A640.

10. Letter from M. Halter, NRC, to L. J. Weber, I&M, "Response Regarding Licensee Phase 2 Staffing Submittals Associated with Near-Term Task Force Recommendation 9.3 Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident (TAC Nos. MF4310, MF4311, MF4312, MF4313, MF4321, MF4322, MF4323, MF4324, MF4325, MF4326, and MF4327)," dated September 29, 2014, ADAMS Accession Number ML14262A296.

Enclosure 3 to AEP-NRC-2014-89

Responses to Items from the U. S. Nuclear Regulatory Commission (NRC) Interim Staff Evaluation, NRC Audit, and 3rd Overall Implementation Plan Update, Which Have Been Previously Addressed by Indiana Michigan Power Company and are Pending NRC Closure

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

Interim Staff Evaluation (ISE) Open Item (OI) 3.2.4.10.B

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Plan dated May 21-2014:

Battery Duty Cycle - Verify approach used to qualify the station batteries duty cycle to 12 hours.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Battery Duty Cycle -The licensee is seeking to extend battery life to 12 hours using load shed procedures. The NRC needs to review battery data that demonstrates satisfactory performance over this long period of time."

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

CNP [Donald C. Cook Nuclear Plant] has not adopted the NEI [Nuclear Energy Institute] position paper on extended battery duty cycles and NRC endorsement (ADAMS Accession No. ML13241A188). Provide battery data.

Response to NRC Audit Item:

A Donald C. Cook Nuclear Plant (CNP) calculation verifies the CNP 250 volt-direct current (VDC) battery 12-hour coping time during an extended loss of all alternating current (AC) power (ELAP) event in conjunction with a battery deep load shed (DLS). The diverse and flexible mitigation strategy (FLEX) station battery run-time for the batteries (1AB, 1CD, 1N, 2AB, 2CD, and 2N) was calculated in accordance with Institute of Electrical and Electronics Engineers 485-2010 methodology. The results of the calculation meet the acceptance criteria associated with the DLS battery analysis, which is to maintain at least a 5 percent (%) margin with respect to installed 250VDC battery cell size. The acceptance criteria (> 5% Margin) were met for all the batteries for the 12-hour DLS duty cycle. The calculation complies with NRC endorsed (Agencywide Documents Access Management System (ADAMS) Accession No. ML13241A188) Nuclear Energy Institute (NEI) position paper titled "Battery Life Issue" (ADAMS Accession No. ML13241A186).

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE Confirmatory Item (CI) 3.1.1.2.A

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Plan dated May 21-2014:

Deployment of FLEX Equipment - Review the potential for soil liquefaction that might impede vehicle movement following a seismic event.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Deployment of FLEX Equipment - The licensee needs to demonstrate that the deployment path from staging area B to staging area A will not be adversely affected by effects such as soil liquefaction in a seismic event.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Provide liquefaction study.

Response to NRC Audit Item:

A FLEX haul path evaluation was performed which qualitatively evaluated the potential for liquefaction of soils beneath the haul path for FLEX equipment from the FLEX Storage Building (FSB) to their point of deployment within the protected area. The evaluation of the liquefaction potential of the haul path roadway used beyond design basis ground motion response spectra.

The evaluation determined estimated Haul Path settlements of up to 3 inches, which may slow traffic somewhat but should not impair transport vehicles from proceeding to the power block area.

The above evaluation was subsequently revised to add the haul path from Staging Area B and the north side of the Independent Spent Fuel Storage Installation area to the main plant access road into the evaluation. The revised evaluation therefore encompasses the entire path from Staging Area B to Staging Area A. The calculation determined that the conclusion of the previous evaluation, (settlements may slow traffic somewhat but should not impair transport vehicles from proceeding to the power block area) remained valid.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE CI 3.2.1.2.A

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Report dated August 13, 2014:

Reactor Coolant Pump [Reactor Coolant Pump] Seals - Confirm applicable analysis and relevant seal leakage testing data, which justifies that (1) the integrity of the associated O-rings will be maintained at the temperature conditions experienced during the ELAP event, and (2) the seal leakage rate of 21 gpm[gallons per minute]/seal used in the ELAP is adequate and acceptable.

Response to NRC Audit Item:

The Unit 1 Reactor Coolant Pump (RCP) seals were upgraded to the Generation 3 SHIELD® equipped low leakage design during the fall 2014 refueling outage. Indiana Michigan Power Company (I&M) is crediting the installation of the Generation 3 SHIELD® seals in its FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter from J. Davis, NRC, to J. A. Gresham, Westinghouse Electric Company, LLC, dated May 28, 2014. That NRC letter endorsed Westinghouse Technical Report TR-FSE-14-1-P and supplemental information provided by Westinghouse letters dated March 19, 2014, and April 22, 2014. The May 28, 2014 NRC letter documented the staff's conclusion that the Westinghouse Technical Report and supplemental information is acceptable for use in ELAP evaluations for Order EA-12-049 subject to four limitations and conditions. Each of these four limitations and conditions is restated below followed by a description of CNP Unit 1 compliance.

(1) *Credit for the SHIELD® seals is only endorsed for Westinghouse RCP Models 93, 93A, and 93A-1. Additional information would be needed to justify use of SHIELD® seals in other RCP models.*

    CNP Unit 1 compliance: The CNP Unit 1 RCPs are Model 93AS. The "S" designation refers to the presence of a spool piece between the pump and the motor that facilitates seal inspection and replacement. The seal package for Model 93A RCPs is identical to that for Model 93AS. Therefore CNP Unit 1 complies with this limitation/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). Nuclear power plants that predict higher cold-leg temperatures shall demonstrate the following:*

    a. *The polymer ring and sleeve O-ring remain at or below the temperature to which they have been tested, as provided in TR-FSE-14-1-P, Revision 1; or,*

b. *The polymer ring and sleeve 0-ring shall be re-tested at the higher temperature.*

CNP Unit 1 compliance: The maximum steady-state RCP seal temperature during an ELAP response is expected to be the T_{cold} corresponding to the lowest Steam Generator safety relief valve setting of 1065 pounds per square inch gage (psig). This corresponds to a T_{cold} value of 556 degrees Fahrenheit (^{o}F) to 557^{o}F. Therefore CNP Unit 1 complies with this limitation/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.*

CNP Unit 1 compliance:. Normal Unit 1 operating pressure is 2085 psig. Assuming a plant cooldown is initiated at the maximum allowed time of eight hours following the ELAP and the cooldown and depressurization is completed within two hours, it is evident that the plant pressure would remain bounded Figure 7.1-2 of TR-FSE-14-1-P, Revision 1, which shows a limit of 2250 psig for the first 24 hours. Therefore CNP Unit 1 complies with this limitation/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.*

CNP Unit 1 compliance: A constant Westinghouse SHIELD® RCP seal package leak rate of 1 gpm per RCP was assumed in the applicable analysis, CN-FSE-13-13-R, "D.C. Cook Unit 1 and Unit 2 (AEP/AMP) (Reactor Coolant System (RCS)) Inventory Control and Long-Term Subcriticality Analysis to Support the Diverse and Flexible Coping Strategy (FLEX)," Therefore, CNP Unit 1 complies with this limitation/condition.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE CI 3.2.1.2.B

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Report dated August 13, 2014:

Reactor Coolant Pump Seals - The low-leakage seals are not currently credited in the FLEX strategies. Testing and qualification of SHIELD is ongoing. I&M is closely following the re-design of SHIELD and will modify analyses and FLEX strategies, as needed, based on the conclusions of the SHIELD modification program. Confirm FLEX strategies are appropriately modified if low-leakage seals are credited.

Response to NRC Audit Item:

Unit 1 RCP seals were upgraded with Generation 3 SHIELD® equipped low leakage design RCP seals during the Fall 2014 refueling outage. I&M is crediting the installation of the Generation 3 SHIELD® seals in its FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter from J. Davis, NRC, to J. A. Gresham, Westinghouse Electric Company, LLC, dated May 28, 2014. That NRC letter endorses Westinghouse Technical Report TR-FSE-14-1-P and supplemental information provided by Westinghouse letters dated March 19, 2014, and April 22, 2014. The response to ISE CI 3.2.1.2.A describes compliance with the May 28, 2014, NRC letter for CNP Unit 1.

I&M affirms that the installation of low-leakage seals does not adversely affect any other aspects of mitigation strategy or result in significant changes to the strategy that could affect conclusions that the NRC staff has made (e.g., cooldown initiation timing, terminus, etc.). This affirmation is based on CNP site-specific Westinghouse Calculation Note CN-FSE-13-13-R. This Calculation Note provides for FLEX implementation using either standard or Generation 3 SHIELD® equipped (low leakage) Westinghouse RCP seals. As documented in CN-FSE-13-13-R, acceptable analysis results are achieved using the Generation 3 SHIELD® equipped RCP seals. The CNP FLEX implementation strategies credit the Westinghouse RCP seals equipped with Generation 3 SHIELD® RCP seals.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE 3.2.1.2.C

NRC Audit Item as stated in Interim Staff Evaluation dated January 24:

Reactor Coolant Pump Seals - If the seals are changed to the newly designed Generation 3 SHIELD seals, or non-Westinghouse seals, the acceptability of the use of the newly designed Generation 3 SHIELD seals, or non-Westinghouse seals should be addressed. Confirm that the RCP seal leakage rates used in the ELAP analysis have been acceptably justified.

Response to NRC Audit Item:

Unit 1 RCP seals were upgraded with Generation 3 SHIELD® equipped low leakage design RCP seals during the Fall 2014 refueling outage. I&M is crediting the installation of the Generation 3 SHIELD® seals in its FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter from J. Davis, NRC, to J. A. Gresham, Westinghouse Electric Company, LLC, dated May 28, 2014. That NRC letter endorses Westinghouse Technical Report TR-FSE-14-1-P and supplemental information provided by Westinghouse letters dated March 19, 2014, and April 22, 2014. The response to ISE CI 3.2.1.2.A describes compliance with the May 28, 2014, NRC letter for CNP Unit 1. The RCP seal leakage rate of 1 gpm used in the CNP Unit 1 ELAP analysis s in accordance with the May 28, 2014, NRC letter.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE CI 3.2.1.6.B

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Report dated August 13, 2014:

SOE [Sequence of Events] Timeline - Confirm that the revised SOE timeline reflects the change in strategy of not taking credit for low leakage seals and the new site specific boration analysis.

Response to NRC Audit Item:

Unit 1 RCP seals were upgraded with Generation 3 SHIELD® equipped low leakage design RCP seals during the fall 2014 refueling outage. I&M is crediting the installation of the Generation 3 SHIELD® seals in its FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter from J. Davis, NRC, to J. A. Gresham, Westinghouse Electric Company, LLC, dated May 28, 2014. That NRC letter endorses Westinghouse Technical Report TR-FSE-14-1-P and supplemental information provided by Westinghouse letters dated March 19, 2014, and April 22, 2014. The response to ISE CI 3.2.1.2.A describes compliance with the May 28, 2014, NRC letter for CNP Unit 1. The updated SOE timeline provided in the Final Integrated Plan to be issued within 60 days of achieving compliance with NRC Order EA-12-049 for CNP Unit 2 will reflect credit for low leakage seals and the site specific boration analysis.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE CI 3.2.4.6.A

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Report dated August 13, 2014:

Personnel Habitability – Confirm that the FLEX validation process will address personnel accessibility and habitability concerns based on site specific evaluations.

Response to NRC Audit Item:

The limiting habitability concern is local manual operation of Steam Generator (SG) Power Operated Relief Valves (PORV) during the response to an ELAP. Therefore, an evaluation was performed to determine if personnel can safely access the Steam Stop Enclosures and locally manually operate the valves. The evaluation utilized an existing Station Blackout (SBO) calculation, and the guidance in the CNP program that addresses thermal hazards to personnel. The evaluation concluded that conditions in the Steam Stop Enclosure would allow access for manual operation of the SG PORVs.

Additionally, the following factors would also facilitate operator access:

- Once the RCS cool down is completed, the resulting drop in heat load can be expected to improve the environment in the Steam Stop Enclosure.
- Specialized personal protection equipment (PPE), provided as part of the FLEX implementation, would enhance the operator's ability to perform the required function in the expected Steam Stop Enclosure environment.

DETAILS

Required Operator Actions :

In order to establish manual control of the SG PORVs, an operator would enter the Steam Stop Enclosure, climb a permanently installed ladder, and cross a platform. The operator would open the two PORVs in that Steam Stop enclosure and then exit the room. Once the same operation is completed in the opposite Steam Stop enclosure, the initial operation of these valves would be completed and the plant would be in a cooldown mode. At the direction of Control Room (CR) personnel, it may be necessary for the operator to re-enter to room to adjust the PORV position. As with the initial opening of the SG PORVs, this is expected to be accomplished by manual operation of the PORV handwheel(s). Operators are expected to leave the Steam Stop Enclosures when not actually operating the PORV. It is anticipated that initial entry to open the valves, and/or re-entry to adjust the valves would take no longer than 10 minutes for each entry.

Expected Environmental Conditions:

The acceptability of the Steam Stop Enclosure environment for responding to an SBO was evaluated in an existing calculation which considered a 150°F temperature in the enclosure. The calculation had been prepared to identify dominant areas of concern during an SBO and provide reasonable assurance that the necessary equipment could be operated given the expected environmental conditions. The calculation was utilized for the ELAP evaluation because conditions under an SBO bound the ELAP for the first four hours. The primary difference between the SBO event and ELAP, as it relates to SG PORV operation, is that the SBO evaluation is limited to a four hour coping period, while the intermittent SG PORV operation for an ELAP is not limited to a specific duration by the FLEX strategies.

Operators are expected to initiate the ELAP response cooldown within eight hours. However, operator experience on the simulator shows that the plant cooldown is typically started within approximately 30 minutes of the event. Operators must complete the cooldown within 2 hours of initiation. This is well within the 4 hour time frame assessed for the SBO event. Therefore, the calculation which determined the 150°F expected temperature to be acceptable for SBO responses may be considered to bound the expected ELAP response. The acceptability of the 150°F expected temperature for the intermittent SG PORV manual operation is also consistent with the CNP program addressing thermal hazards to personnel.

Personal Protective Equipment

To provide additional assurance that operators can perform the manual SG PORV operations, heat reflective PPE suits would be available to the operators as part of the pre-staged FLEX equipment and tools.

Communication

The operators are expected to exit the Steam Stop Enclosure to communicate with the CR due to noise or to provide adequate radio reception. However, this would not significantly impede manual operation of the SG PORV as needed for the plant cooldown.

Lighting

In addition to the flashlights and head lamps normally available to operators, a supply of dedicated personnel head lamps is maintained in the Shift Manager's office as part of the fire protection program.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE CI 3.2.4.9.A

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Plan dated May 21-2014:

Fuel Consumption Data - Confirm that sufficient fuel is available on-site for operation of FLEX equipment considering the as procured equipment fuel consumption rates and duration of operation before fuel needs to be replenished from off-site sources.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Fuel Consumption Data -The licensee should have procedures which direct the refueling of the portable diesel-powered equipment at appropriate intervals from onsite fuel supplies, and provide direction for obtaining fuel from offsite before the onsite supplies are used up.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Provide overall refueling strategy.

Response to NRC Audit Item:

A fuel consumption study was conducted which estimated the total run time with available on-site fuel to be approximately 351 hours or approximately 14 days. It was assumed that all major on-site diesel powered FLEX equipment was running continuously at full load. This assumption provides significant conservatism because the FLEX strategies do not require all equipment running simultaneously.

The FLEX Support Guideline (FSG) for initial event assessment and flex equipment staging (1-OH P-4027-FSG-5) provides direction to commence diesel driven equipment refueling per the FSG for FLEX equipment refueling operations (12-OHP-4027-FSG-511). The FSG for initial event assessment and flex equipment staging also provides follow up actions to consult with the Site Emergency Director to obtain diesel fuel from offsite sources before the onsite supplies are used up. The FSG for FLEX equipment refueling operations provides direction to 1) move diesel fuel from the underground emergency diesel fuel oil storage tanks to the mobile fuel mules, and 2) from the mobile fuel mules to various FLEX equipment diesel engine fuel tanks.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

ISE CI 3.2.4.10.A

NRC Audit Item as stated in Interim Staff Evaluation dated January 24, 2014 and NRC Audit Plan dated May 21-2014:

Load Shedding - Confirm DC load profile, final load shedding approach including the actions necessary to complete each load shed, the equipment location (or location where the required action needs to be taken), the time to complete each action and identify which functions are lost as a result of shedding each load and any impact on defense-in-depth strategies and redundancy.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

DC Load Shedding -The licensee is revising the procedures for DC load shedding in order to shed additional loads. The NRC will review the final analysis and procedures.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Analysis and procedures for DC load shedding

Response to NRC Audit Item:

Calculation 12-ES-250D-FLEX-001, documenting the direct current (DC) load shed analysis, has been issued along with the FSG for ELAP power management (1-OHP-4027-FSG-4) which performs the DC bus deep load shedding actions consistent with the analysis assumptions. Additionally, labels have been installed in the plant to identify the DC loads to be shed.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

Audit Question (AQ)-36

NRC Audit Item as stated in NRC Audit Plan dated May 21-2014:

Pages 48 and 50 of the integrated plan state that, "It may be desirable to open the Control Room complex doors ... " in different phases of an ELAP event; however, there are no time-sensitive actions specified in the integrated plan which support main Control Room habitability and/or equipment survivability.

Has a plant-specific, thermal hydraulic calculation been performed to determine what the maximum Control Room temperature would be based on the NEI 12-06 conditions? If so, please provide a summary of the calculation, the actions credited in it, and its results concluding that Control Room limits would be maintained in all phases of the ELAP event. If not, please provide the basis and justification for concluding that actions supporting main Control Room ventilation for habitability and/or equipment survivability will not be required on a time-sensitive basis during the course of an ELAP event.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Room Conditions for Personnel Habitability and Equipment Functionality - The licensee was still working on room analyses, which will result in some specific operator actions to be placed in FSG-5. The NRC will review the final analyses and procedural actions.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Room analyses for personnel habitability and equipment functionality.

Response to NRC Audit Item:

Currently, CR instrumentation cabinet doors are opened as a 30 minute time-credited-action in the procedure for responding to loss of all AC power (1-OHP-4023-ECA-0.0). The applicable procedure step is designed to open the cabinet doors as soon as possible. Vital instrument cooling is assured if the cabinet doors are opened within 30 minutes from loss of all AC power. This action ensures adequate instrument cabinet cooling during the four-hour SBO response.

Since the postulated ELAP exceeds four hours, the evaluation described below was performed to determine what actions would be needed to provide adequate CR ventilation during the FLEX Phase 1 and Phase 2 response. Based the evaluation, the FSG for FLEX equipment staging (1-OHP-4027-FSG-501) directs installation of temporary fans similar to the fire protection program based response for loss of CR ventilation. The attached sketch shows the approximate fan locations.

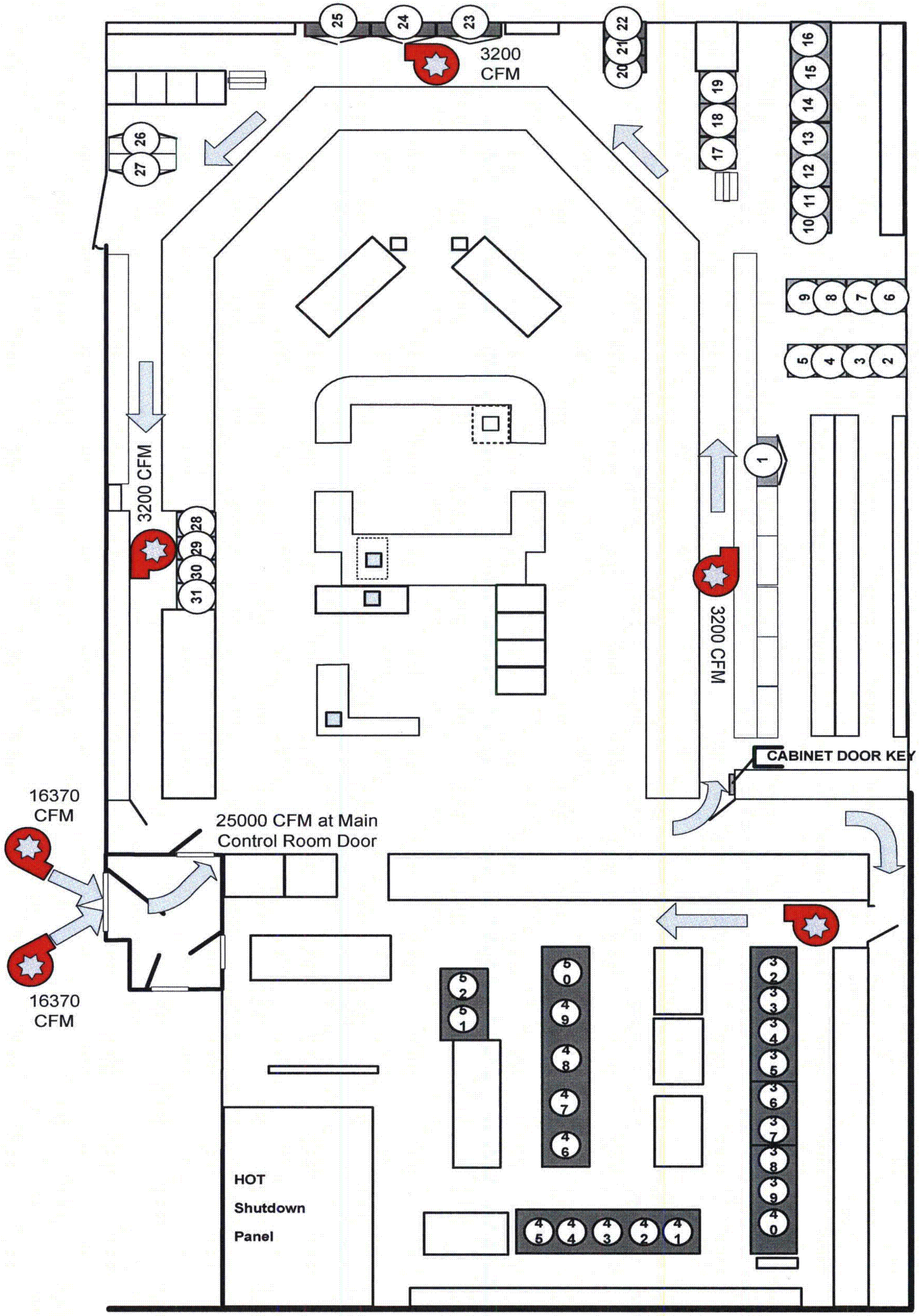
EVALUATION

The acceptability of this action was determined by qualitatively extrapolating the results of the calculation for CR temperatures during a postulated fire event. This calculation used a GOTHIC model and showed a nominal 117°F maximum resultant CR temperature with a similar temporary fan flow rate. The GOTHIC time transient model served to verify that adequate time existed to install and energize the temporary fans. The following simplified steady state computation validates that the CR temperature of 117°F is reasonable given a 104°F turbine building temperature.

Delta T = Q / (cfm x 1.08)

333,000 btu/hr / (23943 x 1.08) = 12.8°F.

From other calculations, the SBO bounding heat load in the CR is 125,343 British Thermal Unit (btu)/ hour (hr), which is much less than the 330,000 btu/hr assumed in the calculation for CR temperatures during a postulated fire event. This SBO heat load reflects equipment powered by the station batteries and is a conservative representation for longer term loss of power events. With this reduced heat load the temperature rise over the temperature of the Turbine Building in which the CR is located is less than 5°F. Therefore, even considering elevated outside temperatures up to 110°F and a corresponding Turbine Building temperature (without heat loads), the CR temperature would remain bounded by the 117°F previously evaluated in the calculation for CR temperatures during a postulated fire event.


3200
CFM
3200 CFM
3200 CFM
CABINET DOOR KEY
16370
CFM
25000 CFM at Main
Control Room Door
16370
CFM
HOT
Shutdown
Panel

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

AQ 53

NRC Audit Item as stated in NRC Audit Plan dated May 21-2014:

Discuss the habitability conditions for operators who have to operate the steam generator power-operated relief valves (PORVs).

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Personnel Habitability for Operation of Steam Generator (SG) Power Operated Relief Valves (PORVs) -The licensee is developing an analysis of habitability requirements for the local operation of the PORVs. SG PORVs. The NRC will review the final analysis.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Provide engineering evaluation to demonstrate that personnel can safely access the Steam Stop Enclosures and locally operate the SG PORVs.

Response to NRC Audit Item:

See the response to ISE CI 3.2.4.6.A for a discussion of habitability issues for operation of the SG PORVs.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

SE #1

NRC Audit Item as stated in NRC Audit Plan dated May 21-2014:

1. (RCS Venting) The generic analysis in WCAP-17601-P strictly addressed ELAP coping time without consideration of the actions directed by a site's mitigating strategies. WCAP-17792-P extends these analytical results through explicit consideration of mitigating strategies involving RCS makeup and boration. In support of the RCS makeup and boration strategies proposed therein, a generic recommendation is made that Pressurized-Water Reactors vent the RCS while makeup is being provided. Please provide the following information in regard to this topic:

a. Will the mitigating strategy include venting of the RCS?

b. If so, please provide the following information:

i. The vent path to be used and the means for its opening and closure.

ii. The criteria for opening the vent path.

iii. The criteria for closing the vent path.

iv. Clarification as to whether the vent path could experience two-phase or single phase liquid flow during an ELAP. If two-phase or liquid flow is a possibility, please clarify whether the vent path is designed to ensure isolation capability after relieving two-phase or liquid flow.

v. If relief of two-phase or liquid flow is to be avoided, please discuss the availability of instrumentation or other means that would ensure that the vent path is isolated prior to departing from single-phase steam flow.

vi. If a pressurizer PORV is to be used for RCS venting, please clarify whether the associated block valve would be available (or the timeline by which it could be repowered) in the case that the PORV were to stick open. If applicable, please further explain why opening the pressurizer PORV is justified under ELAP conditions if the associated block valve would not be available.

vii. If a pressurizer PORV is to be used for RCS venting, please clarify whether FLEX RCS makeup pumps and FLEX steam generator makeup pumps will both be available prior to opening the PORV. If they will not both be available, please provide justification.

c. If RCS venting will not be used, please provide the following information:

i. The expected RCS temperature and pressure after the necessary quantity of borated makeup has been added to an unvented RCS.

ii. Adequate justification that the potential impacts of unvented makeup will not adversely affect the proposed mitigating strategy (e.g., FLEX pump discharge pressures will not be challenged, plant will not reach water solid condition, adequate boric acid can be injected, increased RCS leakage will not adversely affect the integrated plan timeline, etc.).]

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

RCS venting in support of mitigating strategies for RCS makeup and boration.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

The generic FSG-8 issued by the PWROG states that the vessel head vent should be used before using the pressurizer PORV, but the CNP FSG-8 switches the order. As the vessel head vents were designed to be used in this type of situation, justify this deviation from the generic guidelines. Also, ECA-0.0 does not appear to require injection of water into the RCS prior to reaching reflux cooling. The intent is to avoid reflux cooling if possible.

Response to NRC Audit Item:

CNP site-specific calculation CN-FSE-13-13-R shows that Unit 1 is not expected to need venting/letdown of the RCS to complete boration during an ELAP response. Although not credited in the calculation, the FSG for Alternate RCS Boration (1-OHP-4027-FSG-8) provides instructions for use of the reactor vessel head vents as the primary method of RCS venting and the pressurizer PORV as the contingency method in accordance with Westinghouse vendor-specific guidelines.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

SE #2

NRC Audit Item as stated in NRC Audit Plan dated May 21-2014:

NSAL-14-1 - On February 10, 2014, Westinghouse issued Nuclear Safety Advisory Letter (NSAL)-14-1, informing licensees of plants with standard Westinghouse RCP seals that 21 gpm may not be a conservative leakage rate for ELAP analysis. This value had been previously used in the ELAP analysis referenced by many Westinghouse Pressurized-Water Reactors, including the generic reference analysis in WCAP-17601-P. Therefore, please clarify whether the assumption of 21 gpm of seal leakage per RCP (at 550 degrees F, 2250 psia) remains valid in light of the issues identified in NSAL-14-1. In so doing, please identify the specifics of the seal leak off line design and #1 seal faceplate material relative to the categories in NSAL-14-1 and identify the corresponding presumed leakage rate from NSAL-14-1 that is deemed applicable.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Resolution of Westinghouse Nuclear Safety Advisory Letter (NSAL) 14-1 - NSAL -14-1 indicates there may be higher leakage from the reactor coolant pump (RCP) seals during an extended loss of ac power (ELAP) than was previously analyzed. The license is working to resolve this issue. The NRC will review the final resolution.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Resolution of NSAL 14-1. See also CI 3.2.1.2.A.

Response to NRC Audit Item:

Unit 1 RCP seals were upgraded with Generation 3 SHIELD® equipped low leakage design RCP seals during the Fall 2014 refueling outage. I&M is crediting the installation of the Generation 3 SHIELD® seals in its FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter from J. Davis, NRC, to J. A. Gresham, Westinghouse Electric Company, LLC, dated May 28, 2014. That NRC letter endorses Westinghouse Technical Report TR-FSE-14-1-P and supplemental information provided by Westinghouse letters dated March 19, 2014, and April 22, 2014. The response to ISE CI 3.2.1.2.A describes compliance with the May 28, 2014, NRC letter for CNP Unit 1. The RCP seal leakage rate of 1 gpm used in the CNP Unit 1 ELAP analysis (CN-FSE-13-13-R) is in accordance with the May 28, 2014, NRC letter.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

SE #5

NRC Audit Item as stated in NRC Audit Plan dated May 21-2014:

Please provide adequate basis that calculations performed with the NOTRUMP code (e.g., those in WCAP-17601-P, WCAP-17792-P) are adequate to demonstrate that criteria associated with the analysis of an ELAP event (e.g., avoidance of reflux cooling, promotion of boric acid mixing) are satisfied. NRC staff confirmatory analysis suggests that the need for implementing certain mitigating strategies for providing core cooling and adequate shutdown margin may occur sooner than predicted in NOTRUMP simulations.

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Accuracy of the NOTRUMP Computer Code - Westinghouse used the NOTRUMP computer code to develop certain timelines for operator actions in an ELAP event (see WCAP-17601-P for example). NRC simulations using the TRACE code indicate some differences, which may be significant enough to affect the timeline for operator actions. The pressurized water reactor owner's group (PWROG) is working with the NRC on a resolution, which may be applicable to all PWRs.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

PWROG project PA-ASC-1274 should provide resolution on the accuracy of the NOTRUMP code and the ability to predict the time that reflux cooling starts.

Response to NRC Audit Item:

PWROG project PA-ASC-1274 was established to provide resolution of the NRC questions/concerns regarding the accuracy of the NOTRUMP computer code and the ability to appropriately predict the time that reflux cooling starts following a postulated ELAP event. The PWROG project deliverables were formally issued via PWROG project letter OG-14-339, 'PWR Owners Group, Transmittal of PWROG-14064-P, Revision 0, "Application of NOTRUMP Code Results for PWRs in Extended Loss of AC Power Circumstances," For Information Only (PA-ASC-1274)," dated September 26, 2014.

This project letter had a Position Paper attached as PWROG-14064-P, Revision 0, "Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances, PWROG ASC Committee, PA-ASC-1274," dated September 2014.

The summary paragraph from the Position Paper (PWROG-14064-P) pertaining to NOTRUMP & TRACE computer code differences states:

"To summarize, the comparison of results from the NOTRUMP and TRACE computer codes for the parameters of interest show that the NOTRUMP predicted results agree well or are conservative with respect to the TRACE predicted results. The comparison shows that NOTRUMP provides a conservative estimate of the required time when the primary make-up pumps are required for an ELAP event as compared to TRACE. Therefore, it is concluded that NOTRUMP is acceptable for simulation of the ELAP event within the constraints listed herein with regards to reflux cooling and boron mixing."

I&M therefore considers that PWROG-14064-P resolves the NRC FLEX concerns regarding differences between the NOTRUMP and TRACE computer codes pertaining to the onset of reflux cooling.

Regarding the discussion in the Position Paper about the application of restrictions on the use of NOTRUMP, it is noteworthy that the content of such statements pertains to the use of generic analyses work documented in WCAP-17601 and WCAP-17792 as the sole basis for a plant's FLEX strategies. This is not the case for CNP since plant-specific analyses have been performed (e.g., CN-FSE-13-13-R) showing RCP make-up analyses with and without the SHIELD® design RCP seals (i.e., "low leakage" or "shutdown" seals). Thus, the restrictions discussed in the Position Paper such as the 17 hour RCS make-up implementation requirement presented in Table 1 on Page 4-3 of the Position Paper does not apply to CNP for two reasons:

(1) CNP Unit 1 RCP seals were upgraded with Generation 3 SHIELD® equipped low leakage design RCP seals during the Fall 2014 refueling outage. I&M is crediting the installation of the Generation 3 SHIELD® seals in its FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter from J. Davis, NRC, to J. A. Gresham, Westinghouse Electric Company, LLC, dated May 28, 2014. That NRC letter endorses Westinghouse Technical Report TR-FSE-14-1-P and supplemental information provided by Westinghouse letters dated March 19, 2014, and April 22, 2014. and

(2) The CNP-specific analysis (CN-FSE-13-13-R) notes an RCS make-up time (if Generation 3 SHIELD RCP seals are assumed) to be greater than 72 hrs. for the two-phase natural circulation case.

The CNP-specific analysis also states that RCS make-up is required to commence no later than 16 hours, if Generation 3 SHIELD® RCP seals are assumed, to ensure single phase natural circulation is maintained, and full RCS Boration must be completed within 24 hours of the ELAP. The strategy for CNP Unit 1 is to ensure single phase natural circulation.

NRC AUDIT ITEM RESPONSE

NRC Audit Item Reference:

SE#8

"Item Description" as stated in NRC Audit Report dated August 13, 2014:

Validation and Verification-The licensee was developing procedures for validation and verification of the revised plant procedures and the new FSG's, which are different from the NEI guidance in this area. The NRC will review those procedures.

"Licensee Input Needed" as stated in NRC Audit Report dated August 13, 2014:

Validation and Verification procedures which also address human factors concerns.

Response to NRC Audit Item:

The CNP procedure for FSG maintenance (PMP-4027-FSG-001) provides verification and validation guidance for addressing human factor concerns when development or revising FSGs. I&M has conducted a review which documented consistency of the CNP Unit 1 FLEX validation actions with those prescribed in the NEI document titled "FLEX Validation Process." The review also confirmed that the CNP Unit 1 FLEX strategies are feasible and can be executed with the constraints identified in the current FLEX Mitigation Strategies Summary. I&M commits to follow the NEI guidance for validation of future FSGs and changes.

OIP OPEN ITEM CLOSURE

OIP Open Item 27

Perform calculation to verify time required to establish flow to the RCS in MODE 6 with the Reactor Cavity filled.

OIP Closure

A calculation has been performed to determine the minimum available time for establishing alternate makeup strategies, such as Refueling Water Storage Tank gravity drain, accumulator drain, and/or deploying FLEX Phase 2 equipment, under an assumed ELAP event with the refueling cavity flooded. The calculation determined the minimum available time to be approximately 49 hours.

OIP Open Item 29

Fabricate a tool to provide large volume Phase 3 raw water tie-in to Essential Service Water (ESW) supporting component cooling water cooling for Residual Heat Removal (RHR).

OIP Closure

The tool has been fabricated. The tool consists of a temporary ESW strainer lid. If an ESW pump is not available during Phase 3, the West ESW pump discharge strainer lid would be removed and replaced with the temporary lid. The temporary lid is equipped with hose connections to accept discharge of the National Strategic Alliance for FLEX Emergency Response Center (NSRC) large raw water pumps. Water would be pumped from the circulating water Intake forebay to the NSRC high volume Raw Water Pump suction using two NSRC high flow, low discharge pressure, lift pumps. The ESW strainer lid connection can supply sufficient raw water to the West ESW system discharge piping to provide Component Cooling Water Heat Exchanger cooling supporting RHR and CR Ventilation System cooling. The tool is stored, along with other FLEX equipment, in the FLEX storage building which is protected from, and accessible following, a Beyond Design Basis External Event. NEI 12-06 does not require redundant Phase 3 equipment, therefore there is single tool. The tool is part of the onsite inventory, and is listed on the FSG for FLEX equipment inventory and checks (12-OHP-5030-FSG-523).