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October 11, 2016

AEP-NRC-2016-83
10 CFR 50.91(a)(5)

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

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
Emergency License Amendment Request for One-Time Extension of Completion Time for Inoperable AC Source - Operating

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Unit 1, proposes to amend Facility Operating License DPR-58. I&M proposes a one-time extension of the Completion Time for Technical Specification (TS) 3.8.1, "AC Sources – Operating." The proposed amendment is being requested on an emergency basis pursuant to 10 CFR 50.91(a)(5).

TS 3.8.1 Required Action A.3 requires that, with the unit in Modes 1 through 4, one inoperable required offsite circuit be restored to operable status within 72 hours. If the inoperable offsite circuit is not restored to operable status within 72 hours, TS 3.8.1 Condition G requires that the unit be in Mode 3 within 6 hours and in Mode 5 within 36 hours. As detailed in Enclosure 2 to this letter, on October 7, 2016, at approximately 2315 hours, a Taylor Machine Works Forklift was traveling along the plant service road outside of the protected area, the left rear wheel broke through a section of a below grade Plastibeton cable raceway. This raceway contains the 34.5Kv cables that feed the Train 'B' reserve feed to both Unit 1 and Unit 2. There was no load on the forks at the time and there was no obvious damage to the cables and no injuries. However, there is the likelihood that as a result of the tire penetrating the Plastibeton and coming to rest on the cables, that once the forklift and failed Plastibeton are removed, there will be some cable damage that needs to be repaired. The best estimate schedule for deenergizing the run of 34.5Kv cable, removing the forklift, repairing damaged cables, and performing post-maintenance work is 71 hours. Based on the uncertainty associated with these time estimates it is likely that this will put actions to complete the repairs over the 72 hour TS 3.8.1 Required Action A.3 Completion Time. Based on the I&M time estimates for replacement of the cable, and the fact that this will be a first time evolution, I&M is requesting an additional 28 hours, making the completion time 100 hours.

Enclosure 1 to this letter provides an affirmation affidavit pertaining to the proposed amendment. Enclosure 2 provides a detailed description and safety analysis to support the proposed amendment, including justification for approving the amendment on an emergency basis, an evaluation of significant hazards considerations pursuant to 10 CFR 50.92(c), and an environmental assessment. Enclosure 3 provides Probabilistic Risk Assessment (PRA) Technical Adequacy.

ADD1
NRR

Enclosure 4 provides PRA calculation, Calculation PRA-QNT-007, Calculation of Regulatory Guide 1.177 Risk Parameters for Potential One-Time Emergency Technical Specification Completion Time Change for Unit 1 and Unit 2 Train B Reserve Feed. Enclosure 5 provides the applicable license page marked to the proposed change. Enclosure 6 provides the license page with the proposed changes incorporated.

The activities to address the identified condition are currently scheduled to start at 0400 hours on October 12, 2016. The start of these activities requires entry into TS 3.8.1, the A. 3 Required Action which the associated completion time is being requested to be changed by this license amendment request. I&M requests approval of the proposed amendment by 1600 hours on October 14, 2016, to allow for the timely correction of the condition and preclude an unnecessary shutdown of Unit 1.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

There are no new regulatory commitments in this letter. Should you have any questions, please contact Mr. Michael Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Quinton S. Lies

Q. Shane Lies
Site Vice President

DB/rdw

Enclosures:

1. Affirmation.
2. License Amendment Request for One-Time Extension of Completion Time for an Inoperable AC Source – Operating
3. Probabilistic Risk Assessment Technical Adequacy
4. Calculation PRA-QNT-007, Calculation of Regulatory Guide 1.177 Risk Parameters for Potential One-Time Emergency Technical Specification Completion Time Change for Unit 1 and Unit 2 Train B Reserve Feed
5. License Page - Marked to Show Proposed Change
6. License Page - Changes Incorporated

c: R. J. Ancona – MPSC
A. Dietrich, NRC Washington, D.C.
MDEQ- RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
A. J. Williamson – AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2016-83

AFFIRMATION

I, Q. Shane Lies, 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

Quinton S. Lies

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 11 DAY OF October, 2016

Danielle Burgoyne
Notary Public

My Commission Expires 04-04-2018

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2018
Acting in the County of Berrien

Enclosure 2 to AEP-NRC-2016-83

LICENSE AMENDMENT REQUEST FOR ONE-TIME EXTENSION OF COMPLETION TIME FOR AN INOPERABLE AC SOURCE - OPERATING

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, proposes to amend Facility Operating License DPR-58. I&M proposes a one-time extension of the Technical Specification (TS) Completion Time for an inoperable AC Electrical Source. The proposed amendment is being requested on an emergency basis pursuant to 10 CFR 50.91(a)(5).

2.0 PROPOSED CHANGE

The proposed change would add the following Footnote to the 72-hour Completion Time for TS 3.8.1 Condition A.3:

> "For Train B only, the Completion Time that Train B can be inoperable as specified by Required Action A.3 may be extended beyond the "72 hours" up to "100 hours," to support repair and restoration of the Train B Reserve Feed. Upon completion of the repair and restoration associated with this event which occurred on October 7, 2016, this footnote is no longer applicable."

3.0 BACKGROUND

AC Electrical Power Distribution System Description

As shown on the attached sketches, the onsite alternating current (AC) electric power distribution system for each unit contains four, 4160V (4.16 kV) non-safety-related buses designated 1A, 1B, 1C, and 1D for Unit 1 and 2A, 2B, 2C, and 2D for Unit 2. These buses are referred to as the "RCP" buses because they power the reactor coolant pumps. Each of the non-safety-related RCP buses feed a downstream safety-related 4.16 kV bus. These safety-related buses are designated T11A, T11B, T11C, and T11D for Unit 1 and T21A, T21B, T21C, and T21D for Unit 2. These buses are referred to as the "T" buses. With the main generator on-line, the RCP buses are normally fed from the Unit Auxiliary Transformers (UATs), which receive power from the main generator.

Upon a trip of the main generator, the station auxiliaries are automatically fast transferred to the preferred offsite power source (i.e., to reserve auxiliary transformers (RATs) TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line. The ESF Loads are sequenced onto the RATs, under accident conditions, using the same timing relays and sequence as used for the EDG sequencing. The RATs supply the reserve auxiliary power for both units.

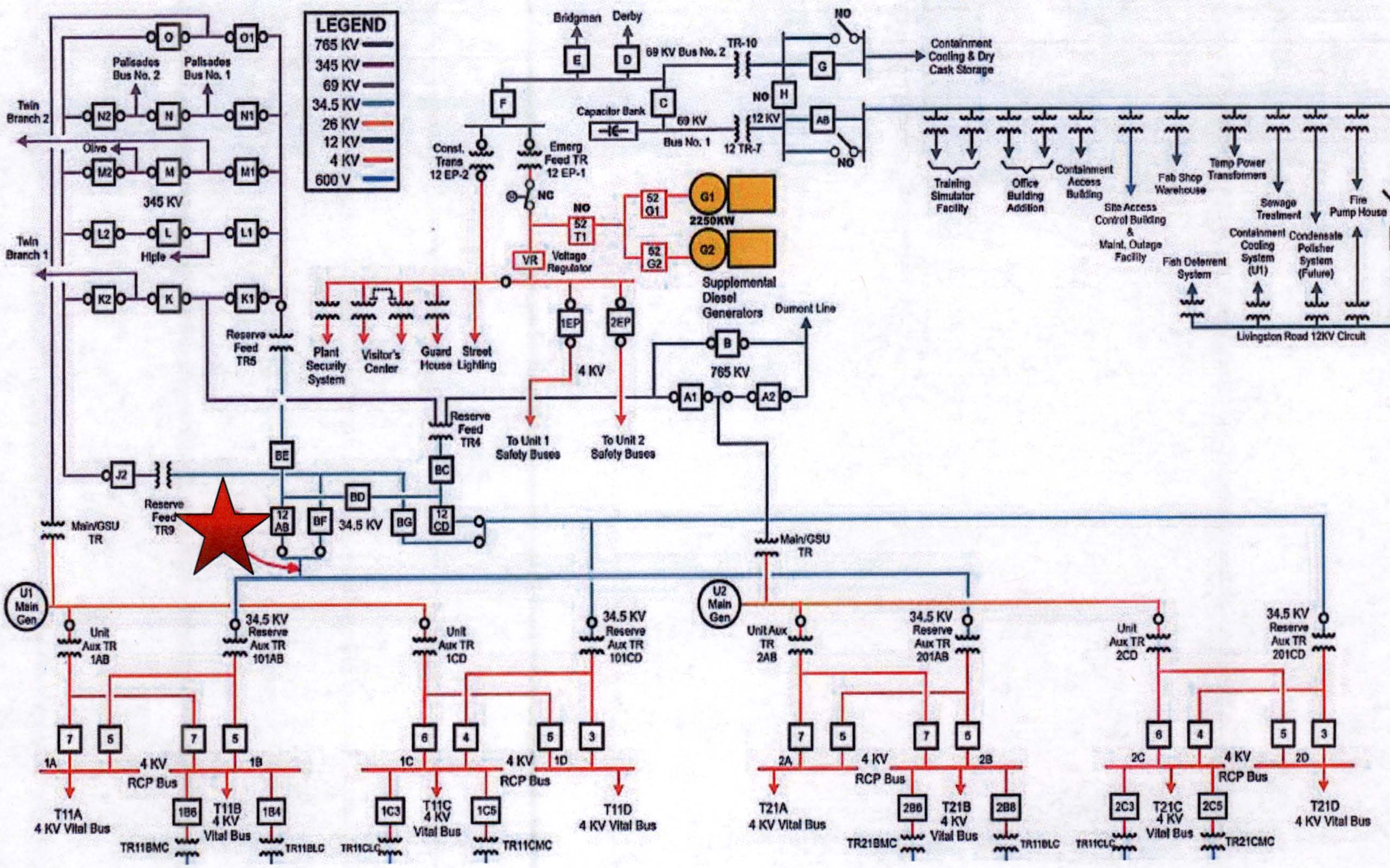
The preferred offsite power source for both units can be arranged so that transformer No. 4 or transformer No. 9 can supply reserve auxiliary transformers (RATs) TR101CD and TR201CD

and transformer No. 5, or transformer No. 9 supplies TR101AB and TR201AB. Under certain plant conditions, it is possible for transformer No. 4, transformer No. 5, or transformer No. 9 to feed the entire plant auxiliary load.

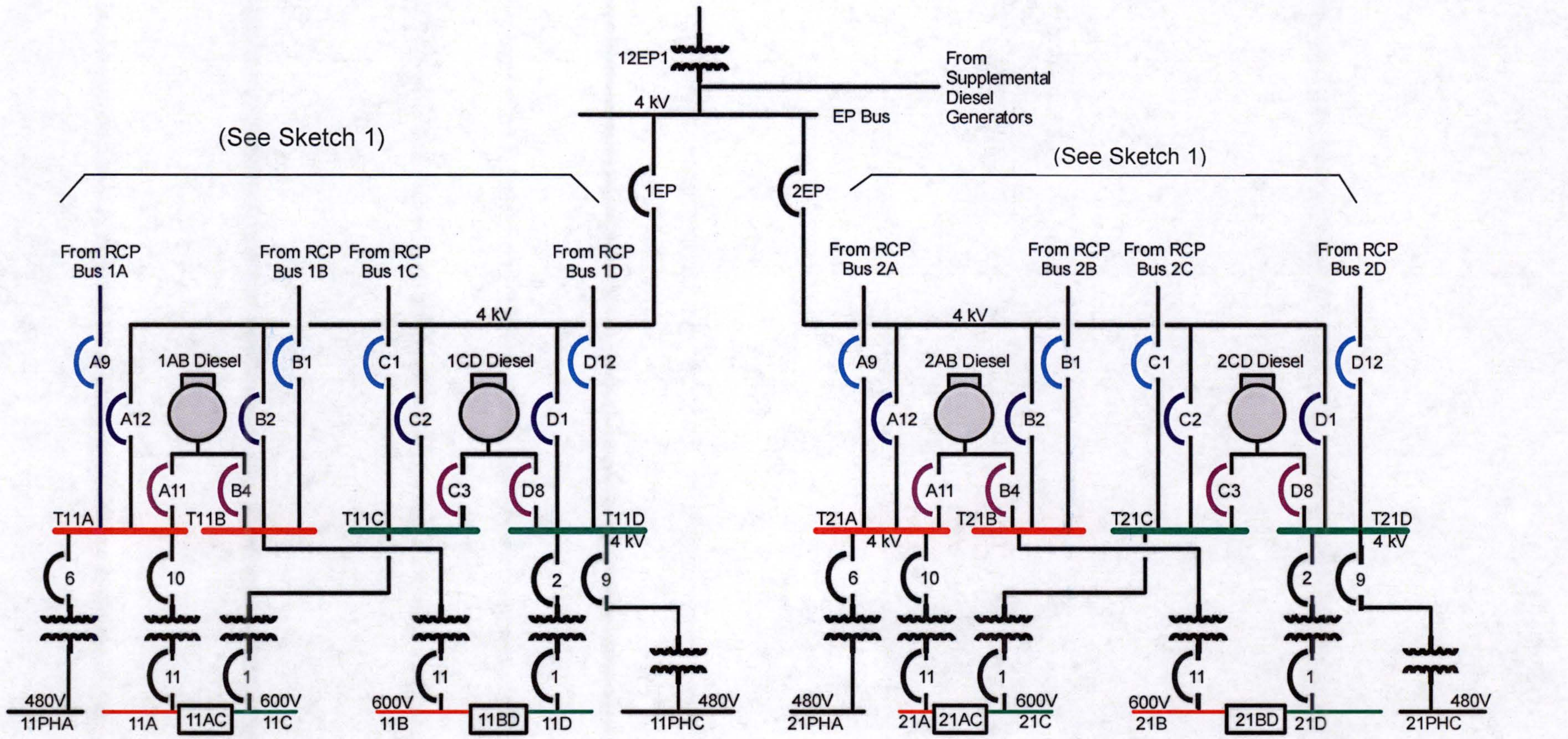
The other qualified circuit required to be operable by TS LCO 3.8.1.a is the alternate offsite circuit. The alternate qualified offsite circuit consists of a 69/4.16 kV transformer (TR12EP-1), the cabling and switches to a 4.16 kV bus, designated as the EP Bus, which supplies breakers 1EP and 2EP, and the cabling, switches, and breakers to the T buses. Connection of the T buses to transformer 12EP1 requires manual switch operations in the control room. The alternate offsite power source has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other. The T buses can also be powered from the emergency diesel generators (EDGs). TS LCO 3.8.1.a requires two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System.

An additional independent onsite standby AC power source consisting of two supplemental diesel generators (SDGs) is provided to automatically supply power to the EP bus, which is normally supplied by the alternate qualified offsite circuit and can be manually aligned to directly supply the T buses.

SKETCH 1


LEGEND
765 KV
345 KV
69 KV
34.5 KV
26 KV
12 KV
4 KV
600 V
Twin Branch 2
Twin Branch 1
Palisades Bus No. 2
Palisades Bus No. 1
Olive
Hiple
345 KV
Reserve Feed TR5
Reserve Feed TR4
Reserve Feed TR9
Main/GSU TR
Bridgman
Derby
69 KV Bus No. 2
TR-10
Capacitor Bank
69 KV Bus No. 1
12 TR-7
Containment Cooling & Dry Cask Storage
Const. Trans 12 EP-2
Emerg Feed TR 12 EP-1
Voltage Regulator
2250KW
Supplemental Diesel Generators
Dumont Line
765 KV
Plant Security System
Visitor's Center
Guard House
Street Lighting
4 KV
To Unit 1 Safety Buses
To Unit 2 Safety Buses
Training Simulator Facility
Office Building Addition
Containment Access Building
Site Access Control Building & Maint. Outage Facility
Fab Shop Warehouse
Temp Power Transformers
Fish Deterrent System
Sewage Treatment
Containment Cooling System (U1)
Condensate Polisher System (Future)
Fire Pump House
Livingston Road 12KV Circuit
34.5 KV
U1 Main Gen
U2 Main Gen
Unit Aux TR 1AB
34.5 KV Reserve Aux TR 101AB
Unit Aux TR 1CD
34.5 KV Reserve Aux TR 101CD
Unit Aux TR 2AB
34.5 KV Reserve Aux TR 201AB
Unit Aux TR 2CD
34.5 KV Reserve Aux TR 201CD
4 KV RCP Bus
T11A 4 KV Vital Bus
T11B 4 KV Vital Bus
T11C 4 KV Vital Bus
T11D 4 KV Vital Bus
T21A 4 KV Vital Bus
T21B 4 KV Vital Bus
T21C 4 KV Vital Bus
T21D 4 KV Vital Bus
TR11BMC
TR11BLC
TR11CLC
TR11CMC
TR21BMC
TR21CMC

SKETCH 2


12EP1
From Supplemental Diesel Generators
4 kV
EP Bus
(See Sketch 1)
1EP
2EP
(See Sketch 1)
From RCP Bus 1A
From RCP Bus 1B
From RCP Bus 1C
From RCP Bus 1D
From RCP Bus 2A
From RCP Bus 2B
From RCP Bus 2C
From RCP Bus 2D
4 kV
4 kV
A9
1AB Diesel
B1
C1
1CD Diesel
D12
A9
2AB Diesel
B1
C1
2CD Diesel
D12
A12
B2
C2
D1
A12
B2
C2
D1
A11
B4
C3
D8
A11
B4
C3
D8
T11A
T11B
T11C
T11D
4 kV
T21A
4 kV
T21B
T21C
T21D
4 kV
6
10
2
9
6
10
2
9
11
1
11
1
11
1
11
1
480V
11PHA
11A
11AC
600V
11C
600V
11B
11BD
11D
480V
11PHC
480V
21PHA
21A
21AC
600V
21C
600V
21B
21BD
21D
480V
21PHC

Description of Events

On October 7, 2016, at approximately 2315 hours, as a Taylor Machine Works Forklift was being driven along the plant service road outside of the protected area, the left rear wheel broke through a below grade section of a Plastibeton cable raceway. This raceway is situated along the shoulder of the service road and contains the 34.5Kv cables that are immediately downstream of the Circuit Breaker "12 AB", which is the common high side feed to the Unit 1 and Unit 2 RATs (TR101AB and TR201AB) (notated by a large star on Sketch 1). There was no load on the forks at the time. There is no indication of cable damage based on visual inspection and plant parameters and there were no injuries.

In the intervening time since the incident, I&M has been investigating to determine the cause of the event, evaluating the damage, preparing repair planning estimates, obtaining replacement parts, ensuring vendor resources, and conducting a risk analysis. Detailed scheduling of the replacement activities indicated that the cable could be replaced and the new cable declared operable within 100 hours.

Cable repair and replacement activities are scheduled for 71 hours. However, because the actual condition of the cable is unknown and the fact that this is a first-time evolution, the repair time could take longer.

TS 3.8.1 Required Action A.3 requires that one inoperable required offsite circuit be restored to operable status within 72 hours. If the inoperable offsite circuit is not restored within 72 hours, TS 3.8.1, Condition G requires that the unit be in Mode 3 within 6 hours and in Mode 5 in 36 hours.

Reason the Amendment is Requested on an Emergency Basis

Regulation 10 CFR 50.91(a)(5) states that where the U. S. Nuclear Regulatory Commission (NRC) finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation, or of increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. The regulation requires that a licensee requesting an emergency amendment explain why the emergency situation occurred and why the licensee could not avoid the situation. As explained below, an emergency amendment is needed to effect timely resolution of the issue and to preclude an unnecessary plant shutdown, and I&M could not have reasonably avoided the situation or made timely application for an amendment.

Reason Emergency Situation Has Occurred

The emergency situation was initiated by the Taylor Machine Works Forklift unexpectedly dropping through the Plastibeton cable raceway onto the 34.5Kv cables. Detailed scheduling, and accounting for uncertainties, indicates that the cable replacement could be accomplished within 100 hours, which is greater than the TS 3.8.1 Condition A.3 Completion Time of 72 hours. I&M estimates that completion of cable replacement would likely exceed the 72 hours Completion Time of TS 3.8.1 Required Action A.3 if repair of all 3 phases of 34.5kV cable is

determined necessary, which would require that the unit be shutdown. Neither a routine nor an exigent amendment would allow for a timely resolution of the issue. Therefore, an emergency amendment is requested to effect repairs in a timely manner and to preclude a shutdown.

Reason the Situation Could Not Have Been Avoided

The failure of the Plastibeton cable raceway could not have been foreseen. The Plastibeton cable raceway is designed to be installed below grade and the forklift had been successful in routinely driving this path. Any complications in replacing damaged cable also cannot be reasonably foreseen, as this will be an infrequent evolution, if it becomes necessary, for I&M to splice these 34.5Kv cables. Due to the uncertainty of the repairs that will be required, there are three restoration schedules. The worst case schedule with a margin for uncertainties may require 100 hours for completion. Personnel are aware of the activities and durations needed to accomplish a routine replacement.

The condition of the cable won't be known until the forklift tire and damaged Plastibeton pieces are removed to reveal the cable. Once the cable in the affected section of the cable raceway is revealed, an assessment for any damage can be conducted. Visual inspection and tan-delta testing will be conducted, which could lead to three possible scenarios; no damage, minor damage to the cable outer jacket, or significant damage requiring the cable(s) to be replaced.

The three possible outcomes and their time frames are described below, and the time estimates are for known activities. The time estimates do not reflect contingency activities:

- Visual inspection and Tan-delta cable diagnostic testing shows no repair is needed for the cable and the raceway will be repaired – this evolution is scheduled for 27 hours.
- Visual inspection and Tan-delta testing indicates damage to outer jacket and the need to perform a repair of the outer jacket. The time estimate for this scenario is 33 hours. Once the damage on the outer jacket has been repaired, Tan-delta testing will be performed to ensure cable integrity. Tan-delta testing could indicate the need to replace the cable.
- Either the visual inspection or tan-delta testing leads to removal of the damaged cable sections for all three phases, replacing the damaged section(s), and performing up to six splices. After the splicing is complete a second Tan-delta test will be performed to verify the integrity of the "repaired" cable. This scenario is estimated to require 71 hours for completion.

The worst case scenario, occurring if the visual inspection or the Tan-delta testing leads us to cable replacement and splicing, requires three steps in the project schedule: 1) cut out and remove the damaged cable for all three phases, 2) replace all three cables and splice in the new cables, 3) then retest with Tan-delta testing to verify cable integrity with the new "spliced" cables.

The reason for requesting 100 hours for completion is due to the unknown nature of any delays that may occur in the planned activities. A few examples of the delays that could be experienced are: adverse weather, tooling issues, repair equipment failures, stuck material that

reveals itself to be excessively difficult to remove from the raceway. A delay greater than one hour of any type would therefore cause the exceedance of the 72-hour action statement. Additionally, the language of the footnote expires the action statement at completion of any repairs and declaration of operability such that the entire 100-hour Completion Time would not be utilized if not required.

Considering that the cable was energized at the time of the event and power was never lost, interrupted, or showed any indication of degradation based on observable plant parameters, there is reasonable assurance that the cable is currently intact. However, because the level of damage, if any, is unknown, the extent of repair, if any, required cannot be ascertained until the forklift and damaged material are removed from the cable raceway.

I&M therefore considers that there is sufficient justification for requesting the proposed license amendment on an emergency basis.

4.0 TECHNICAL ANALYSIS

The proposed amendment to allow a one-time extension of the TS 3.8.1 Condition A.3 Completion Time for the current inoperability of the Unit 1 AC Power Source is supported by a quantitative risk evaluation.

4.1 Tier 1: Probabilistic Risk Assessment (PRA) Capability and Insights

Technical Adequacy of the PRA model

A full discussion and justification of the PRA technical adequacy is provided in Enclosure 3.

The CNP PRA model is generally robust and suitable to support this amendment. Specific issues identified by the recent peer review that may have a significant impact on the model have been addressed to reduce or eliminate their impact on the results. The ongoing PRA maintenance and update activities associated with the CNP PRA program ensure that the PRA models represent the as-built, as-operated plant moving forward. Therefore, the CNP PRA model has the technical adequacy required to support the amendment.

Quantitative Analysis

The PRA risk impact of operation with Unit 1 and Unit 2 Train B Reserve Feed unavailable can be quantitatively estimated using the CNP Full Power Internal Events (FPIE) model of record and a Fire PRA model which has been updated for this application as describe in Enclosure 3, Section 4.1. The full analysis is provided in Enclosure 3 and is summarized in this section.

The risk assessment is quantified by analyzing the risk of the current plant configuration and then subtracting the risk from a base case (nominal equipment unavailability) core damage frequency (CDF) and large early release frequency (LERF) for both the FPIE and Fire PRA models. Since Unit 2 is currently offline for a scheduled refueling outage, only the risk from

Unit 1 is analyzed in this assessment. The following assumptions are used for the current plant configuration:

- Unit 2 is currently in a scheduled refueling outage. The majority of the mitigating systems on Unit 2 Train B are currently unavailable and will remain unavailable during the duration of the Unit 1 and Unit 2 Train B Reserve Feed outage. Although it is possible that some Unit 2 Train B equipment will be available during the outage, it will not be credited for this analysis.

- The Unit 1 East – Unit 2 West ESW crosstie is scheduled to be closed for outage work on the Unit 2 West ESW pump. This crosstie valve will be assumed to be closed for this risk analysis.

- No additional Unit 1 equipment will be unavailable for the duration of the Unit 1 and Unit 2 Train B Reserve Feed outage.

- No surveillance testing will occur on Unit 1 PRA credited equipment.

The following compensatory measures are explicitly accounted for in the risk quantification assumed to be in effect for the duration of the Unit 1 and Unit 2 Train B Reserve Feed outage. Additional compensatory actions, which are not quantitatively considered, are listed in Section 4.3:

- To the extent practicable and controllable, no other work is assumed to be undertaken that could jeopardize operation of Unit 1. For example, main turbine valve testing or similar activities, or maintenance work on BOP components that have potential to initiate a unit trip, are assumed to be avoided while repair of Unit 1 and Unit 2 Train B Reserve Feed is in progress.

- The following Unit 2 Train A equipment is available

    - Unit 2 East Motor-Drive Auxiliary Feedwater (AFW) Pump
    - Unit 2 East Component Cooling Water (CCW) Pump
    - Unit 2 East Charging Pump
    - Unit 2 East Essential Service Water (ESW) Pump

- All PRA-related equipment for Unit 1 is available.

The compensatory measures are accounted for in the quantitative risk analysis by not assuming any unavailability of the protected or guarded equipment listed above. Test and maintenance is only considered for those systems identified above that were determined to be unavailable for the LCO period. Fire watch tours and maintenance restrictions are not assumed to modify the likelihood of any fire or internal initiating events.

Since the plant electrical system normally receives power from the main generator, the units do not automatically trip off if offsite power is lost. For this reason, the likelihood of initiating events is not adjusted for the risk analysis. The loss of the Train A reserve auxiliary transformers is

already accounted for in the FPIE model as a random failure. Fire events are not considered to be any more likely due to the Train B Reserve Feed Outage.

FPIE Results

The FPIE PRA model was re-quantified with the configuration information as discussed above. Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Probability (ICLERP) are calculated for the full (100 hours) duration of the LCO. The full calculation, which follows the approach of RG 1.177, is provided in Enclosure 4.

FPIE Case	Internal Events CDF (/yr)	Internal Events LERF (/yr)	ICCDP	ICLERP
FPIE Base Case	2.0E-05	2.7E-06	1.1E-07	1.6E-08
Unit 1 and Unit 2 Train B Reserve Feed Current Outage Configuration	3.0E-05	4.1E-06		

External Events – Fire PRA Results

The Fire PRA model was re-quantified with the configuration information as discussed above. ICCDP and ICLERP are calculated for the full (100 hours) duration of the LCO. The full calculation, which follows the approach of RG 1.177, is provided in Enclosure 4.

Fire PRA Case	Fire CDF (/yr)	Fire LERF (/yr)	ICCDP	ICLERP
Fire PRA Base Case	5.4E-05	4.0E-06	1.1E-06	1.1E-07
Unit 1 and Unit 2 Train B Reserve Feed Current Outage Configuration	1.5E-04	1.4E-05		

External Events – Seismic

Offsite power is generally considered to have a low seismic fragility due to the connection to the larger electrical grid. The unavailability of a single train of offsite power does not significantly affect seismic risk, because seismic events large enough to cause an automatic reactor trip would generally also result in a loss of offsite power. For this reason, the seismic risk due to the reserve feed outage time extension is considered to be negligible.

External Events - Other

Other external events, such as high winds and external flooding, would be expected to result in a loss of offsite power during the event. Similar to the discussion for seismic events, significant external events that would cause a reactor trip would be expected to also cause a loss of offsite

power. For this reason, the external event risk due to the reserve feed outage time extension is considered to be negligible.

Conclusion

Total CDF and LERF Results

Case	ICCDP	ICLERP
FPIE	1.1E-07	1.6E-08
Fire PRA	1.1E-06	1.1E-07
Total	1.2E-06	1.3E-07

The thresholds for low risk in R.G. 1.177, Revision 1, are ICCDP < 1E-06 and ICLERP < 1E-07; however, the thresholds of ICCDP < 1E-05 and ICLERP < 1E-06 are acceptable should appropriate compensatory measure be implemented to reduce the sources of risk.

I&M has evaluated the risk implications of the proposed amendment. The risk assessment was performed assuming a 100-hour Completion Time. Therefore, the requested 28-hour extension in the allowed outage time is bounded by the risk assessment and associated compensatory actions.

4.2 Tier 2: Avoidance of Risk Significant Plant Configurations

CNP plant risk associated with the proposed extended Train B Reserve Feed Completion Time is calculated using the CNP FPIE and Fire PRA models (including internal flooding). Associated actions to avoid or respond to these events on one or both units through function of onsite emergency backup power supplies, and inclusion of additional onsite emergency power, are discussed in Tier 3 information, below.

Ultimately for this extended Completion Time request, CNP provides assurance that any other risk significant plant equipment outage configurations will not occur during the extended Completion Time period by flatly ruling out elective maintenance on other PRA risk significant plant equipment and avoiding other activities that could challenge unit operation or cause fires in risk significant areas. Refer to actions discussed in Tier 3, below. The Tier 3 actions mitigate additional plant risk due to events beyond those associated with Train B Reserve Feed unavailability represented in the ICCDP and ICLERP values furnished in the Tier 1 discussion above.

A full analysis of the FPIE and Fire PRA model results is provided in Enclosure 3. A summary of the relevant risk contributors is provided in this section.

IMPACT ON INTERNAL EVENTS (IE)

The internal events risk impact is included in the ICCDP and ICLERP metrics provided in Tier 1. The updated PRA model used in this assessment includes a typical Internal Events (IE) model and an internal flooding model. Based on a review of the results, risk management actions should focus on protecting the Unit 2 Train A equipment, and the Unit 1 ESW and CCW systems. Operators should focus on the most risk-significant actions related to a loss of CCW / ESW and the potential for loss of power due to only one train of reserve feed remaining.

Equipment protection for electrical systems and CCW and ESW are of particular importance. Although steam generator tube rupture was identified as a contributor, the additional risk from this event is primarily due to the potential for loss of power, so an emphasis on protecting equipment is appropriate.

IMPACT ON INTERNAL FLOODING

Internal flooding as noted above is part of the updated PRA model used to determine the PRA metrics provided in Tier 1. Based on review of the model, internal flooding events were not determined to be risk-significant for this specific configuration. Compensatory measures discussed in Tier 3 are therefore focused on the more risk-significant initiating events.

IMPACT ON FIRE RISK

As discussed above, the fire risk impact is included in the ICCDP and ICLERP metrics provided in Tier 1. Based on the review of the results, risk management actions should focus on maintaining the Unit 2 Train A equipment as available, and establishing hourly fire watches in areas identified as risk significant. Operator action review should focus on crosstie CVCS and tripping the RCPs to mitigate an RCP seal LOCA, both of which are contained within the fire emergency procedures. The Tier 3 information below includes actions to assure that fire detection and suppression systems for these areas are functional, that likelihood of fire initiation from work or operating equipment in the area is reduced/eliminated, and that flammable transient material is not in these high risk areas.

SUMMARY

For the Risk Management Actions (RMA) presented in the Tier 3 discussion below, CNP will avoid risk significant plant configurations such as performing elective maintenance or intrusive surveillances on the listed plant equipment, and minimizing activities that could initiate plant transients or challenge continued operation of Unit 1. Unit 2 is currently shutdown for a scheduled refueling outage.

RG 1.177 indicates that actions modifying plant design or operating procedures, or to obtain additional backup equipment, should be considered in the Tier 1 evaluation. However, no plant modifications have been made to reduce the risks associated with these Tier 2 considerations. An improvement estimate for some operational restrictions listed as Tier 3 actions are quantified as discussed in the quantitative analysis. Additional Tier 3 actions, which are developed to reduce the risk from risk-significant configurations identified in the quantitative analysis, are also included.

4.3 Tier 3: Risk Informed Configuration Management

Compensatory Measures

Given the difficulty of identifying all possible risk-significant configurations, for this one time Unit 1 TS. 3.8.1, Condition A, Required Action A.3, Completion Time change, CNP will reduce plant risk exposure through a combination of RMAs that prevent planned high risk

configurations and other non-quantifiable risk-reducing actions to reduce risk through availability of additional power supplies requiring manual actions.

RMAs to prevent high risk configurations (due either to fire initiation or other significant plant events), and establish non-quantifiable actions to monitor for high risk (fire or other internal or external) events and provide readily usable alternate power sources are listed below:

Note: These actions include a provision that if emergent plant conditions require actions to stabilize the unit(s), and if any of those actions conflict with any of the RMAs below, then those actions should be taken without delay, and the RMA restored after the emergent condition has passed and the plant is stabilized.

Operation and Maintenance Restrictions

Maintenance and testing during the Completion Time extension will be rescheduled for both units as warranted to minimize risk of unit transients. The development of these maintenance restrictions is based on an identification of risk-significant initiating events, operator actions, and equipment identified both before and after completion of the quantitative analysis. As discussed previously, some restrictions (specifically identified above) were credited in the quantitative analysis and others were developed by a thorough review of quantitative analysis results. These actions will specifically include:

- To the extent practicable and controllable, no other work is assumed to be undertaken that could jeopardize operation of Unit 1. For example, main turbine valve testing or similar activities, or maintenance work on BOP components that have the potential to initiate a unit trip, are assumed to be avoided while repair of Unit 1 and Unit 2 Train B Reserve Feed is in progress.

- Unit 1 Train A will be protected in accordance with plant procedures.

- No additional Unit 1 PRA equipment will be voluntarily removed from service.

- The Unit 2 RWST will maintain level at or above 35% to support the Unit 2 East Charging Pump capability to provide crosstie flow to Unit 1. This is the minimum level required by plant procedures for NFPA 805 crosstie support.

- The following Unit 1 equipment will be guarded in accordance with plant procedures:

    - Unit 1 CD EDG
    - Unit 1 AB & CD Batteries and Distribution Panels
    - Unit 1 East CCW Pump & Heat Exchanger

- The following Unit 2 Train A equipment shall be available and guarded in accordance with plant procedures:
    - Unit 2 East Motor-Driven Auxiliary Feedwater (AFW) Pump
    - Unit 2 East Component Cooling Water (CCW) Pump
    - Unit 2 East Charging Pump

  - Unit 2 East Essential Service Water (ESW) Pump

- The following Fire Zones are identified as high risk areas and will be protected (see discussion of activities in the next bullet):
  - Unit 1 CD Emergency Diesel Generator Room - Fire Zone 15
  - Unit 1 Train A Battery Room & Switchgear room cable vault– Fire Zones 55 and 56
  - Unit 1 Train A Switchgear Room - Fire Zone 40B
  - Unit 1 Turbine building El' 609 Air Compressor Area – Fire Zone 91
  - Screenhouse MCC ESW Equipment Area – Fire Zones 29G & 29E

- For each listed fire zone, the following activities will serve to protect the fire zone:
  a. No elective maintenance on fire detection or fire suppression equipment that will cause the fire detection or fire suppression equipment in the impacted fire zones to be inoperable.
  b. Verify installed Fire Detection and Suppression systems will be available, as applicable
     AND -
     Establish an hourly fire watch tour of the area
     OR -
     Establish a continuous fire watch in the area
  c. Transient combustible permits will be reviewed for the area and any unnecessary transients will be removed.
  d. No hot work will be allowed in the area.

- Operations crews will brief on the following procedures before entering the extended CT period
  - 1-OHP-4023-ECA-0-0, Loss of all AC Power
  - 1-OHP-4022-016-004, Loss of Component Cooling Water
  - 1-OHP-4025-001-001, Emergency Remote Shutdown
  - 12-OHP-4025-001-002, Fire Response Guidelines

I&M will ensure the recovery of the Unit 1 AC electrical source is of the highest priority and will exit the proposed action following satisfactory completion of the final operability runs.

5.0 REGULATORY EVALUATION

Applicable Regulatory Requirements/Criteria

10 CFR 50.36 (c)(2)(ii), stipulates that a TS LCO must be established for each item meeting one or more of the following criteria:

1. Installed instrumentation that is used to detect, and indicate in the CR, a significant abnormal degradation of the reactor coolant pressure boundary.
2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. A structure, system, or component which operating experience or PRA has shown to be significant to public health and safety.

The proposed changes do not modify any plant equipment that provides emergency power to the safety-related 4160v buses in the event of a LOOP. This one-time amendment request to extend the CT for TS 3.8.1, Condition A, has been prepared to comply with risk considerations from RG 1.177, Revision 1. Evaluation of the proposed changes has determined that the reliability of AC electrical sources is not significantly affected by the proposed changes and that applicable regulations and requirements continue to be met.

In conclusion, CNP has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements/criteria.

No Significant Hazards Consideration Determination

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   Response: No

   The proposed change is a one-time extension of a 72-hour Technical Specification (TS) required Completion Time for TS 3.8.1, Condition A.3. The proposed change does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. The proposed change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, the proposed completion time does not involve a significant increase in the probability of occurrence of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed amendment makes a one-time allowance of a 100-hour completion time for TS 3.8.1, Condition A.3. The proposed amendment does not introduce any new equipment, create any new failure modes for existing equipment, or create any new limiting single failures. The plant equipment considered when evaluating the existing completion time remains unchanged. The extended completion time will permit completion of repair activities without incurring transient risks associated with performing a shutdown with one train of

reserve feed unavailable. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed license amendment makes a one-time allowance of a 100-hour completion time for TS 3.8.1, Condition A.3. The proposed completion time has been evaluated on a risk-informed basis. The proposed configuration controls and compensatory measures provide reasonable assurance that no significant reduction to the margin of safety will occur. Therefore, the proposed change does not involve a significant reduction in margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed change involves no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 ENVIRONMENTAL CONSIDERATIONS

I&M has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared concerning the proposed amendment.

7.0 REFERENCES

None.

Enclosure 3

Probabilistic Risk Assessment (PRA) Technical Adequacy

1 Overview

PRA technical adequacy has been addressed through Nuclear Regulatory Commission Regulatory Guide (RG) 1.200, Revision 2 (Reference 2), which references the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard, RA-Sa-2009 (Reference 4), for internal events at power, internal flooding, and fire PRA models. External events (including Seismic Events) and shutdown risk impact will be considered quantitatively or qualitatively as described herein.

This enclosure demonstrates the technical adequacy of the Donald C. Cook Nuclear Plant (CNP) PRA model to be used as the basis for the emergency license amendment request (LAR), consistent with the requirements of Section 3.3 and Section 4.2 of RG 1.200, Revision 2 (Reference 2).

2 Relationship of the PRA Model to the As-Built, As-Operated Plant

The PRA model supporting the amendment for CNP Units 1 and 2 is the 2016 PRA model of record as documented in CNP calculation PRA-NB-QU Rev 2 (Reference 7), with an approved date of June 30, 2016, and the related PRA supporting documents. This is the most recent evaluation of the CNP internal events at-power risk profile, including internal flooding. To support the transition of its Fire Protection Program to National Fire Protection Association (NFPA) 805 (Reference 6), CNP developed a stand-alone Fire PRA model using the methodology defined in NUREG/CR-6850 (Reference 15). In October 2009, the initial Fire PRA was peer reviewed (Reference 9) against Section 4 of ASME RA-Sa-2009 (Reference 4), which includes the supporting requirements for fire PRAs. An additional focused-scope peer review of the Fire PRA with respect to modeling of Large Early Release Frequency was performed in November 2015 (Reference 17). The full-power, internal events (including internal flooding) PRA model and the Fire PRA model is maintained and updated under a PRA configuration control program in accordance with CNP procedures.

Plant changes, including physical and procedural modifications and changes in performance data, are reviewed and the PRA model is updated to reflect such changes periodically by qualified personnel, with independent reviews and approvals. An electronic tracking system documents such plant changes and performs Pending Change Evaluations for each, prioritizing them for inclusion into the PRA model of record as appropriate. The internal events PRA model update resulting in the June 2016 model of record evaluated all open Pending Change Evaluations and incorporated all changes identified to have a potential effect on the model results. The Fire PRA model was last updated in a working model in September of 2016. Refer to Section 4 for a detailed discussion on the Fire PRA.

The internal events PRA Model of Record is currently a CAFTA event tree / fault tree integrated model.

3 Conformance with the ASME/ANS PRA Standard for Internal Events and Internal Flooding

Indiana Michigan Power Company (I&M) considers the CNP Internal Events and Internal Flooding PRA as adequate to support the requested amendment. A Peer Review was conducted in July 2015 against ASME RA-Sa-2009 and RG 1.200, Rev. 2. The final Peer Review has been delivered to I&M and can be used to reliably evaluate the technical adequacy of the PRA against the Standard and RG (Reference 10). Previous peer reviews conducted prior to the 2015 peer review are considered to be superseded by the results of this latest peer review.

Each applicable supporting requirement (SR) in ASME RA-Sa-2009 was evaluated against a goal of Capability Category (CC) II. For each SR not meeting at least CC II, an evaluation is provided in the Technical Adequacy Justification Table, Section 8 of this enclosure, with respect to its impact on the proposed amendment.

The results of the peer review identify only 15 percent of the SRs as less than CC II. Many of these requirements are related to documentation, but a few technical issues are also identified. The key technical issues identified in the table relate to the areas of modeling of actuation signals (SY-B10), completeness of the pre-initiator human reliability analysis (HR-A1, A2, A3, B1, B2, C2), review of post-initiator human reliability analysis timing (HR-G4, G5), modeling of repair of essential service water (ESW) and component cooling water (CCW) systems (DA-C15), and ensuring the inclusion of all necessary internal flood scenarios (IFSN-A14, A16, A17, IFEV-A8, IFQU-A3). Resolution of the documentation issues is not expected to result in any impact to the PRA model or its associated metrics.

The resolution of the pre-initiator human reliability analysis (SRs HR-A1, A2, A3, B1, B2, C2) F&Os was subsequently determined to be a PRA upgrade during a recent RAI response to the NRC (Reference 20). The resolution of the F&Os for the listed SRs was subject to a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that all SRs (SRs HR-A1, A2, A3, B1, B2, C2) were met at CC II or higher. This discussion is provided for each SR in the Technical Adequacy Justification Table in Section 8.

In addition to the documentation issues and technical issues already identified, several CC I SRs are related to modeling or documentation associated with large early release frequency (LERF). The approach to many of the PRA Large Early Release SRs was determined to be conservative or have minimal numerical impact on LERF. Therefore, CC I is considered to be sufficient for these LE SRs for the purposes of this amendment.

4 Fire Events and External Events

4.1 Fire PRA

To support the transition of its Fire Protection Program to NFPA 805 (Reference 6), CNP developed a stand-alone Fire PRA model using the methodology defined in NUREG/CR-6850 (Reference 15). In October 2009, the initial Fire PRA was peer reviewed (Reference 9) against

Section 4 of ASME RA-Sa-2009 (Reference 4), which includes the supporting requirements for fire PRAs.

The transition to NFPA 805 received NRC approval in October 2013 (Reference 5). The stand-alone Fire PRA model, updated in response to the peer review and NRC requests for additional information during the NFPA 805 review process, is documented in calculation PRA-FIRE-17663-005-LAR (Reference 8). This model has been further refined in a documented working model to use for applications (Reference 18). This current working Fire PRA model supports the NFPA-805 program.

The Fire PRA model was initially developed from a previous version of the Internal Events PRA model of record, and continued to be developed as a standalone model to support the transition as documented in calculation PRA-FIRE-17663-005-LAR (Reference 8). The elements of the Fire PRA model taken from the Internal Events model were not reviewed in the 2015 peer review against Part 2 of ASME RA-Sa-2009 (Reference 4) and RG 1.200, Revision 2 (Reference 2). However, the SRs from Part 2 of ASME RA-Sa-2009 and RG 1.200, Revision 2 not meeting at least CC II as shown in the Technical Adequacy Justification Table, Section 8, are considered to be applicable to the Fire PRA as well.

An additional focused-scope peer review of the Fire PRA with respect to modeling of LERF was performed in November 2015 (Reference 17). The focused peer review assessed only the LERF (LE) and related Plant Response Model (PRM) requirements as related to the LERF portion of the Fire PRA model. As with the Internal Events PRA, each applicable supporting requirement (SR) in ASME RA-Sa-2009 was evaluated against a goal of Capability Category (CC) II. For each SR not meeting at least CC II, an evaluation is provided in the Technical Adequacy Justification Table, Section 8 of this enclosure, with respect to its impact on the proposed amendment.

4.2 External Events

The full-power, internal events PRA model does not include explicit consideration of external events such as seismic events, severe winds, and external flooding. The CNP Individual Plant Examination of External Events (IPEEE) (References 13 and 14) included an analysis of the following external events:

- Seismic Events
- External Flooding
- Aircraft Accidents
- Severe Winds (strong winds and tornadoes)
- Ship Impact Accidents
- Off-Site Hazardous Material Accidents
- On-Site Hazardous Material Accidents
- Turbine Missiles
- External Fires

The risk from external events is assessed qualitatively for the hazards listed.

5 Key Assumptions

Based on evaluations supporting the 2016 Internal Events PRA model of record, two key assumptions were identified as key model uncertainties:

5.1 Westinghouse Generation III Reactor Coolant Pump (RCP) Shutdown Seals

The modeling of the Westinghouse Generation III RCP shutdown seals is the first key model uncertainty for the CNP PRA. If the new RCP seals do not actuate or fail to remain actuated, severe accident sequences become much more likely. Risk metrics such as CDF and LERF increase significantly if failure of the shutdown seals is assured. The current PRA model utilizes the Pressurized Water Reactor Owners Group guidance (Reference 11) for PRA modeling of the shutdown seals, supported by the Westinghouse Owners Group 2000 RCP seal failure model (Reference 12), both of which are industry consensus models. The 2015 peer review also found the modeling of the shutdown seals acceptable. The RCP shutdown seals are only credited in the full-power, internal events model and are not credited in the CNP Fire PRA.

5.2 Component Repair Probabilities

The repair probabilities for appropriate ESW and CCW component failures are the second key model uncertainty for the CNP PRA. If repair is not credited, some key accident sequences become more likely. The 2015 peer review also identified the importance of this assumption, noting that the recovery credit is based on analysis of information from NSAC-161 (published in 1992) and therefore, this data is not representative of current plant operations (DA-C15). The Technical Adequacy Justification Table, Section 8 of this enclosure, of supporting requirements also identifies this issue and provides the resolution. This credit is no longer included in the model of record.

6 Conclusions

The CNP PRA models are generally robust and suitable to support this amendment. Specific issues identified by the recent peer review that may have a significant impact on the model have been addressed to reduce or eliminate their impact on the results. The ongoing PRA maintenance and update activities associated with the CNP PRA program ensure that the PRA models represent the as-built, as-operated plant moving forward. Therefore, the CNP PRA model has the technical adequacy required to support the amendment.

7 References

1. Not Used.

2. RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

3. RG 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision on Plant-Specific Changes to the Licensing Basis," Revision 2, March 2011.

4. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.

5. Letter from NRC to L. Weber, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) (TAC NOS. ME6629 and ME6630)," dated October 24, 2013, ADAMS Accession Number ML13140A398.

6. NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

7. PRA-NB-QU, "Internal Events Quantification Notebook," Revision 2, 6/30/2016.

8. PRA-FIRE-17663-005-LAR, "DC Cook Fire PRA Fire-Induced Risk Model," Revision 1, 10/28/2014.

9. LTR-RAM-II-10-041, "Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for the DC Cook Fire Probabilistic Risk Assessment," July 20, 2010.

10. PWROG-15076-P, Peer Review of the D. C. Cook Nuclear Plant Internal Events Probabilistic Risk Assessment, September 2015.

11. PWROG-14001-P, "PRA Model for the Generation III Westinghouse Shutdown Seal, Revision 1, July 2014.

12. WCAP-15603, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," Revision 1-A, June 2003.

13. "Individual Plant Examination, Other External Events Notebook," Revision 1, April 5, 1995.

14. "Individual Plant Examination, Addendum to Seismic Probability Risk Assessment Notebook," Revision 0, February 1995.

15. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Volume 2, September, 2005.

16. NUMARC 91-06, "Guidelines for Industry Actions to Address Shutdown Management," December 1991.

17. ERIN Engineering and Research, Inc., "D. C. Cook Focused Scope Peer Review for Fire PRA," Document #D0403140002-1515, November 2015.

18. PRA-NB-FIRE-W, "Fire PRA Working Model Notebook," Revision 1, September 2016.

19. Jensen Hughes, Inc. "D. C. Cook focused Scope Peer Review – Pre-Initiator HRA," Report No. 1BTI1V001-RPT-01, dated October 10, 2016.

20. Letter from Q. Shane Lies, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission, "Donald C. Cook nuclear Plant Unit 1 and Unit 2, Follow-Up Response to Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Relocate Surveillance Frequencies Program to Licensee Control-Risk Informed Technical Specification Task Force (RITSTF) Initiative 5B," dated September 9, 2016.

8 Technical Adequacy Justification Table

NOTE: Items listed below in the Supporting Requirements column specifically refer to supporting requirements in ASME RA-Sa-2009 (Reference 4).

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IE-A5	I	*The basis for screening control air and loss of 4160 volt (v) safety bus are not strong enough to assure that they can be screened or combined with another initiator. The generic partitioning of loss of offsite power (LOSP) into single and dual unit LOSP does not fit the plant and should have more analysis done to show the chosen set of initiating events is conservative.*	Loss of 4kV bus initiating events have been added to the model. Loss of Control Air modeling for AFW valves has been improved. The initiating event frequency is calculated individually as one of several types of Transients Without Steam Conversion Systems Available. Investigation into plant-specific SLOOP/DLOOP split fractions determined that all or most LOOPs would be dual-LOOPs. However, investigation into the actual industry events determined that the Plant-Centered and Switchyard-Centered events would not be expected to cause a LOOP for DC Cook. Therefore, the current frequencies and split fraction for PC and SC are conservative. To address Grid-Related and Weather-Related LOOPs being likely to be DLOOP, PDLOOP-WR/GR fractions are set to 1.0 and PSLOOP-WR/GR fractions are set to zero in the current model. IE F&Os have no impact on the Fire PRA since it has its own initiating events.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IE-B3	Not Met	*Basis: See previous discussion with Loss of Instrument air, LOSP, and Loss 4160 v bus. Additionally transients with steam generator overfill potential (which would fail the turbine driven auxiliary feedwater pump (TDAFP) have not been addressed.*	See the discussion in IE-A5. Excessive Feedwater is expected to have minimal impact due to the availability of two motor driven auxiliary feed pumps from the accident unit and two from the opposite unit via the crosstie that can supply AFW in the event that the TDAFP is failed. Therefore, the impact on this amendment is expected to be minor. IE F&Os have no impact on the Fire PRA since it has its own initiating events.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IE-C12	Not Met	*Basis: PRA-NB-INIT Section 5.2 documents the comparison of the posterior results and support system initiating event (SSIE) results to generic values. However, the explanation of differences is not very thorough. For example, the Loss of CCW initiating event frequency for CNP is approximately an order of magnitude higher than the generic value from NUREG/CR-6928. The explanation is that the difference is "reasonable based on the plant-specific cooling requirements of CCW." However, a more reasonable explanation may be differences in the degree of redundancy in the system or differences in requirements for operator action to start standby pumps. It is recommended that information from other sources like WCAP-16464-NP, "Westinghouse Owner's Group Mitigating Systems Performance Index Cross Comparison" or EPIX be used to identify plant-specific differences that may affect the initiating event frequencies.*	The explanation of differences is not expected to result in any changes to the SSIEs, as the modeling has been reviewed in detail and determined to match the plant systems. This issue is therefore a documentation issue only. IE F&Os have no impact on the Fire PRA since it has its own initiating events.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
AS-A10	I	*Basis: The CNP operator actions use a bounding timing to evaluate the operator actions associated with key safety functions. In addition, the timing for injection and recirculation modes of emergency core cooling system (ECCS) was assumed the same for all loss-of-coolant accidents (LOCAs) and Feed and Bleed scenarios.*	ECCS injection mission times only impact passive failures of the refueling water storage tank motor operated valves (MOVs) for residual heat removal, safety injection, and charging. Since more valves are opened during recirculation mode, a longer mission time for recirculation mode is conservative. The modeled operator action time for ECCS recirculation time is different for small LOCA than medium or large LOCA. Primary Bleed and Feed operator action times are also modeled per initiator. This approach is conservative, and only minor benefits are expected if operator action timing was evaluated on different accident sequences within the event trees. These issues have the same impact on the Fire PRA.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
SC-B5	Not Met	*Basis: PRA-NB-SC Revision 0 does not document a reasonableness and acceptability check with results from other sources for CNP specific analysis. There may be some reasonableness check in earlier Electric Power Research Institute (EPRI) reports (TR-100741, TR-1016750), however, those do not address modular accident analysis program (MAAP) 5 or plant-specific calculations. Recommend comparing the MAAP run results with Updated Final Safety Analysis Report and other plant-specific Safety Analysis.*	Since the MAAP results have been reviewed in detail, and MAAP has not been used outside of its applicability, the reasonableness check is not expected to result in changes to success criteria. Therefore, the impact on this amendment is expected to be minor. These issues have the same impact on the Fire PRA.
SY-A4	I	*Basis: Most of the notebooks indicate that interviews with knowledgeable plant personnel were conducted to confirm that the systems analysis adequately reflected the as-built, as-operated plant and that plant-specific data was appropriately collected where required. However, a record of such interviews was neither referenced nor provided.* *Walkdowns are discussed in a generic walkdown document created in June 1991. There is no record of recent system walkdowns conducted with knowledgeable plant personnel.*	Plant walkdowns were performed for all systems during initial model development and subsequent plant modifications have been tracked via the PRA configuration management process. Therefore, no model changes are expected during resolution of this issue for either the FPIE or Fire PRA.
SY-A8	Not Met	*Basis: Definitions of component boundaries are found in an excel spreadsheet used in development of PRA-NB-DATA, but the spreadsheet was not included with the balance of the documentation. There is no discussion related to ensuring that the system model uses component boundary definitions that match the boundaries used in the data collection.*	The assessment basis recognizes that the boundaries are included in files used for development of the Data Analysis, so this is a documentation improvement that would not affect model results.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
SY-B10	Not Met	*Basis: Actuation signals affecting multiple components are modeled in the 1CPA, 1CPB, 1RCA, 1RCB, 1SGA, 1SGB, 1SIA, and 1SIB fault trees and are documented in the PRA-NB-SY-ESFAS notebook. However, the actuation logic components other than the final master and slave relay for each signal are modeled as a single basic event representing failure of the logic resulting in generation of an SI, Reactor Trip, etc. The calculation of the probability of signal failure which is applied across all actuation models is based on common cause failure of two relays. On a train basis, the limiting component is typically the Safeguards Output Card or Undervoltage Output Card with a failure probability in the 10-4 range compared to the 5E-7 value being used. WCAP-15376, Table 8.25 shows SI signal failure probability with no operator action for 2/4 logic as 8.96E-04 and Diverse RT probability as 2.2E-05 from NUREG/CR-5500. These values are also significantly higher than the 5E-7 value being used for CNP.* *In addition, logic associated with the undervoltage signal to execute load shed and start the diesels on LOOP is not modeled.*	Modeling of actuation signals has been expanded in the FPIE to meet CC-II. These issues have the same impact on the Fire PRA, but are not expected to have a significant impact on the risk calculation due to their very low random failure rates.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-A1	Not Met	*Basis: CNP screened all component mispositioning. Some of these were screened using post-maintenance test, independent verification, or daily checks. Individual failure of these pre-initiator recovery actions or even combinations of these recoveries results in higher probabilities than passive valve failures and should not be screened.*	Pre-Initiator HFE screening has been re-evaluated to not pre-emptively screen mispositioning in the current model. The Fire PRA does not explicitly model pre-initiator HFEs, but any such events (including some identified in the FPIE related to the diesel generators) are not expected to have a significant impact on the risk calculation due to their very low failure rates for risk-significant actions. The resolution of the F&Os this SR was the subject of a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that this SR was met at CC II or higher.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-A2	Not Met	*Basis: Miscalibrations of control instruments and/or miscalibration of instruments that are used to direct operator actions (e.g., Refueling Water Tank (RWT) Level alarms) are needed to support mitigating system models and support applications like Fire PRA. There was no systematic review for identification of miscalibration events documented.*	Pre-Initiator HFE screening has been re-evaluated in the current model to include miscalibrations that can have an adverse impact on the automatic initiation of standby safety equipment as stated by the SR. For the impact on Fire PRA, see HR-A1. The resolution of the F&Os this SR was the subject of a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that this SR was met at CC II or higher.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-A3	Not Met	*Basis: Work practices that involve a mechanism that simultaneously affects equipment in either different trains of a redundant system or diverse systems are not specifically addressed. Miscalibrations of pressurizer pressure and high containment pressure are included (although not directly linked to the ESFAS and containment isolation signals). However, other potentially important and common miscalibrations, such as miscalibrations of RWT level transmitters preventing operators from diagnosing the need to swap ECCS suction to the sump, are not considered.*	Pre-Initiator HFE screening has been re-evaluated in the current model to include cross-train and redundant equipment, and miscalibrations. Pressurizer pressure and containment pressure are now linked to ESFAS via the new ESFAS modeling. For the impact on Fire PRA, see HR-A1. The resolution of the F&Os this SR was the subject of a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that this SR was met at CC II or higher.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-B1	I	*Basis: Rules were established for screening of individual activities, but application of the rules to specific procedures and activities using the rules is not documented. The screening process for component mispositioning and instrument calibration goes beyond the criteria provided in CC II of this SR. This process essentially screens classes of activities (e.g., mispositioning events and most instrument calibrations).*	Pre-Initiator HFE screening has been re-evaluated to not pre-emptively screen mispositioning in the current model. Pre-Initiator HFE screening has been re-evaluated in the current model to include miscalibrations that can have an adverse impact on the automatic initiation of standby safety equipment as stated by the SR. For the impact on Fire PRA, see HR-A1. The resolution of the F&Os this SR was the subject of a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that this SR was met at CC II or higher.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-B2	Not Met	*Basis: As noted in SR HR-B1, the screening of pre-initiator HRAs is aggressively beyond what is included in this standard. This includes miscalibrations and mispositioning that could impact support systems that impact multiple trains or systems*	Pre-Initiator HFE screening has been re-evaluated to not pre-emptively screen mispositioning in the current model. Pre-Initiator HFE screening has been re-evaluated in the current model to include miscalibrations that can have an adverse impact on the automatic initiation of standby safety equipment as stated by the SR. Pre-Initiator HFE screening has been re-evaluated in the current model to include cross-train and redundant equipment, and miscalibrations. Pressurizer pressure and containment pressure are now linked to ESFAS via the new ESFAS modeling. For the impact on Fire PRA, see HR-A1. The resolution of the F&Os this SR was the subject of a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that this SR was met at CC II or higher.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-C2	I	*Basis: The screening performed eliminates pre-initiator failure modes that have been experienced in generic operating experience and possibly would have occurred during plant-specific experience.*	Pre-Initiator HFE screening has been re-evaluated to not pre-emptively screen mispositioning in the current model. Pre-Initiator HFE screening has been re-evaluated in the current model to include miscalibrations that can have an adverse impact on the automatic initiation of standby safety equipment as stated by the SR. Pre-Initiator HFE screening has been re-evaluated in the current model to include cross-train and redundant equipment, and miscalibrations. Pressurizer pressure and containment pressure are now linked to ESFAS via the new ESFAS modeling. For the impact on Fire PRA, see HR-A1. The resolution of the F&Os this SR was the subject of a follow-on focused scope peer review (Reference 19) and the peer review determined that the resolutions were acceptable and that this SR was met at CC II or higher.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-D3	I	*Basis: The performance shaping factors in the HRA Calculator address procedure quality and administrative controls, but do not justify the basis for determining that these PSFs are good. Even the procedure references are hidden in notes within the HRA event.*	CNP emergency operating procedures follow emergency response guidelines and are generally of high quality. Control Room layouts are routinely trained on and familiar to operators. This issue was addressed during the updates to the pre-initiator HRA discussed in HR-A and HR-B above. For the impact on Fire PRA, see HR-A1.
HR-E3	I	*Basis: The operator interviews provided in Appendix C of the HRA report provide a walk-through of Emergency Operating Plan steps in general but does not provide a detailed review of the operator actions modeled (including time to perform the actions, whether the action has to be performed outside the Main Control Room, or any special equipment needed to perform the action (keys, jumpers, etc.)*	Post-initiator HRA updates were performed to include more detailed operator interviews with specific review of the necessary details. Post-initiator HRA updates were performed to include talk-throughs with operators to confirm response models. These updates are already included in the current model. For the Fire PRA, post-initiator HFEs were already updated and documented in response to the peer review and 805 RAIs (Reference 8).
HR-E4	I	*Basis: No simulator observations of talk-throughs with operators to confirm the response models for the scenarios modeled were noted in PRA-NB-HRA.*	See discussion for HR-E3

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-G4	Not Met	*Basis: It is not clear what the basis for the system time windows for many operator actions is. Some of the bases seem to be older Thrust analysis or generic studies when current MAAP analyses and other recent plant-specific evaluations are currently available.*	Post-initiator HRA updates were performed with detailed review of stated timings for all events, ensuring that all timings have a basis or a new basis is developed from operator interview and/or thermal-hydraulic calculations. Details are documented in the revision of the HRA notebook (and Success Criteria notebook for supporting MAAP runs). For the Fire PRA, see discussion for HR-E3.
HR-G5	I	*Basis: The completion times and median response times appear to be estimated for most Human Error Probabilities (HEPs).*	Post-initiator HRA updates were performed with detailed review of stated timings for all events, ensuring that all timings have a basis or a new basis is developed from operator interview and/or thermal-hydraulic calculations. Details are documented in the revision of the HRA notebook (and Success Criteria notebook for supporting MAAP runs). For the Fire PRA, see discussion for HR-E3.
HR-G6	Not Met	*Basis: Table D-1 in the HRA report provides a consistency check. However, there is no clear criteria established to determine consistency and HEPs seem to be more consistent with the method used to perform than the difficulty of the event and the time available to perform the action.*	An updated consistency check was performed along with the post-initiator HRA update. Documentation of consistency check is in the revision of the HRA notebook. For the Fire PRA, see discussion for HR-E3.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-I1	Not Met	*Basis: The current HRA analysis references previous analyses as the basis for event timing and in two cases assigns current HEPs screening values based on previous analysis (e.g., 1----FRH1-COGHE and 1K----INJECTNHE2). This makes it difficult to review the current analysis without recourse to the analysts. Since some of the authors of the previous analyses are no longer employed by the utility, the basis of inputs that vary from the norm cannot be verified.* *A lot of the HRA information was apparently copied from earlier versions when entering in the HRA Calculator code. This makes it difficult to determine if all of the information needed for the HRA evaluation exists.*	This SR is a documentation SR, so there is no impact on this amendment. Refer to the other HR SRs listed above for specific technical concerns with the pre and post-initiator human failure events (HFEs) (HR-A1,-A2,-A3, -B1, -B2, -C2 for pre-initiators and HR-D3, -E3, -E4, -G4,-G5, -G6 for post-initiators). The assumptions were updated as part of other HR SR resolutions. For the Fire PRA, see discussion for HR-E3.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
HR-I2	Not Met	*Basis: PRA-NB-HRA and the HRA Calculator database taken together document the HRA analysis. However, a number of steps in the process are not adequately documented. These include:* - *Lack of information concerning the process used to screen plant procedures and identify pre-initiator HFEs.* - *Source documents used to establish HEP timing for a number of actions.* - *Incomplete identification of procedures applicable to HFEs in the HRA Calculator* - *Dependence on previous analyses without verification that the timing information from the previous analysis is consistent with the current analysis.* *Lack of criteria for establishing consistency of the final HRA results*	This SR is a documentation SR, so there is no impact on this amendment. Refer to the other HR SRs listed above for specific technical concerns with the pre and post-initiator HFEs (HR-A1,-A2,-A3, -B1, -B2, -C2 for pre-initiators and HR-D3, -E3, -E4, -G4,-G5, -G6 for post-initiators). For the Fire PRA, see discussion for HR-E3.
DA-B1	I	*Basis: Does not meet CC II. With respect to the valves, the mission type (standby vs. operating) and service condition (e.g., treated vs. raw water) should factor into the grouping (to the extent supported by the data).* *Examples:* *AV type code – Air Operated Valves are used for the instrument air, aux. feedwater and essential service water systems.* *MV type code - MOVs are used in both clean and raw water applications.*	Updates to the generic data used for the component data that does not meet CC-II is not expected to significantly impact the model, since the most important component events use plant-specific data which would meet CC-II. For the Fire PRA random component failures are generally not important, and so will not have a significant impact on the risk results.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
DA-C8	I	*Basis: The CNP model estimates the fraction of time a component is in standby. For example, basic event, ESW-1WEST2WEST, represents the fraction of time Unit 1 AND Unit 2 west ESW pumps operating at 0.5.*	While collecting pump-specific run time data would improve overall model accuracy, it will not change the results significantly. CC I is considered sufficient for this amendment for this SR.
DA-C10	Not Met	*Basis: The PRA-NB-DATA Revision 4 notebook is silent on the use of surveillance test data. Based on a discussion with ERIN Engineering (Consulting company responsible for the latest data update), the system manager provided the estimated or actual number of demands. It is unclear if the system manager reviewed actual surveillance test data with respect to the PRA modeling of possible failure modes.*	The model may be slightly conservative due to this requirement, but any changes in resolving this issue are expected to be minor or negligible. For the Fire PRA random component failures are generally not important, and so will not have a significant impact on the risk results.
DA-C15	Not Met	*Basis: INIT-PRA-SY-ESW notes that "Recovery of the ESW system in the loss of ESW event trees includes recovery of failures caused by valves or pumps." The same recovery factor is used for CCW based on INIT-PRA-SY-CCW. The recovery credit is based on analysis of information from NSAC-161 to identify recoverable failures applicable to CNP. However, NSAC-161 was published in 1992 before implementation of the maintenance rule and mitigating systems performance index. In addition, the data is based on repair experience during normal plant operations which may not be applicable to conditions existing after an initiating event when the emphasis is in putting the plant in a safe condition. Therefore, this data is not representative of current plant operations and does not meet the requirements of DA-C15*	Credit for repair has been removed from the model until a new basis is developed. Credit is not included in the Fire PRA.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
QU-D4	I	*Basis: Plant comparison performed in Section 5.3.7 of the quantification notebook.* *The CNP results were compared to a peer plant, but the comparison is at a high level and does not present CNP and peer plant results side-by-side in the same format to allow identification of similarities and differences. The documentation is also unclear in some cases. For example, the first table in PRA-NB-QU, Attachment 6 has three results columns, but no headings. As it is, it is difficult to determine if all significant differences are explained in Section 5.3.7 since we do not have the results presented using the same breakdown as used in the peer plant pie charts.*	Resolution of this SR is expected to involve documentation improvements only since the comparison was already performed at a high level. CC I is considered sufficient to support amendments for this SR. No unique impact on the Fire PRA.
QU-F3	I	*Basis: The dominant contributors are listed and can also be found from the model results. Discussion of the events is less comprehensive than the first PRA quantification, but the information is available if desired*	Dominant contributors are discussed and reviewed in the quantification notebook (Reference 7), and no model changes are expected when the documentation is improved. No unique impact on the Fire PRA.
LE-C1	I	*Basis: The CNP LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C2	I	*Basis: The CNP LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1 which is considered conservative rather than realistic.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C3	I	*Basis: No repair of equipment after core damage was considered.*	CC I is considered to be sufficient to support amendments for this SR.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-C4	I	*Basis: The CNP LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1 and the event trees developed in those reports.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C5	I	*Basis: The CNP LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1 which is considered conservative rather than realistic.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C9	I	*Basis: No credit is taken for continued operation of equipment or operator actions in adverse environments.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C10	I	*Basis: No credit is taken for survivability of equipment or operator actions in adverse environments.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C11	I	*Basis: Containment failure equals LERF and ends the analysis. No events beyond containment failure are postulated.*	CC I is considered to be sufficient to support amendments for this SR.
LE-C12	I	*Basis: Containment failure equals LERF and ends the analysis. No continued operation of equipment beyond containment failure is postulated*	CC I is considered to be sufficient to support amendments for this SR.
LE-C13	I	*Basis: Bypass was a deterministic event (YES or NO). No source terms or scrubbing or decontamination was evaluated. All steam generator tube rupture (SGTR) sequences go to LERF.*	CC I is considered to be sufficient to support amendments for this SR.
LE-D5	I	*Basis: Models from WCAP-16341 are used for TI-SGTR and PI-SGTR. SGTR initiator taken directly to containment bypass*	CC I is considered to be sufficient to support amendments for this SR. Secondary Side Isolation is not considered to result in a direct containment bypass.
LE-E2	I	*Basis: Data is taken from NUREG/CR-6595 or WCAP-16341.*	CC I is considered to be sufficient to support amendments for this SR.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-F2	Not Met	*Basis: The CNP results were compared to a peer plant, but the comparison is at a high level and does not present CNP and peer plant results side-by-side in the same format to allow easy identification of similarities and differences. Although differences are discussed, the cause of significant differences in the contribution due to igniter failure was not fully explained*	Resolution of this SR is expected to involve documentation improvements only since the comparison was already performed at a high level. Containment Failure Probabilities given hydrogen igniter failure are taken from NUREG/CR-6595, and no issues were noted with the hydrogen igniter system model.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IFSN-A16	I	*Basis: CNP IE HRA is performed in PRA-FLOOD-013 Revision 1, which is developed using the EPRI HRA Calculator. The HEPs account for flood indication in the control room, potential to isolate flood sources, and accounting for the likelihood of mitigative actions. Initial screening of internal flooding events is documented in PRA-FLOOD-012 Revision 0 Attachment A, Internal Flood Source Qualitative, Screening. This initial qualitative screening is based, in part, on time to consequential IE or time to failure of PRA equipment. For example, FA-01-1a represents a flood from the Unit 1 ESW, which is screened based on "...only 1 train ESW lost at a time, no IE or other PRA damage for 170 min." This could meet the intent of IFSN-A14 CC I item (c) "...time to the damage of safe shutdown equipment is significantly greater than the expected time for human mitigative actions to be performed..." However based on the screening note, this does not meet the intent of IFSN-A14 CC II item (c) "...mitigative action can be performed with high reliability for the worst flooding initiator. High reliability is established by demonstrating, for example, that the actions are procedurally directed, that adequate time is available for response, that the area is accessible, and that there is sufficient manpower available to perform the actions."*	The quantitative screening has not yet been fully rechecked, but any new scenarios would not likely be affected by the specific equipment outage in the amendment. Significant equipment related to recovery from loss of offsite power was reviewed to ensure the flooding model deficiencies are not expected to significantly impact this application. Battery and switchgear rooms are above grade and not a propagation pathway for floods. Sprays and floods initiated in these rooms are present in the PRA model. Only a large flood from a CW pipe is considered credible to fail the EDGs. These large CW breaks are also currently modeled.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IFSN-A17	Not Met	*Basis: CNP performed internal flooding walkdowns to verify the accuracy of information. The walkdown notes are documented in Attachment C of PRA-FLOOD-004 Revision 1. For each flood area the following is identified: 1) Flood Targets (PRA Equipment which is defined as "equipment that may be adversely affected due to the accumulation of water, spray, dripping, and steam damage."), 2) Flood/High Energy Line Break (HELB) sources, 3) Barrier Openings, and 4) Flood Mitigation Features.* *Some of the assumptions made during evaluation of flood propagation require additional analysis to substantiate. PRA-FLOOD-004 Revision 1 makes the following assumptions:* *a) Doors that have a gap beneath them of less than 1/8 inch are considered adequately sealed against possible propagation.* *b) Single doors that open toward a projected flood are assumed to remain intact when subjected to flood forces.* *c) The fixed doors of double doors at CNP are latched at the top and bottom. It is assumed the fixed door being supported by the hinges and at the top and bottom on the free standing side would remain in place given a flood from either direction.* *Specifically, an engineering analysis needs to be performed to verify the closed doors can handle the applied force when the water level has reached its maximum potential. CNP PRA meets (a) and (b), however, (c) requires additional analysis/documentation to verify propagation paths are appropriately analyzed.*	Evaluation of the assumption related to doors could impact the model and results if new flooding pathways are identified. However, these would not likely be affected by the specific equipment outage in the amendment. Significant equipment related to recovery from loss of offsite power was reviewed to ensure the flooding model deficiencies are not expected to significantly impact this application. Battery and switchgear rooms are above grade and not a propagation pathway for floods. Sprays and floods initiated in these rooms are present in the PRA model. Only a large flood from a CW pipe is considered credible to fail the EDGs. These large CW breaks are also currently modeled.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IFEV-A8	Not Met	*Basis: Quantitative screening is documented in PRA-FLOOD-014 Table 1, Unit 1 Internal Flood PRA Bounding Screening Results, and further analyzed in Table 2, Unit 1 Flooding PRA Best Estimate. This quantitative screening analysis was performed in 2006. These results were based on the core damage frequencies using pipe rupture initiating event frequencies from EPRI TR-1013141, "Pipe Rupture Frequencies for Internal Flooding PRAs - Revision 1" March 2006 and in some cases EPRI TR-1 02266 "Pipe Failure Study Update", April 1993. The generic pipe rupture frequencies have been updated a couple of times since these reports with more current information and understanding of the pipe rupture frequencies. The latest CNP internal flooding analysis, PRA-FLOOD-016, is based on the latest pipe rupture frequencies from EPRI 3002000079, Pipe Rupture Frequencies for Internal Flooding PRAs. Revision 3, 2013.* *The current quantitative screening analysis is based on outdated pipe rupture failure rates. The latest pipe rupture frequencies from 2013 report are significantly higher than the earlier values. It is recommended that the flood scenarios be updated with the latest values and reanalyze the screening.*	The quantitative screening has not yet been fully rechecked, but any new scenarios would not likely be affected by the specific equipment outage in the amendment. Significant equipment related to recovery from loss of offsite power was reviewed to ensure the flooding model deficiencies are not expected to significantly impact this application. Battery and switchgear rooms are above grade and not a propagation pathway for floods. Sprays and floods initiated in these rooms are present in the PRA model. Only a large flood from a CW pipe is considered credible to fail the EDGs. These large CW breaks are also currently modeled.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
IFQU-A3	Not Met	*Basis: Quantitative screening of flood groups is documented in PRA-FLOOD-012, Attachments B and C. No flood groups were screened if the CDF using a bounding CCDP was greater than 1E-9.* *However, the screening in PRA-FLOOD-012 was based on flood source initiating frequencies calculated using EPRI pipe break data older than that used for the update of the 'dominant' flood groups and CCDP values calculated using an earlier PRA model. A number of the flood groups were screened with bounding CDF values just below 1E-9. These groups may be above the screening threshold using updated flood frequencies and CCDP values. All of the information for the flood groups should be re-calculated using the updated data and current model to ensure that the flood group screening represents the current state of knowledge.*	The quantitative screening has not yet been fully rechecked, but any new scenarios would not likely be affected by the specific equipment outage in the amendment. Significant equipment related to recovery from loss of offsite power was reviewed to ensure the flooding model deficiencies are not expected to significantly impact this application. Battery and switchgear rooms are above grade and not a propagation pathway for floods. Sprays and floods initiated in these rooms are present in the PRA model. Only a large flood from a CW pipe is considered credible to fail the EDGs. These large CW breaks are also currently modeled. New scenarios would be expected to be at or near the 1E-9 screening criteria and would not be expected to significantly contribute to the proposed equipment outage risk.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-C1 (Fire PRA)	I	*The D. C. Cook LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1.*	CC I is considered acceptable for Fire PRA for this amendment for this SR. The CNP LERF model includes several conservatisms since a full generic containment model for ice condenser containments was not developed for WCAP-16341. These conservatisms overestimate LERF and thus have a conservative impact on decisions made using the Fire PRA LERF results. Severe accident containment failure modes are mostly phenomenological and thus do not depend on the type of initiating event that might cause a severe accident. Fire-induced containment bypass events, such as loss of containment isolation, are explicitly modeled. Fire-induced loss of the hydrogen igniters is also explicitly modeled.
LE-C2 (Fire PRA)	I	*The D. C. Cook LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1 which is considered conservative rather than realistic.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.
LE-C3 (Fire PRA)	I	*No repair of equipment after core damage was considered.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-C4 (Fire PRA)	I	*Basis: The D. C. Cook LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1 and the event trees developed in those reports.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.
LE-C5 (Fire PRA)	I	*The D. C. Cook LERF analysis follows methods in WCAP-16341 and NUREG/CR-6595, Revision 1 which is considered conservative rather than realistic.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-C7 (Fire PRA)	Not Met	*Sections 6.5.3 and 6.5.6 of PRA-NB-FIRE-LE describes the operator actions credited in the LERF Model. The level of detail of the analysis does not reflect the use of the applicable requirements in Section 2-2.5 of the PRA Standard.*	Several specific issues are identified in the associated F&Os. The main impact of fire on the identified HFEs is the level of stress during a fire event. Since these are Level 2 HFEs that occur after severe core damage has already occurred or is imminent, stress is already expected to be high. Therefore, fire events that result in core damage are considered to not be significantly more stressful than other events (such as large LOCAs, small LOCAs, SGTRs) once core damage has occurred. Thus, it is judged that it is not necessary to adjust the value of the Level 2 HFEs generally for fire events. Another impact of fire events is the possibility that operators would have to evacuate the control room due to a fire. Where it is unlikely that the operators would be able to perform a function, no credit is taken for this action in main control room evacuation scenarios.
LE-C9 (Fire PRA)	I	*No credit is taken for continued operation of equipment or operator actions in adverse environments.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-C10 (Fire PRA)	I	*No credit is taken for survivability of equipment or operator actions in adverse environments.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.
LE-C11 (Fire PRA)	I	*Containment failure equals LERF and ends the analysis. No events beyond containment failure are postulated.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.
LE-C12 (Fire PRA)	I	*Containment failure equals LERF and ends the analysis. No continued operation of equipment beyond containment failure is postulated.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.
LE-C13 (Fire PRA)	I	*Bypass was a deterministic event (YES or NO). No source terms or scrubbing or decontamination was evaluated. All SGTR sequences go to LERF.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-C1.
LE-D1 (Fire PRA)	I	*CNP has a plant-specific containment fragility analysis (Attachment 1 of PRA-L2 MODEL, Revision 0) that predicts the ultimate containment capacity and the location of containment failure on pressure. However, it is not clear if and how this calculation was factored into the simplified Level 2 model documented in PRA-NB-FIRE-LE. Attachment 1 of PRA-L2 MODEL is not cited in Section 8 of PRA-NB-FIRE-LE and not discussed in Section 6.5.1 of the report. The simplified Level 2 model appears to be using NUREG/CR-6595 if the igniters fail.*	CC I is considered to be sufficient to support this amendment for this SR. The plant-specific containment analysis is now credited for improving the total time available to energize the igniters, allowing the recovery action to repower the igniters to be credited in more fire areas. This was the most significant conservatism present in the LERF model for the Fire PRA.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-D2 (Fire PRA)	I	*CNP has a plant-specific containment fragility analysis (Attachment 1 of PRA-L2 MODEL, Revision 0) that predicts the ultimate containment capacity and the location of containment failure on pressure. However, it is not clear if and how this calculation was factored into the simplified Level 2 model documented in PRA-NB-FIRE-LE. Attachment 1 of PRA-L2 MODEL is not cited in Section 8 of PRA-NB-FIRE-LE and not discussed in Section 6.5.1 of the report. The simplified Level 2 model appears to be using NUREG/CR-6595 if the igniters fail.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-D1.
LE-D3 (Fire PRA)	I	*CNP has a plant-specific containment fragility analysis (Attachment 1 of PRA-L2 MODEL, Revision 0) that predicts the ultimate containment capacity and the location of containment failure on pressure. However, it is not clear if and how this calculation was factored into the simplified Level 2 model documented in PRA-NB-FIRE-LE. Attachment 1 of PRA-L2 MODEL is not cited in Section 8 of PRA-NB-FIRE-LE and not discussed in Section 6.5.1 of the report. The simplified Level 2 model appears to be using NUREG/CR-6595 if the igniters fail.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-D1.
LE-D5 (Fire PRA)	I	*Models from WCAP-16341 are used for TI-SGTR and PI-SGTR. SGTR initiator taken directly to containment bypass.*	CC I is considered to be sufficient to support this amendment for this SR. See LE-D1.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-E1 (Fire PRA)	Not Met	*Sources for parameter values are shown in Table 2 of PRA-NB-FIRE-LE. Appropriate parameter values were selected consistent with the requirements of technical element DA. Operator actions identified in Sections 6.5.3 and 6.5.6 were not selected in accordance with Section 2-2.5 of the PRA Standard.*	See LE-C7.
LE-E2 (Fire PRA)	I	*Basis: Data is taken from NUREG/CR-6595 or WCAP-16341.*	CC I is considered to be sufficient to support this amendment for this SR. Containment Isolation modeling is taken from the Internal Events model, with some specific Fire PRA components added as necessary. This is related to LE-C8, which was Met.
LE-F1 (Fire PRA)	Not Met	*The results of the LERF quantification and cutset reviews are provided in PRA-FIRE-17663-014-LAR-R1-final-1017, Tables 5-1, 5-7, 5-9, 5-17 and 5-19. The results do not provide contributions by LERF PDS designation and LERF failure mechanism.*	The updated Fire PRA model includes the quantification results by failure mechanism.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-F2 (Fire PRA)	Not Met	*The CNP results were not compared to a peer plant.*	The ASME standard specifically states in SR FQ-E1, that SR QU-D3 (plant comparison of results) is not applicable to a Fire PRA (LE-F2 is not discussed, but the derived requirement is related). Additionally, LE-F2 states that a reasonableness check is required (i.e. review the results to ensure excessive conservatism has not skewed the results). While a comparison to a peer plant could function as a reasonableness check for an internal events model, Fire PRAs are inherently different since the spatial location of cables varies widely at other plants. Thus, a general discussion of the results is provided in the working model notebook, including some review of results from another plant. This discussion is focused at a high level to provide insights into possible conservatisms, as opposed to a direct comparison of results.
LE-G3 (Fire PRA)	Not Met	*The results of the LERF quantification and cutset reviews are provided in PRA-FIRE-17663-014-LAR-R1-final-1017, Tables 5-1, 5-7, 5-9, 5-17 and 5-19. The results do not provide contributions by LERF PDS designation and LERF failure mechanism.*	The updated Fire PRA model includes the quantification results by failure mechanism.

Supporting Requirement	Capability Category	Peer Review Assessment Basis	Assessment
LE-G6 (Fire PRA)	Not Met	*Sections 5.2 and 5.3 of PRA-FIRE-17663-014-LAR-R1-final-1017 provide a quantitative definition used for significant core damage accident progression sequence that is consistent with Part 1-2 of the standard. However, there is no equivalent definition for LERF.*	The internal events LERF notebook now provides a definition of LERF sufficient to meet the standard.
PRM-B2 (Fire PRA)	Not Met	*An assessment of Internal Event PRA peer review deficiencies is required to evaluate the impact on the Fire PRA.*	As this is linked to the internal events model, CC I is considered to be sufficient to support this amendment for this SR.
PRM-B14 (Fire PRA)	Not Met	*Provide documentation demonstrating an evaluation for this SR. Evaluate the potential for screened LERF scenarios impacting the Fire PRA, e.g., LERF bypass pathway screened based on size, where a fire may impact multiple pathways where the sum of the pathway sizes may exceed the LERF bypass pathway screening criteria.*	The potential for multiple containment isolation valve failures to result in a large early release is now included in the analysis and working model notebook.
PRM-B15 (Fire PRA)	Not Met	*Provide documentation demonstrating an evaluation for this SR.*	The potential for multiple containment isolation valve failures to result in a large early release is now included in the analysis and working model notebook.

Enclosure 5 to AEP-NRC-2016-83

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION PAGE
MARKED TO SHOW PROPOSED CHANGES

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	---------------NOTE-------------- Not applicable if a required Unit 2 offsite circuit is inoperable. ------------------------------------	
		Perform SR 3.8.1.1 for required OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours (See Footnote) AND 17 days from discovery of failure to meet LCO 3.8.1.a or b

Footnote: For Train B only, the Completion Time that Train B can be inoperable as specified by Required Action A.3 may be extended beyond the "72 hours" up to "100 hours," to support repair and restoration of the Train B Reserve Feed. Upon completion of the repair and restoration associated with this event which occurred on October 7, 2016, this footnote is no longer applicable.

Enclosure 6 to AEP-NRC-2016-83

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION PAGE CHANGES INFORPORATED

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	--------------NOTE------------- Not applicable if a required Unit 2 offsite circuit is inoperable. -------------------------------------	
		Perform SR 3.8.1.1 for required OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours (See Footnote) AND 17 days from discovery of failure to meet LCO 3.8.1.a or b

Footnote: For Train B only, the Completion Time that Train B can be inoperable as specified by Required Action A.3 may be extended beyond the "72 hours" up to "100 hours," to support repair and restoration of the Train B Reserve Feed. Upon completion of the repair and restoration associated with this event which occurred on October 7, 2016, this footnote is no longer applicable.