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U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 1 and Unit 2
Response (Part 2) to Fourth Request for Additional Information Regarding the License Amendment Request to Adopt TSTF-490 and Implement Alternative Source Term

References:

1. Letter from J. P. Gebbie, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC), "Donald C. Cook Nuclear Plant, Units 1 and 2, License Amendment Request to Adopt TSTF-490, Revision 0, 'Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification' and Implement Full-Scope Alternative Source Term," dated November 14, 2014, Agencywide Documents Access and Management System (ADAMS) Accession No. ML14324A209.

2. Letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Unit 1 and Unit 2 - Supplemental Information for the License Amendment Request to Adopt TSTF-490, Rev 0, "Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification" and Implement Full-Scope Alternative Source Term," dated February 12, 2015, ADAMS Accession No. ML15050A247.

3. E-mail capture from A. W. Dietrich, NRC, to H. L. Kish, I&M, "D.C. Cook Units 1 and 2 – ARCB RAI Concerning LAR to Adopt TSTF-490 and Implement Full-Scope AST (TAC NOS. MF5184 and MF5185)," dated September 15, 2015, ADAMS Accession No. ML15259A577.

4. Letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Unit 1 and Unit 2 - Response (Part 1) to Fourth Request for Additional Information Regarding the License Amendment Request to Adopt TSTF-490 and Implement Alternative Source Term," dated November 16, 2015, ADAMS Accession No. ML15323A434.

5. E-mail capture from A. W. Dietrich, NRC, to H. L. Kish, I&M, "D.C. Cook Units 1 and 2 – ARCB RAI Concerning LAR to Adopt TSTF-490 and Implement Full-Scope AST (TAC NOS. MF5184 and MF5185)," dated December 7, 2015, ADAMS Accession No. ML15342A049.

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NRR

This letter provides Indiana Michigan Power Company's (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, response (part 2) to the fourth Request for Additional Information (RAI) by the U. S. Nuclear Regulatory Commission (NRC) regarding a license amendment request (LAR) to adopt Technical Specification Task Force (TSTF)-490 and implement Alternative Source Term (AST).

By Reference 1, as supplemented by Reference 2, I&M submitted a request to amend the Technical Specifications to CNP Units 1 and 2 Renewed Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to adopt TSTF-490, Revision 0, and implement full scope AST radiological analysis methodology. By Reference 3, the NRC transmitted an RAI from the Radiation Protection and Consequence Branch (ARCB) regarding the LAR submitted by I&M in Reference 1. This RAI contains eight separate items for which additional information is requested. By reference 4, I&M provided responses for items RAI-ARCB-1, -3, -4, -5, -6, and -7, and stated the intent to respond to items RAI-ARCB-2 and -8 by December 16, 2015. Upon agreement with NRC Project Manager, the date of submittal for this letter is acceptable for addressing items RAI-ARCB-2 and -8. By Reference 5, the NRC transmitted an additional RAI from the ARCB, RAI-ARCB-9, which is also addressed by this letter.

Enclosure 1 to this letter provides an affirmation statement. Enclosure 2 to this letter provides I&M's response to the final two items of the NRC's RAI in Reference 3 and the RAI in Reference 5. Copies of this letter 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 made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

TLC/ams

Enclosures:

1. Affirmation
2. Response (Part 2) to Fourth Request for Additional Information Regarding the License Amendment Request to Adopt TSTF-490 and Implement Alternative Source Term

c: R. J. Ancona, MPSC
A. W. 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

AFFIRMATION

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

Indiana Michigan Power Company

Joel P. Gebbie
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 17 DAY OF December, 2015

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-2015-111

Response (Part 2) to Fourth Request for Additional Information Regarding the License Amendment Request to Adopt TSTF-490 and Implement Alternative Source Term

By letter dated November 14, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14324A209), as supplemented by letter dated February 12, 2015 (ADAMS Accession No. ML15050A247), Indiana Michigan Power Company (I&M), the licensee for the Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, submitted a license amendment request. The proposed amendment consists of adoption of Technical Specifications Task Force-490, Revision 0, and implementation of a full scope alternative source term (AST) radiological analysis methodology.

The U.S. Nuclear Regulatory Commission (NRC) staff in the Radiation Protection and Consequence Branch (ARCB) of the Office of Nuclear Reactor Regulation is currently reviewing the submittal, as supplemented, and has determined that additional information is needed in order to complete the review. The text of the requests for additional information (RAIs) and I&M's responses are provided below.

RAI-ARCB-2

The gap fractions in Table 3 of RG 1.183 are applicable to fuel with a maximum rod burnup of 62,000 megawatt-days per metric ton of uranium and a maximum linear heat generation rate of 6.3 kilowatt per foot, for rods exceeding burnups of 54 gigawatt-days per metric ton of uranium, per Footnote 11. To provide margin for future core designs, the licensee's radiological dose analysis is based upon a number of rods exceeding the burnup limits of Footnote 11. To address these rods, the licensee modeled the gap inventory to two times the values shown in Table 3 of Reg. Guide 1.183 for all rods in each fuel assembly that contains high burnup rods.

The licensee states that "similar high burnup concerns were raised in AST submittals by Fort Calhoun, Byron/Braidwood, and St. Lucie stations." For these plants, the issue was addressed by doubling the gap fractions for 100% of the rods in the affected assemblies and applying the maximum radiation peaking factor.

In the SE for Fort Calhoun (ADAMS Accession No. ML013030027), the NRC staff concluded that there was reasonable assurance that the radiation doses analyzed using the gap fractions proposed by Omaha Public Power District (OPPD) would bound the radiation doses resulting from an actual event. The NRC staff based this conclusion on the site-specific OPPD analysis. I&M states that this same methodology, multiplying the gap fractions by a factor of two, was applied to the AST analysis for addressing high burnup fuel at CNP.

a) *Demonstrate that CNP is within the assumptions of the referenced Fort Calhoun analysis. Alternatively, provide a site-specific analysis to address the presence of high burnup fuel, using an NRC-approved methodology.*

I&M Response to RAI-ARCB-2:

As noted in the request, the fraction of the core fission product inventory located within the fuel rod gap for non-loss of coolant accident (LOCA) events is specified in Table 3 of Regulatory Guide (RG) 1.183 (Reference 1). Footnote 11 of Reference 1 states that the fuel rod gap inventories are acceptable for fuel rods with a peak burnup of up to 62,000 megawatt-days per metric ton of uranium (MTU) provided that the linear heat generation rate does not exceed 6.3 kilowatts per foot (kw/ft) for rods with burnups greater than 54 gigawatt-days per MTU. To provide future core design margin, a total of 150 rods in two fuel assemblies are assumed to exceed the burnup limits from Footnote 11 for non-LOCA events. This is addressed in the non-LOCA accident analyses by doubling the gap fractions for 100 percent of the rods in the affected assemblies and applying the maximum radial peaking factor. Note that for the fuel handling accident (FHA) analysis, since the high burnup adjustment involves doubling the gap inventory of all rods in the affected assembly, and since all of the rods in the dropped assembly are assumed to fail, the entire source term for the FHA analysis is increased by a factor of two.

Per the safety evaluation for Fort Calhoun (Reference 2), the NRC staff concluded that there is reasonable assurance that the radiation doses analyzed using the gap fractions proposed by the licensee will bound the radiation doses resulting from an actual event based on the following:

(1) A nearly factor of four margin in the short-lived isotopes

(2) The conservative application of a radial peaking factor based on the fuel assembly in the highest power position to all assemblies projected to be damaged

(3) The conservatively assumed higher linear heat generation rate (plus 200 degrees Fahrenheit (°F) margin)

(4) The general acceptability of the American National Standards Institute/American Nuclear Society (ANSI/ANS)-5.4 method

As noted in Item 1 above, Fort Calhoun applied a nearly factor of four margin for the short-lived isotopes. This is due to the plant-specific analysis which showed that the gap fractions for short-lived isotopes are approximately half of the corresponding values tabulated in Reference 1. Similar results would be expected for the affected assemblies at CNP using the methods of ANSI/ANS-5.4. This is based upon similar results obtained by the plant-specific analysis performed by Prairie Island (see Table 4-2 from Exhibit F of Reference 7), which also used the ANSI/ANS-5.4 method through the use of the gap fraction analysis (GAP) code.

Regarding Item 2 above, a radial peaking factor of 1.65 was utilized in the CNP analyses outlined in AEP-NRC-2014-65 (Reference 3) to bound CNP Units 1 and 2. This value is conservatively based upon the nuclear enthalpy rise hot channel factor plus uncertainties and was applied to the analysis in accordance with Section 3.1 of Reference 1.

To account for the conservatively assumed higher linear heat generation rate (plus 200°F margin) as noted in Item 3 above, I&M confirms the peak linear heat generation rate with the fuel vendor during core reload design activities to ensure that this limit is not exceeded. The current linear heat generation rate limit of 6.3 kw/ft is confirmed on a per cycle basis in accordance with a vendor Reload Safety Analysis Checklist. In the event that high burnup must be evaluated for a given cycle, the fuel vendor will be requested to perform additional analyses to ensure the assumptions of Reference 2 are not exceeded.

Noting the discussion above, including confirmation of the peak linear heat generation rate during reload activities, the gap fractions utilized in the analyses outlined in Reference 3 would bound the radiation doses resulting from an actual event at CNP

RAI-ARCB-8

Appendix B of RG 1.183, Regulatory Position 1.1 states:
The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.

By letter dated November 13, 2001, the NRC issued amendment Nos. 258 for Unit 1, and No. 241 for Unit 2 (ADAMS Accession No. ML012980378), which approved a portion of the June 12, 2000, proposed license amendment which requested use of the AST associated with a fuel-handling accident (FHA). The CR radiation doses reported by the licensee are tabulated in Table 2 of amendment Nos. 258 and 241. In performing this analysis, the licensee selected parameters that demonstrated results that would bound the consequences of both a FHA inside containment and a FHA in the fuel handling building.

The FHA activity is assumed to be released from (1) the damaged fuel via the spent fuel pool to the fuel handling building, or (2) from the damaged fuel via the reactor cavity to the containment, from which it is assumed to be released to the environment over two hours as an unfiltered ground-level release. When evaluating the dose to operators in the CR, it was assumed that the operators would manually place the CR ventilation in emergency operation mode at 30 minutes following the start of the event.

Limiting Condition for Operation 3.7.10, Control Room Emergency Ventilation (CREV) System, states, "Two CREV trains shall be operable," and is applicable during "MODES 1, 2, 3, and 4; and During movement of irradiated fuel assemblies in the containment, auxiliary building, and the Unit 1(2) containment."

Limiting Condition for Operation 3.7.13, Fuel Handling Area Exhaust Ventilation (FHAEV) System, states, "One FHAEV train shall be operable and in operation," and is applicable "During movement of irradiated fuel assemblies in the auxiliary building."

Limiting Condition for Operation 3.9.4, Residual Heat Removal (RHR) and Coolant Circulation – High Water Level, is applicable in "MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange." Limiting Condition for Operation 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level, is applicable in "MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

Limiting Condition for Operation 3.9.6, Refueling Cavity Water Level, states "Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange," and is applicable "During movement of irradiated fuel assemblies within containment."

Limiting Condition for Operation 3.7.14, Fuel Storage Pool Water Level, states, "The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks," and is applicable "During movement of irradiated fuel assemblies in the fuel storage pool."

a) *Explain how the proposed revised FHA analysis meets or bounds RG 1.183, Regulatory Position 1.1, taking into account that the TS applicability does not specifically require CREV or FHAEV to be operable or water level to be at least 23 feet during core alterations in which other loads (such as a fresh fuel assembly, sources, etc.) can be moved. In addition, clarify how the revised FHA analysis determines the most limiting case, and how it shows that the limiting case is not the drop of an object other than an irradiated fuel assembly.*

I&M Response to RAI-ARCB-8:

In the body of RG 1.183 (Reference 1), Regulatory Position 1.1 discusses generic considerations of AST implementation, including safety margins, defense in depth, and facility design basis. More specifically, Regulatory Position 1.1.3 of Reference 1, Integrity of Facility Design Basis, states, in part, "Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. Although a complete reassessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses would generally not be necessary."

In the Updated Final Safety Analysis Report (UFSAR) for CNP (Reference 4), Section 14.2.1.6 states, "For a fuel handling accident in the auxiliary building, the accident analysis postulates that a spent fuel assembly is dropped onto the racks in the spent fuel pool or on the floor of the SFP or transfer canal, rupturing the cladding of all the fuel rods despite the administrative controls and physical limitations imposed on the fuel handling operations."

UFSAR Section 14.2.1.6, states, "For a fuel handling accident in the containment building, the accident analysis postulates that a spent fuel assembly is dropped onto the reactor core or on the floor of the reactor cavity pool, rupturing the cladding of all the fuel rods despite the administrative controls and physical limitations imposed on the fuel handling operations."

These sections of the UFSAR describe a FHA as being based on handling a spent fuel assembly. The proposed accident analysis for the FHA using AST is bounded by the same UFSAR design bases for spent fuel pool (SFP) and reactor cavity water levels as the current accident analysis. Because there were no facility modifications proposed to support this amendment request, there was an expectation that the existing facility design basis would be sufficient if the proposed accident analysis would be no less bounded by the existing design than the current accident analysis.

Additional Information for Auxiliary Building - Spent Fuel Pool

Section 9.7.2.6.6, "Auxiliary Building Cranes," of the CNP UFSAR (Reference 4) describes load restrictions applicable to the auxiliary building. The loads carried over the spent pool racks and the heights at which they may be carried over spent fuel racks containing fuel shall be limited in such a way as to preclude impact energies over 55,800 inch-pounds if the loads were to be dropped from the crane. Loads in excess of 2,500 pounds shall be prohibited from travel over stored fuel assemblies in the storage pool. The restrictions ensure that the activity release will be limited to that contained in a single fuel assembly.

Additionally, per Procedure 12-OHP-4050-FHP-006 (Reference 5), "Control of Loads Over the Spent Fuel Pool and Fuel/Insert Handling in the Spent Fuel Pool", the following pertinent Technical Specification (TS) and UFSAR required conditions will be maintained during crane operations with loads over the storage pool:

- SFP Exhaust Ventilation OPERABLE [TS 3.7.13]
- SFP level ≥ 23 ft [TS 3.7.14]
- Auxiliary Building integrity [TS 3.7.13]
- Unit 1/2 Control Room (CR) Emergency Ventilation System components are in compliance with TS 3.7.10
- Unit 1/2 CR Air Condition System components are in compliance with TS 3.7.11

Additional Information for Containment – Reactor Cavity

When fuel is present in the open reactor vessel, the vessel is considered a containment restricted load area per Figure 1 of Procedure PMP-4050-CHL-001 (Reference 6), "Control of Heavy Loads." Per Reference 6, loads are prohibited from being moved within the containment restricted load area other than the following:

a) Those loads that are required to refuel the reactor such as shoe horns, lights, cameras, poles and other loads necessary to reassemble the reactor such as the reactor vessel head and upper internals.

b) In-service inspection tools, missile shields, cavity bulkhead sections, and in-containment equipment (including reactor coolant pump internals assembly and motor).

If need arises to handle a load other than those listed above, then information must be prepared and incorporated into a Plant Operations Review Committee approved procedure. Additionally, if a heavy load is being moved over irradiated fuel in the reactor vessel during core alterations or with the upper internals removed, then the capability of isolating the containment must be ensured.

Though items may be moved within the containment restricted load area, programs are in place to ensure safe movement of heavy loads. UFSAR Section 12.2.1, "Control of Heavy Loads," describes the program established to comply with CNP commitments to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Reference 9). The section identifies the features of the program that limit the possibility of a load drop by a number of cranes at the plant, including the polar crane and the east and west auxiliary building cranes. As noted in Section 12.2.1.8, the Control of Heavy Loads Program was submitted to NRC and approved on September 20, 1983. The approved controls make the probability of a load drop very unlikely.

UFSAR Section 14.2.1.3, "Shock Absorbing Characteristics," describes the design of the fuel assembly and its inherent ability to absorb impact energy during refueling operations without failing. This section also describes previous analyses that have been performed by Westinghouse wherein a fuel assembly is dropped on top of another and where a fuel assembly strikes a sharp object. The results showed that the buckling load at the bottom section of the fuel rod, which would receive the highest loadings, was below the critical buckling load and the stresses were relatively low and below the yield stresses.

For the case where one assembly is dropped on top of another fuel assembly, the loads will be transmitted through the end plates and the control rod cluster guide tubes of the struck assembly before any of the loads reach the fuel rods. The end plates and guide thimbles absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. Also, energy is absorbed in the struck assembly top end plate before any load can be transmitted to the fuel rods. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling loads, and that the stresses in the cladding were relatively low and below yield.

The wording in the current CNP TS (Reference 12), which limits applicability to the movement of irradiated fuel assemblies, was taken directly from the NRC-approved Standard Technical Specifications, NUREG-1431, Rev. 2 (Reference 10), when the plant transitioned to Improved Technical Specifications in 2005. The basis for this requirement in Reference 10 is the premise that the design basis FHA must involve the dropping of an irradiated fuel assembly. This presumption that the FHA is based on an irradiated fuel assembly is reiterated in Section 14.2.1.6 of Reference 4. The radioactive release is deterministically based on an assumed cladding rupture of fuel rods equivalent to that contained in one fuel assembly. This assumption conservatively bounds any credible radioactive release from a dropped load onto an irradiated fuel assembly.

RAI-ARCB-9

The Donald C. Cook Nuclear Plant (CNP) licensing basis includes analyses for the radiological consequences of a ruptured gas decay tank and a ruptured volume control tank. Regulatory Guide (RG) 1.183 does not provide any guidance relative to either accident. Guidelines for the waste gas decay tank and volume control tank rupture analyses are given in Branch Technical Position 11-5 (BTP 11-5) of NUREG-0800, Standard Review Plan (SRP), with additional guidance contained in Regulatory Issue Summary (RIS) 2006-04.

Item 11 of RIS 2006-04 states the following:

> *As part of full AST [alternative source term] implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this accident, they have proposed acceptance criteria of 500 millirem (mrem) total effective dose equivalent (TEDE).*

The acceptance criteria for this event is that associated with the dose to an individual member of the public as described in 10 CFR [Title 10 of the Code of Federal Regulations] Part 20 , "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

The licensee has proposed that these event-types be evaluated against the existing 500 mrem acceptance criterion for both the exclusion area boundary (EAB) and low population zone (LPZ). Item 11 of RIS-2006-04 only allows the use of the existing current licensing basis acceptance criterion of 500 mrem whole body at the EAB when these types of events are not submitted as part of the AST implementation. Since the licensee has submitted these event-types as part of the full AST implementation, the results of both accidents should be evaluated against the revised 100 mrem acceptance criterion for both the EAB and LPZ.

- *Revise the waste gas decay tank rupture and volume control tank rupture analyses in accordance with the guidance provided in RIS-2006-04, which states that dose evaluations at the EAB and LPZ should be evaluated against the revised 100 mrem TEDE acceptance criterion.*

I&M Response to RAI-ARCB-9:

In Reference 11, the NRC approved the use of AST at CNP for CR habitability accident analysis. Included in that approval was use of AST for analysis of the effects of volume control tank (VCT) rupture and waste gas decay tank (WGDT) rupture accidents on CR habitability. In preparing the revision of the CNP accident analyses proposed by Reference 3, the WGDT rupture and VCT rupture events were evaluated against the acceptance criteria provided in 10 CFR 50.67 for their effect on CR habitability and off-site dose consequences to apply a consistent source term to all of the radiological consequence analyses described in the CNP UFSAR.

After evaluating the results of the analyses, as noted in Section 3 of Enclosure 2 of Reference 3, I&M determined that the CR habitability analysis will continue to use AST for the WGDT rupture and VCT rupture event dose analyses, but the use of AST for offsite dose consequence analyses of the exclusion area boundary and low population zone would not be requested. This approach is in accordance with the existing CNP licensing basis and is consistent with full implementation of AST based on the description in Regulatory Position 1.2.1 of Reference 1, which says that full implementation of the AST revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the total effective dose equivalent dose as the new acceptance criteria. A description of selective implementation is found in Regulatory Position 1.2.2 of Reference 1, which states that selective implementation is a modification of the facility design basis that is based on one or more of the characteristics of the AST.

Based on these descriptions, and with each proposed accident analysis using all characteristics of AST, full implementation is being requested for the design basis accidents as identified in Reference 3. Further clarification regarding VCT and WGDT is found in Footnote 2 of Reference 8, which states that "(a)n off-gas or waste gas system release does not need to be addressed for a full AST implementation unless a design change is being proposed for the waste gas tank or systems at the same time."

Because the accident analyses for VCT rupture and WGDT rupture each have a separate analysis for CR habitability and off-site dose consequences, and no design change is being proposed, it is consistent with full implementation to continue using AST as the licensing basis for CR habitability and to use the current source term for off-site dose analysis.

REFERENCES

1. U. S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. "Fort Calhoun Station, Unit No. 1 – Issuance of Amendment (TAC No. MB1221)," dated December 5, 2001, Agencywide Documents Access and Management System (ADAMS) Accession No. ML013030027.

3. AEP-NRC-2014-65, "License Amendment Request to Adopt TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification" and Implement Full-Scope Alternative Source Term," November 2014, ADAMS Accession No. ML14324A209.

4. D.C. Cook Updated Final Safety Analysis Report, Revision 26.

5. Procedure 12-OHP-4050-FHP-006, Rev. 13, "Control of Loads Over the Spent Fuel Pool and Fuel/Insert Handling in the Spent Fuel Pool," September 2015.

6. Procedure PMP-4050-CHL-001, Rev. 15, "Control of Heavy Loads," May 2015.

7. Prairie Island License Amendment Request L-PI-04-001, "Selective Scope Implementation of Alternate Source Term for Fuel handling Accident Applied to Containment Technical Specifications," dated January 20, 2004, ADAMS Accession No. ML040270067.

8. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006.

9. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

10. NUREG-1431, Rev. 2, "Standard Technical Specifications Westinghouse Plants," June 2001.

11. Letter from NRC to Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 And 2 - Issuance of Amendments (TAC Nos. MB5318 AND MB5319)," dated November 14, 2002 (ADAMS Accession No. ML022980619).

12. D.C. Cook Units 1 and 2 Technical Specifications, Revs. 51 (U1) and 48 (U2).