From: Dietrich, Allison
Sent: Tuesday, September 15, 2015 5:07 PM
To: 'hlkish@aep.com'
Cc: 'dmburgoyne@aep.com'; Terry L Curtiss (tlcurtiss@aep.com); Pelton, David; Shoop, Undine; Dean, Jeremy; Dickson, Elijah
Subject: D.C. COOK UNITS 1 AND 2 - ARCB RAI CONCERNING LAR TO ADOPT TSTF-490 AND IMPLEMENT FULL-SCOPE AST (MF5184 AND MF5185)
Attachments: ARCB RAI regarding Alternate Source Term MF5184 MF5185_final.pdf

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) submitted a license amendment request for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendment consists of adoption of Technical Specifications Task Force (TSTF)-490, Revision 0, and implementation of a full scope alternate source term radiological analysis methodology.

The U.S. Nuclear Regulatory Commission (NRC) staff in the Radiation Protection and Consequence Branch of the Office of Nuclear Reactor Regulation is currently reviewing your submittal, as supplemented. The staff has determined that additional information is needed in order to complete the review, as described in the attached Request for Additional Information (RAI). The draft RAI was sent to I&M via electronic mail on August 4, 2015. Clarification telephone conferences were held on August 17, 2015 and September 10, 2015. Based on our discussions, we understand that a response will be provided by November 16, 2015 for RAI-ARCB-1, -3, -4, -5, -6, and -7, and that a response will be provided by December 16, 2015 for RAI-ARCB-2 and -8.

Please let me know if you have any questions or concerns.

Sincerely,

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
301-415-2846

Hearing Identifier: NRR_PMDA
Email Number: 2386

Mail Envelope Properties (Allison.Dietrich@nrc.gov20150915170700)

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Created By: Allison.Dietrich@nrc.gov

Recipients:
"'dmburgoyne@aep.com'" <dmburgoyne@aep.com>
Tracking Status: None
"Terry L Curtiss (tlcurtiss@aep.com)" <tlcurtiss@aep.com>
Tracking Status: None
"Pelton, David" <David.Pelton@nrc.gov>
Tracking Status: None
"Shoop, Undine" <Undine.Shoop@nrc.gov>
Tracking Status: None
"Dean, Jeremy" <Jeremy.Dean@nrc.gov>
Tracking Status: None
"Dickson, Elijah" <Elijah.Dickson@nrc.gov>
Tracking Status: None
"'hlkish@aep.com'" <hlkish@aep.com>
Tracking Status: None

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REQUEST FOR ADDITIONAL INFORMATION REGARDING

ADOPTION OF TSTF-490, REVISION 0

AND IMPLEMENTATION OF FULL-SCOPE ALTERNATE SOURCE TERM

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

TAC NOS. MF5184 AND MF5185

RAI-ARCB-1

Indiana Michigan Power Company (I&M, the licensee) has requested to fully implement an alternate source term (AST) methodology at Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The license amendment request (LAR) provides the evaluation of the radiological consequences of the design basis loss-of-coolant accident (LOCA) for implementation of a full-scope AST under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, using the methodology described in Regulatory Guide (RG) 1.183. Previous LARs submitted by the licensee have sought selective implementation of the AST pursuant to RG 1.183, Staff Position 1.2.2, Selective Implementation, to modify the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails re-evaluation of a limited subset of the design basis radiological analyses. The U.S. Nuclear Regulatory Commission (NRC) safety evaluations (SEs) approving the selective implementations of the AST can be found as follows:

- By letter dated October 24, 2001, the NRC issued amendment Nos. 256 for Unit 1 and 239 for Unit 2 (ADAMS Accession No. ML012690136). The amendments approved the October 24, 2000, LAR to incorporate a supplemental methodology for the analysis of steam generator overfill following a steam generator tube rupture;

- By letter dated October 24, 2001, the NRC issued amendment Nos. 257 for Unit 1 and 240 for Unit 2 (ADAMS Accession No. ML012630334). The amendments approved a portion of the June 12, 2000, LAR related to Generic Letter 99-02, "Laboratory Testing of Nuclear-grade Activated Charcoal;"

- By letter dated November 13, 2001, the NRC issued amendment Nos. 258 for Unit 1 and 241 for Unit 2 (ADAMS Accession No. ML012980378). The amendments approved a portion of the June 12, 2000, LAR to revise the fuel-handling accident with an AST pursuant to 10 CFR 50.67 using the methodology described in RG 1.183;

- By letter dated November 14, 2002, the NRC issued amendment Nos. 271 for Unit 1 and 252 for Units 2 (ADAMS Accession No. ML022980619). The amendments approved a portion of the June 12, 2000, LAR that replaced the TID-14844 accident source term used in design-basis radiological analyses for control room (CR) habitability with an AST pursuant to 10 CFR 50.67 using the methodology described in RG 1.183; and,

- By letters dated December 20, 2002, and May 2, 2003, the NRC issued amendment Nos. 273 for Unit 1 and 259 for Unit 2 (ADAMS Accession Nos. ML023470126 and ML030990094). The amendments approved the June 28, 2002, and November 15, 2002, LARs that requested an increase of the licensed reactor core power level by 1.66 percent from 3250 megawatts thermal (MWt) to 3304 MWt for each unit.

As discussed in amendment Nos. 273 and 259, the NRC staff reviewed the impact of the proposed changes on design basis accident (DBA) radiological analyses as they pertained to an increase of the licensed reactor core power level. The NRC staff concluded that for a power uprate, the radiological consequences of the DBAs would continue to be bound by the doses estimated in amendment Nos. 271 and 252. Thus, the NRC staff reaffirmed the DBA parameters applied in amendment Nos. 271 and 252 as the current licensing basis (CLB).

In reviewing and verifying the DBA radiological analyses proposed in the current LAR, the NRC staff referred to amendment Nos. 256/239 and 271/252 and the existing CNP Updated Final Safety Analysis Report, Chapter 14, radiological analysis source terms and steam release assumptions. However, the sources of the proposed parameters and modeling assumptions in the current LAR remain unclear.

As stated by the licensee:

> The majority of the input parameters originate from calculations performed for previous submittals such as CNP Units 1 and 2 license amendment numbers (Nos.) 271 and 252 for implementation of AST for control room (CR) habitability and license amendment Nos. 256 and 239 to address steam generator tube rupture (SGTR) overfill. Other inputs are obtained from projects implemented under 10 CFR 50.59, such as the Unit 1 Replacement Steam Generator (SG) modification as documented in the applicable annual report, as well as values currently presented in the CNP Units 1 and 2 Updated Final Safety Analysis Report. The CR habitability and offsite dose consequence analyses were revised in 2011 and implemented under 10 CFR 50.59 as documented in the applicable annual report.

Furthermore:

> Errors were introduced into the CLB dose consequence re-analyses performed in the 2007-2010 time-frame at CNP, subsequent to the previous control room habitability AST LAR (ADAMS Accession No. ML022980619) [amendment Nos. 271 and 252]. The errors are being managed via D.C. Cook's corrective actions program and have been appropriately assessed for operability.

The NRC staff requests the following information in a table format:

a) List the CR habitability and offsite dose consequences parameters and modeling assumptions for each amendment and each subsequent 10 CFR 50.59 change after the approved amendments discussed above;

b) Explain, justify, and provide a reference (ADAMS Accession number if available) for each value in Table 1: Control Room Parameters that was obtained from projects implemented through the use of the 10 CFR 50.59 process that were not already approved in the amendments discussed above;

c) Explain how errors introduced into the CLB radiological consequence re-analyses performed in the 2007-2010 time-frame subsequent to amendment Nos. 271 and 252 affect the values in Table 1: Control Room Parameters; and,

d) Describe the new CR ventilation modeling in each mode of operation utilizing the CR parameters, since the LAR modeling description differs from that discussed in amendment Nos. 271 and 252.

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.

RAI-ARCB-3

The following statement is made in Section 2.4 of the LAR:

> In addition to the dose from contamination of the control room atmosphere by intake or infiltration, the total control room dose also requires consideration of direct shine dose contributions from control room filters, from the external radiation plume, and from radioactive material in the containment building. The filter shine dose is calculated by first determining the maximum activity loading on the control room ventilation system filters during the LOCA event. This is done by considering the control room ventilation maximum fan capacity flow rate along with filter efficiencies of 100%. The activities from the recirculation filter edit of the RADTRAD output files are then input into a MicroShield [version] 8.03 model that reflects the geometry of the control room filter housing and the recirculation air handler unit position with respect to the control room.

> Credit is taken for shielding by structural materials and attenuation in air. An integrated 30-day dose is calculated for control room personnel.

In evaluating the LAR, the NRC staff could not thoroughly review and perform confirmatory calculations with the assumptions and methodologies described in the licensee's computation of the direct shine dose to CR personnel. Therefore, provide the following additional information:

a) The source terms (nuclide activity on the CR ventilation system filter at the same time-steps considered in the direct shine dose analysis); and,

b) A thorough description of the CR geometry used in the computation of the direct shine dose to CR personnel.

<u>RAI-ARCB-4</u>

Appendix B of RG 1.183, Regulatory Position 2.0 states:

> If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors [DFs] for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85 percent) and organic iodine (0.15 percent) species results in the iodine above the water being composed of 57 percent elemental and 43 percent organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.

Regulatory Information Summary 2006-04, "Experience with Implementation of Alternative Source Terms," clarifies that:

> Appendix B to RG 1.183, provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.

The licensee's analysis for the fuel handling accident credits a DF of 285 for elemental iodine and 1.0 for organic iodine with 23 feet of water level above damaged fuel. Confirm:

a) The DFs applied in the analysis; and,

b) The depth of water above the damaged fuel for both a drop in the containment building and in the Auxiliary Building.

RAI-ARCB-5

a) Provide the RADTRAD input files, in electronic format, for each of the AST DBAs described in the LAR.

RAI-ARCB-6

Licensees who have requested approval to fully implement an AST using the methodology described in Regulatory Guide 1.183 have also proposed modifications to the technical specifications (TSs) definition of dose equivalent I-131. Some have modified the definition to base it upon the thyroid dose conversion factors of International Commission on Radiation Protection (ICRP) Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation" or ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." Others have proposed a definition which is a combination of different iodine dose conversion factors, (e.g., RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," ICRP Publication 2, and Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steam line break and steam generator tube rupture accident analyses.

The proposed definition of dose equivalent I-131 in the LAR specifies the use of thyroid dose conversion factors from Federal Guidance Report 11. These same dose conversion factors are used in the analysis to establish the equilibrium iodine activities in the reactor coolant system source term and to determine the iodine appearance rates in the main steam line break and steam generator tube rupture events.

a) List each of the dose conversion factors being used to define the dose equivalent I-131.

RAI-ARCB-7

The licensee proposed changes to TS Limiting Condition for Operation 3.4.16 that are based on the generic changes including those in Task Specifications Task Force (TSTF), Subject TSTF-490, Revision 0, "Deletion of E bar Definition and Revision of RCS Specific Activity Tech Spec." The licensee reviewed a number of previously approved TSTF-490 requests, corresponding NRC staff requests for additional information and subsequent letters for issuance of the amendments. The licensee states that, "as a result of those reviews, a deviation was taken from TSTF-490 [as it] relates to mode applicability of surveillance requirements." The licensee elaborates that, "For Surveillance Requirement (SR) 3.4.16.1, the TSTF-490 proposed note "Only required to be performed in Mode 1" will not be added."

a) Explain why the TSTF-490 proposed note for SR 3.4.16.1, "Only required to be performed in Mode 1," will not be added.

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, 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.