Donald C. Cook Nuclear Plant, Unit 2

Safety Evaluation for Amendment No. 259

Measurement Uncertainty Recapture Power Uprate

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 259 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By application dated November 15, 2002, as supplemented February 24 and April 25, 2003, the Indiana Michigan Power Company (I&M, the licensee) requested an amendment to the Operating License and Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Unit 2. The proposed amendment would increase the licensed reactor core power level by 1.66 percent from 3411 megawatts thermal (MWt) to 3468 MWt. The proposed increase is considered a measurement uncertainty recapture (MUR) power uprate.

Specifically, the proposed changes would revise:

1. Paragraph 2.C.(1) in Facility Operating License DPR-74 to authorize operation at a steady-state reactor core power level not in excess of 3468 MWt (100-percent power).

2. The definition of RATED THERMAL POWER (RTP) in TS 1.3 to reflect the increase from 3411 MWt to 3468 MWt.

3. The maximum allowed power level in TS 3.5.2, Action b, from 3250 MWt to 3304 MWt, to increase the maximum allowable core power level with a safety injection cross-tie valve closed.

4. TS Table 3.7-1, “Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves during 4 Loop Operation,” to reflect the maximum allowed power for operation with inoperable main steam safety valves (MSSVs). With one inoperable MSSV per loop, the power reduction would be revised from 61.6 percent RTP to 60.4 percent RTP. With multiple inoperable safety valves per loop, the power reduction and associated reduction in high flux reactor trip setpoints would be revised to 43.0 percent (two inoperable MSSVs) and 25.7 percent (three inoperable MSSVs).

The February 24 and April 25, 2003, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on January 21, 2003 (68 FR 2805).

By letter dated April 18, 2003, the NRC staff issued a draft version of this safety evaluation (SE) and requested that the licensee review it to verify that factual information is accurate and complete. The licensee included its comments on the draft SE in its April 25, 2003, supplemental letter. The NRC staff has evaluated the licensee's comments and has incorporated them as appropriate. The licensee's comments did not change the NRC staff's findings or conclusions discussed in the draft SE.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. Title 10 of the *Code of Federal Regulation* (10 CFR), Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR Part 50 allows licensees to assume a power level lower than 1.02 times the licensed power level (but not less than the licensed power level), provided licensees have demonstrated that the proposed value adequately accounts for instrumentation uncertainties. In its November 15, 2002, application, the licensee proposed to use a value of 1.0034. To achieve this level of accuracy, the licensee will install the more accurate feedwater flow measurement meter described in NRC-approved Caldon, Inc. (Caldon) Topical Report ER-80P[1] and its supplement, Topical Report ER-157P.[2] (The currently installed venturi flow meter will remain in place.) The NRC staff approved Caldon Topical Report ER-80P by a safety evaluation report dated March 8, 1999. The NRC staff approved Caldon Topical Report ER-80P for licensees' use in submitting licensing applications for power level increases to 1 percent and for requesting exemptions from certain requirements of 10 CFR Part 50, Appendix K. The NRC staff approved Caldon Topical Report ER-157P by a safety evaluation report dated December 20, 2001. Caldon Topical Report ER-157P justified power level increases to 1.7 percent.

The licensee proposed to increase the power output of the plant by the difference between the 1.02 multiplier used in the existing analyses of record and the 1.0034 multiplier proposed as a result of the installation of the more accurate flowmeter. Since the analyses of record for LOCA and ECCS performance assumed a power level of 1.02 times the licensed power level, a 1.66-percent increase in power could be achieved without necessitating reanalyses of these events. Other design-basis analyses are evaluated to ensure an appropriate accounting of power level uncertainties.

By application dated June 28, 2002, the licensee requested a similar 1.66-percent MUR power uprate for D. C. Cook Unit 1. The NRC approved the 1.66-percent MUR power uprate for D. C. Cook Unit 1 by License Amendment No. 273, dated December 20, 2002. Given the many commonalities between the D. C. Cook Unit 1 and Unit 2 design and licensing bases, the

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[1] Caldon ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM™ System," March 1997

[2] Caldon ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM™ or CheckPlus™ System," October 2001

licensee utilized a similar approach for assessing the proposed D. C. Cook Unit 2 MUR power uprate as that which was previously approved by the NRC staff for the D. C. Cook Unit 1 MUR power uprate.

3.0 EVALUATION

The NRC staff's evaluation of the proposed D. C. Cook Unit 2 MUR power uprate is based on the guidance provided by NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Applications." RIS 2002-03 delineates the appropriate scope and level of detail for the review of an MUR power uprate application. In keeping with the guidance in RIS 2002-03, the NRC staff has evaluated the licensee's November 15, 2002, application by considering whether the proposed MUR power uprate conditions are bounded by existing design and licensing bases analyses. In particular the NRC staff considered whether the current analyses of record were performed at 102% of the current licensed power level (or a higher power level). Reduction in power level uncertainty through the reduced instrumentation error, as permitted by Appendix K, does not affect the results of such analyses, provided other assumptions upon which the analyses rest remain valid.

For every technical area where the proposed MUR power uprate conditions are bounded by existing design and licensing bases analyses, the NRC staff has confirmed that the proposed conditions will continue to be bounded and has provided a table which summarizes

- the topics identified in RIS 2002-03 within each primary technical area
- where the topic is addressed in Attachment 3 of the licensee's November 15, 2002, application (unless otherwise indicated)
- where the topic is addressed in the D. C. Cook Updated Final Safety Analysis Report
- references to NRC documents which describe analyses that bound the proposed conditions
- whether the topic is similar to the previously approved D. C. Cook Unit 1 MUR power uprate
- the NRC's conclusion of acceptability

The corresponding references and notes for each table immediately follow the table.

For situations where the proposed MUR power uprate conditions are not bounded by existing design and licensing bases, the licensee has performed new analyses. The NRC staff has noted each such area in the tables and has reviewed and evaluated the licensee's analyses. The NRC staff's review included an evaluation of the application of the methodologies used by the licensee for the new analyses.

In several places in this SE, the NRC staff refers to NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition," as guidance used during the review. The NRC staff notes that the SRP was used solely for general technical guidance. The licensee's November 15, 2002, application was reviewed to determine if the D. C. Cook Unit 2 licensing basis was in compliance with the Commission's regulatory requirements, not NUREG-0800.

3.1 Instrumentation and Controls

3.1.1 Regulatory Evaluation

The NRC staff's review in the area of instrumentation and controls covers (1) the proposed plant-specific implementation of the feedwater flow measurement device and (2) the power uncertainty calculations (RIS 2002-03, Attachment 1, Section I). The NRC staff's review is conducted to confirm that the licensee's use of Caldon Topical Report ER-80P and its supplement, Topical Report ER-157P, is consistent with the NRC staff's approvals of these topical reports. The NRC staff also reviews the power uncertainty calculations to ensure that (1) the proposed uncertainty value of 0.34 percent correctly accounts for the uncertainties due to power level instrumentation error and (2) the calculations meet the relevant requirements of Appendix K to 10 CFR Part 50.

3.1.2 Technical Evaluation

The generic bases for the proposed MUR power uprate are provided in Caldon Topical Report ER-80P and its supplement, Topical Report ER-157P. These topical reports document the Caldon leading edge flowmeter check (LEFM ✓™) and LEFM check plus (LEFM ✓+™) systems' abilities to achieve increased accuracy of flow and temperature measurement.

In its February 24, 2003, supplemental letter, the licensee submitted an uncertainty assessment which assesses the accuracy with which reactor core thermal power may be determined using the new flowmeter. The licensee asserts that the new flowmeter will be installed, calibrated, and maintained in accordance with the recommendations of Caldon. On the basis of the proposed installation and instrument application, the licensee anticipates a thermal power measurement uncertainty not in excess of 0.34 percent of RTP. This anticipated uncertainty limit is supported by testing of the LEFM in a piping geometry representative of the actual installed geometry, which will be reconfirmed during the commissioning process following installation. Therefore, the original 2-percent margin would be reduced to 0.34 percent, allowing for an MUR power uprate of 1.66 percent (2 percent - 0.34 percent).

In its safety evaluation reports that approved Caldon Topical Reports ER-80P and ER-157P, the NRC requested licensees to:

- address maintenance and calibration procedures that will be implemented with the incorporation of the LEFM

- address operational and maintenance history of the installed instrumentation and confirm that it is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P (this applies only to plants that currently have LEFMs installed)

- discuss the methodology used to calculate the uncertainty of the LEFM

- justify the use of ultrasonic meters (including the LEFM) that are not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation)

The licensee provided the information concerning each of the above items in its application and supplement. The NRC staff has reviewed the regulatory and technical analyses provided by the licensee. The NRC staff's evaluation is summarized in Table 3.1.2 below.

Table 3.1.2
Instrumentation and Controls - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Meets Criteria in NRC-approved Topical Reports ER-80P & ER-157P	Similar to Unit 1 MUR	NRC Staff Conclusion
Agreement with Caldon Topical Reports	I.1, I.1.A – I.1.C (pages 17 - 18)	n/a	Y (References 2, 3, 4)	Y	Acceptable
Maintenance and Calibration Procedures	I.1.D, I.1.F, I.1.G (pages 18, 19, 22, 25, 26)	n/a	Y[Notes 1, 2] (Reference 2)	Y	Acceptable
Operational and Maintenance History of the LEFM Installation	I.1, I.1.D (Criterion 2)[Note 3] (pages 17 - 19)	n/a	Y[Note 1] (References 2, 3, 4)	Y	Acceptable
Methodology used to calculate the uncertainty of the LEFM system	I.1.D (Criterion 3) (pages 21 - 22) (2/24/03 supplement)	n/a	Y[Note 1] (References 1, 2)	Y[Note 4]	Acceptable
Ultrasonic Meter Installation	I.1.D (Criterion 4) (pages 20 - 21)	n/a	Y[Note 1] (References 2, 3, 4)	Y	Acceptable

Table 3.1.2 References:

1. D. C. Cook Units 1 and 2 License Amendment Nos. 148 and 134, dated August 27, 1990 [Approved the transition to Westinghouse 17x17 VANTAGE 5 fuel and the use of Westinghouse Licensing Topical Report WCAP-11397-P-A, "Revised Thermal Design Procedure," dated April 1989]

2. D. C. Cook Unit 1 License Amendment No. 273, dated December 20, 2002 [Approved Measurement Uncertainty Recapture Power Uprate]

3. Letter from NRC, to C. L. Terry, TU Electric, "Comanche Peak Steam Electric Station, Units 1 and 2 – Review of Caldon Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety while Increasing Power Level Using the LEFM System' (TAC Nos. MA2298 and 2299)," dated March 8, 1999

4. Letter from S. A. Richards, NRC, to M. A. Krupa, Entergy, "Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station – Review of Caldon, Inc. Engineering Report ER-157P (TAC Nos. MB2397, MB2399 and MB2468)," dated December 20, 2001

Table 3.1.2 Notes:

1. The licensee will use the same generic maintenance and calibration procedures for the D. C. Unit 2 LEFM flow measurement system as those approved for D. C. Cook Unit 1. The maintenance and calibration procedures for the LEFM flow measurement system were addressed and found acceptable in the NRC staff's SE for D. C. Cook Unit 1 License Amendment No. 273.

2. D. C. Cook Unit 2 specific maintenance and calibration procedures will be developed as part of the implementation of the LEFM design change package to account for every difference between Unit 1 and Unit 2.

3. As noted in Section 3.1.2.2 of the NRC staff's SE for D. C. Cook Unit 1 License Amendment No. 273, the licensee has "committed to confirm that the installed instrumentation is representative of the LEFM system and

bounds the analysis and assumptions in the Caldon Topical Report ER-80P." The licensee has made the same commitment for D. C. Cook Unit 2. The licensee will document this information following implementation of the proposed MUR power uprate.

4. Methodology used to calculate the uncertainty of the LEFM system for the proposed D. C. Cook Unit 2 MUR power uprate is the same as that approved for D. C. Cook Unit 1. The licensee's overall statistical approach to combining uncertainties is in compliance with ANSI/ISA 67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation," February 2000.

3.1.3 Summary

The NRC staff has reviewed the licensee's proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations. The NRC staff has determined that the licensee's proposed use of Caldon Topical Report ER-80P and its supplement, Topical Report ER-157P, is consistent with the NRC staff's approvals of these topical reports. The NRC staff has also determined that the licensee has adequately accounted for the uncertainties due to power level instrumentation error in their power level uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to instrumentation and controls.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

The NRC staff's review in the area of reactor systems covers the impact of the proposed MUR power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the reactor and reactor coolant system, and (5) LOCA and non-LOCA transient analyses (RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that the licensee's analyses bound plant operation at the proposed power level and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Guidance for the NRC staff's review of reactor systems is contained in Chapters 4, 5, 6, and 15 of NUREG-0800.

3.2.2 Technical Evaluation

The NRC staff reviewed the licensee's application related to reactor systems performance and determined that existing analyses of record for many areas bound operation of the plant at the proposed MUR power level. The results of the NRC staff's review in the reactor systems area are summarized in Table 3.2.2 below. The licensee performed new residual heat removal (RHR) cooldown analyses to support the proposed MUR power uprate because the existing analyses of record did not bound proposed plant operation. The NRC staff's review of the licensee's RHR cooldown analyses is discussed in Section 3.2.2.1 of this SE. In addition, the NRC staff evaluated the impact of several recent Westinghouse Nuclear Safety Advisory Letters (NSALs) on steam generator (SG) performance. This evaluation is provided in Section 3.2.2.2 below.

Table 3.2.2 Reactor Systems - Summary of NRC Staff Review					
Topic	**Unit 2 MUR Application Section**	**UFSAR Section**	**Bounded by NRC-approved analysis**	**Similar to Unit 1 MUR**	**NRC Staff Conclusion**
Accidents and Transients Analyses of Record					
Post-LOCA Long-Term Core Cooling	II.1.3.1 (page 37)	14.3.1	Y (References 3, 4)	Y	Acceptable
Hot Leg Switchover	II.1.3.2 (page 37)	14.3.1	Y (References 3, 4)	Y	Acceptable
SG Tube Rupture – Thermal-Hydraulic Analysis	II.1.4 (page 38)	14.2.4	Y (References 2, 5, 6)[Note 2]	Y[Note 1]	Acceptable
NonLOCA Analysis					
Single Reactor Coolant Pump Locked-Rotor Accident	II.3.6 (page 46)	14.1.6.2	Y (Reference 1)	Y[Note 3]	Acceptable
Loss of External Electrical Load – Overpressure Analysis	II.3.7 (page 47)	14.1.8	Y (Reference 7)	Y[Notes 4, 5]	Acceptable
Loss of Normal Feedwater Flow and Loss of All AC Power	II.3.8 (page 47)	14.1.9 14.1.12	Y (Reference 1)	Y	Acceptable
Rupture of a Control Rod Drive Mechanism Housing	II.3.12 (page 49)	14.2.6	Y (Reference 1)	Y[Note 5]	Acceptable
RCCA Misalignment and RCCA Drop	II.3.1 (page 44)	14.1.3	Y (Reference 1)	Y	Acceptable
Partial and Complete Loss of Forced Reactor Coolant Flow	II.3.5 (page 46)	14.1.6.1	Y (Reference 1)	Y[Note 3]	Acceptable
Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	II.3.2 (page 44)	14.1.1	Y (Reference 1)	Y[Note 3]	Acceptable
Uncontrolled Boron Dilution	II.3.4 (page 46)	14.1.5[Note 6]	Y (Reference 1)	Y	Acceptable
Excessive Heat Removal Due to Feedwater System Malfunctions	II.3.9 (page 47)	14.1.10	Y (Reference 1)	Y[Notes 3, 5]	Acceptable
Excessive Load Increase Incident	II.3.10 (page 48)	14.1.11	Y (Reference 1)	Y	Acceptable
Rupture of a Steam Pipe – Core Response Analysis	II.3.11 (page 48)	14.2.5	Y (Reference 1)	Y[Note 3]	Acceptable

Table 3.2.2
Reactor Systems - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
Rupture of a Control Rod Drive Mechanism Housing MODE 3	II.3.12 (page 49)	14.2.6	Y (Reference 1)	Y[Note 5]	Acceptable
Anticipated Transients Without SCRAM	II.3.13 (page 49)	3.3.1.7	Y (References 8, 9)	Y	Acceptable
Station Blackout	II.3.14 (page 51)	8.7	Y (References 10, 11)	Y	Acceptable
Design Transients	II.4.1 (page 52)	4.1	Y[Note 7] (References 1, 12)	Y	Acceptable
Auxiliary Equipment Design Transients	II.4.2 (page 54)	4.1	Y[Note 7] (References 1, 12)	Y	Acceptable
Feedwater System Malfunctions (full-power case)	II.3.9 (page 47)	14.1.10	Y (Reference 1)	N[Notes 4, 5]	Acceptable
Loss of External Electrical Load – DNB Case	II.3.7 (page 47)	14.1.8	Y (Reference 7)	N[Notes 4, 5]	Acceptable
Uncontrolled RCCA Bank Withdrawal at Power	II.3.3 (page 45)	14.1.2	Y (Reference 1)	N[Note 4]	Acceptable
Fuel Evaluation					
Nuclear Design	IV.8.1 (page 79)	3.3	Y (References 1, 12, 13)	Y[Note 8]	Acceptable
Fuel Rod Design	IV.8.2 (page 80)	3.2.1	Y (References 1, 12, 13)	Y	Acceptable
Core Thermal-Hydraulic Design	IV.8.3 (page 80)	3.4	Y (References 1, 12, 13)	Y	Acceptable
Fuel Structural Evaluation	IV.8.4 (page 81)	3.2.1	Y (References 1, 12, 13)	Y	Acceptable
System Design					
RHR System	VI.1.3 (page 87)	9.3	N[Note 9] (References 12, 14)	N[Note 9]	Acceptable (See Section 3.2.2.1 below)
Emergency Core Cooling System	VI.1.4 (page 88)	6.2	Y (References 12, 13)	Y	Acceptable

Table 3.2.2
Reactor Systems - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
NSSS Control Systems	VI.5 (pages 95 – 98)	7.3	Y[Note 10] (Reference 12)	Y	Acceptable
NSSS Pressure Control Component Sizing	VI.1 (page 86)	4.2.2.2, 4.3.4	Y (References 12, 13)	Y	Acceptable
Low Temperature Overpressure Protection System	VI.5 (page 98)	4.2, 4.2.2.8	Y (References 13, 14, 15)	Y	Acceptable
Other					
Westinghouse NSALs (SG Water Level)	Attachment 4 (Pages 23-24)	n/a	n/a See Section 3.2.2.2 below	N	Acceptable (See Section 3.2.2.2 below)

Table 3.2.2 References:

1. D. C. Cook Units 1 and 2 License Amendment Nos. 148 and 134, dated August 27, 1990 [Approved the transition to Westinghouse 17x17 VANTAGE 5 fuel and the use of Westinghouse Licensing Topical Report WCAP-11397-P-A, “Revised Thermal Design Procedure,” dated April 1989]

2. D. C. Cook Unit 2 License Amendment No. 135, dated September 18, 1990 [Allowed Unit 2 SG stop valve closure within 8 seconds]

3. D. C. Cook Units 1 and 2 License Amendment Nos. 234 and 217, dated December 13, 1999 [Approved containment sump modification, as evaluated in Westinghouse Licensing Topical Report WCAP-15302, “Donald C. Cook Nuclear Plant Units 1 and 2, Modifications to the Containment Systems, Westinghouse Safety Evaluation (SECL 99-076, Revision 3),” dated September 1999]

4. D. C. Cook Units 1 and 2 License Amendment Nos. 236 and 218, dated December 23, 1999 [Rod cluster control assembly insertion credit following a large-break LOCA (LBLOCA)]

5. D. C. Cook Units 1 and 2 License Amendment Nos. 256 and 239, dated October 24, 2001 [Analyses to address SG tube rupture overfill]

6. D. C. Cook Units 1 and 2 License Amendment Nos. 271 and 252, dated November 14, 2002 [Alternative source term for control room habitability]

7. D. C. Cook Units 1 and 2 License Amendment Nos. 182 and 167, dated September 9, 1994 [Approved increase in main steam safety valve setpoint tolerances]

8. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, “Donald C. Cook Nuclear Plant Nos. Units 1 and 2, Compliance with ATWS [Anticipated Transient Without Scram] Rule 10 CFR 50.62 (TAC Nos. 59082 and 59083),” dated April 14, 1989

9. Letter from J. Giitter, NRC, to M. P. Alexich, I&M, “Safety Evaluation for Generic Letter 83-28, Item 4.5.3, Reactor Trip Reliability – On-Line Functional Testing of the Reactor Trip System (TAC Nos. 53971 and 53972),” dated August 16, 1989

10. Letter from T. G. Colburn, NRC, to E. E. Fitzpatrick, I&M, “Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. 68532/68533),” dated October 31, 1991

11. Letter from J. F. Stang, NRC, to E. E. Fitzpatrick, I&M, “Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. 68532 and 68533),” dated April 23, 1992

12. Safety Evaluation Report, “Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant – Units 1 and 2, Docket Nos. 50-315 and 50-0316,” dated September 10, 1973

13. Letter from NRC to Indiana and Michigan Electric Company, “Supplement 7 to Safety Evaluation Report,” dated December 23, 1977

14. D. C. Cook Units 1 and 2 License Amendment Nos. 219 and 203, dated December 10, 1997 [Approved changes to RHR automatic interlock surveillance requirements]

15. D. C. Cook Units 1 and 2 License Amendment Nos. 176 and 161, dated March 9, 1994 [Power-Operated Relief Valve and Block Valve Reliability, and Additional Low-Temperature Overpressure Protection in Response to NRC Generic Letter 90-06]

<u>Table 3.2.2 Notes</u>

1. The licensee performed the D. C. Cook Unit 2 steam generator tube rupture (SGTR) overfill analysis at a core power level of 3588 MWt, which bounds the proposed MUR power uprate; whereas, the Unit 1 SGTR overfill analysis was performed at 3250 MWt, and required a sensitivity analysis.

2. References 2 and 6 of the Table 3.2.2 above addressed radiological consequences of an SGTR; Reference 5 of Table 3.2.2 above approved the supplemental SGTR analysis.

3. The licensee performed the D. C. Cook Unit 2 analysis with a core power level of 3588 MWt, which bounds the proposed MUR power uprate conditions. For the D. C. Cook Unit 1 MUR power uprate, an assessment of the DNB cases of this event was necessary.

4. The licensee performed the D. C. Cook Unit 2 analysis with a core power level of 3588 MWt, which bounds the proposed MUR power uprate conditions. For the D. C. Cook Unit 1 MUR power uprate, reanalysis of this event was necessary.

5. For the proposed D. C. Cook Unit 2 MUR power uprate, the analyses are bounding at the core power level of 3588 MWt, so each accident analysis is evaluated in one section. However, for the D. C. Cook Unit 1 MUR power uprate, several accident analyses were divided into more than one section to clarify where certain cases were either evaluated differently, or reevaluated.

6. D. C. Cook Unit 2 UFSAR Section 14.1.5 is entitled, “Uncontrolled Boron Dilution,” whereas the D. C. Cook Unit 1 UFSAR Section 14.1.5 is entitled, “Chemical and Volume Control System Malfunction.”

7. The design transients for D. C. Cook Unit 2 were last evaluated for fuel Cycle 8, as set forth in the SE for D. C. Cook Unit 1 License Amendment No. 134, dated August 27, 1990, which approved the use of Westinghouse 17 x 17 VANTAGE 5 fuel.

8. The licensee proposed implementation for the D. C. Cook Unit 2 MUR power uprate at the beginning of core operating Cycle 14 (spring 2003), whereas the Unit 1 MUR was implemented in mid-cycle.

9. The licensee re-performed the RHR cooldown analyses to support the proposed D. C. Cook Unit 2 MUR power uprate. The revised analysis, which considers a change to the plant’s RTP only, demonstrates that the licensee will still be able to reach Mode 5 conditions within 36 hours on a single train of RHR, and the time to cool down to <140 °F with two trains of RHR available has increased from less than 20 hours to less than 23 hours. (See Section 3.2.2.1 below) For D. C. Cook Unit 1, the single-train cooldown analyses demonstrated that the plant would be able to reach Mode 5 within 36 hours and the two-train analyses already assumed a bounding initial power level of 3411 MWt.

10. The licensee is in the process of conducting steam dump/margin-to-trip final analyses for D. C. Cook Unit 2. (See Section 3.2.2.4 below)

3.2.2.1 RHR Cooldown

Various D. C. Cook Unit 2 TSs require that the plant be capable of being placed in cold shutdown within 36 hours. This is achieved by a single train of RHR. In addition, the current licensing basis states that under normal operating conditions, RCS temperature could be reduced to 140 °F within 20 hours following a reactor shutdown using two RHR trains. The licensee revised the RHR cooldown analyses for the single and two-train scenarios since the current analyses assumed a core power level of 3411 MWt. The licensee's revised analyses used the same input assumptions, methodologies, and techniques as the current analyses, with the exception of the core power level assumptions. For the revised analyses, the licensee used a core power level of 3482 MWt, which bounds the proposed MUR core power uprate of 3468 MWt.

The licensee's revised analyses show that for a single-train cooldown, the TS requirement of 36 hours is met. The results of the dual train cooldown demonstrate that the plant could be cooled down to 140 °F within 23 hours. The 20-hour cooldown time in the licensing basis of the plant for dual-train operation is based on economic considerations only (i.e., balancing the time required for cooldown against the size and cost of RHR and component cooling water system components, such as heat exchangers, pumps, and valves). Therefore, an increase in the cooldown time from 20 hours to 23 hours would not impact the safety of the plant.

The NRC staff confirmed that the input assumptions (other than thermal power) remain unchanged, and that the licensee's analyses bound the proposed MUR power uprate conditions. Since the revised analyses accounted for the increase in the power level to 3468 MWt, the TS requirement of 36 hours continues to be satisfied for the single-train cooldown, and the new dual-train cooldown time results will be incorporated in the UFSAR, the NRC staff finds the RHR system acceptable for operation at the proposed power level of 3468 MWt.

3.2.2.2 Steam Generators

The Westinghouse Model 51 designed SGs originally installed in D. C. Cook Unit 2 were modified in 1988. Specifically, the lower assembly (including the tube bundle) was replaced with those of a Model 54F design while the upper shell and internals remained the original Model 51 design with upgraded internals. The modified SGs have been analyzed to design specifications for 3425 MWt and 3600 MWt NSSS power operating conditions. The licensee performed a comparison of the applicable MUR power uprate design transient set to the set of values evaluated for the modified SGs at the 3600 MWt operating condition.

Westinghouse issued three NSALs (NSAL-02-3 and Revision 1, NSAL-02-4, and NSAL-02-5) to document potential problems with the Westinghouse-designed SG water level setpoint uncertainties. NSAL-02-3 and its revision, dated February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG water level measurements. These uncertainties affect the low-low level trip setpoint. NSAL-02-4, dated February 19, 2002, deals with a potential indication inaccuracy with the SG water level high-high trip setpoint for water levels above the SG mid-deck plate. NSAL-02-5, dated February 19, 2002, involves the potential effects of the pressure differential across the SG mid-deck plate, with the focus on the potential impact to the initial SG water level modeled in the accident analyses due to increased water level uncertainty.

D. C. Cook Unit 2 SGs were affected by the issue identified in NSAL-02-3. The licensee performed an assessment of this issue at the proposed uprated power level and determined that adequate margin is available in the SG water level low-low trip setpoint calculation to accommodate the effects of a differential pressure across the SG mid-deck plate. The licensee determined that the existing calculation bounds the issue identified by NSAL-02-3 and the proposed MUR conditions and found the SG water level low-low trip setpoint remains unaffected. Consequently, there is no effect on the setpoint values used in the analyses of record for the LOCA, non-LOCA transients, and the anticipated transient without scram event.

The licensee determined that the D. C. Cook Unit 2 water level low-low trip setpoint would be reached before the SG water level would reach the mid-deck plate level. Thus, the indication inaccuracy for water levels above the mid-deck plate is not of concern for D. C. Cook Unit 2, and the existing SG water level high-high trip setpoint remains acceptable. There is no effect on the setpoint values used in the analyses of record, and the current analyses remain conservative.

The assessment of the NSAL-02-4 issue by the licensee determined that the D. C. Cook Unit 2 trip setpoint would be reached before the SG water level would reach the mid-deck plate level. The indication inaccuracy for water levels above the mid-deck plate is not of concern for Unit 2, and the existing SG water level high-high trip setpoint remains acceptable. Thus, there is no effect on the setpoint values used in the analyses of record, and the current analyses remain conservative.

The NSAL-02-5 issue pertained to the potential impact to the initial SG water level modeled in the accident analyses due to increased water level uncertainty. The increased uncertainty is a possible result of postulated pressure differential effects across the SG mid-deck plate. The specific accident analyses of interest are (1) loss of normal feedwater/loss of all AC power to the station auxiliaries, (2) feedwater malfunction, (3) feedline break, (4) steamline break mass and energy release calculations, and (5) LOCA mass and energy release calculations. The licensee performed an assessment of the postulated condition and determined, in all cases, that the conclusions of the current analyses remain applicable and bounding due to existing available margin. The licensee found that the current analyses of record continue to remain bounding. Thus, the current analyses remain conservative and support the proposed MUR power uprate.

The NRC staff reviewed the licensee's assessments of the NSALs discussed above and finds that the licensee's programs for reviewing vendor recommendations has adequately addressed any impact of the issues on the design and operation of the SGs. The NRC staff further finds that the current analyses remain conservative with respect to the proposed D. C. Cook Unit 2 MUR power uprate. Therefore, the NRC staff concludes that the SG water level issues are adequately addressed for the uprated power.

3.2.3 Summary

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed MUR power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the NSSS, and (5) LOCA and non-LOCA transient analyses. The NRC staff has determined that the results of licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Where additional assessments and analyses were necessary, the NRC staff has reviewed these assessments and analyses and finds that the licensee has satisfactorily addressed the areas discussed above, the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results meet the applicable acceptance criteria. Based on the above, the NRC staff finds the proposed MUR 1.66-percent power uprate acceptable with respect reactor systems performance.

3.3 Electrical Systems

3.3.1 Regulatory Evaluation

The NRC staff's review in the area of electrical engineering covers the impact of the proposed MUR power uprate on (1) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformer/reserve auxiliary transformer, (2) emergency diesel generator loading, (3) station blackout, and (4) environmental qualification of electrical equipment (RIS 2002-03, Attachment 1, Section V). This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, 10 CFR 50.63, and 10 CFR 50.49 following implementation of the proposed MUR power uprate.

3.3.2 Technical Evaluation

The NRC staff has reviewed the licensee's application in relation to electrical system performance and determined that existing analyses of record for electrical systems bound the proposed operation of the plant at the uprated power level. The results of the NRC staff's review in the electrical engineering area are summarized in Table 3.3.2 below.

Table 3.3.2
Electrical Systems - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
Grid Stability	V (page 85)	8.5	Y (References 1, 2)	Y	Acceptable
Main Generator	V (pages 83, 84)	8.0 10.3	Y[Note 1] (References 1,2, 3)	Y	Acceptable
Main Transformer	V (page 84)	8.2	Y (References 1, 2, 4, 5)	Y	Acceptable
Isophase Bus	VI.4, VII.3 (pages 95, 101)	8.1.2 10.7	Y[Note 1] (References 1, 2)	Y	Acceptable
Unit Auxiliary Transformer / Reserve Auxiliary Transformer	Table V-1 (page 83)	8.0 8.1.2	Y (References 1, 2, 4, 5)	Y	Acceptable

Table 3.3.2 Electrical Systems - Summary of NRC Staff Review					
Topic	**Unit 2 MUR Application Section**	**UFSAR Section**	**Bounded by NRC-approved analysis**	**Similar to Unit 1 MUR**	**NRC Staff Conclusion**
Emergency Diesel Generators	V (page 85)	8.5 9.8.3	Y (References 1, 2, 6)	Y	Acceptable
Station Blackout	II.3.14, V (page 51, 86)	8.7	Y (References 3, 7)	Y	Acceptable
Environmental Qualification of Electrical Equipment	V, VII.6.1 (page 86)	14.4	Y[Note 2] (References 1, 8)	Y	Acceptable

<u>Table 3.3.2 References:</u>

1. Safety Evaluation Report, "Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant – Units 1 and 2, Docket Nos. 50-315 and 50-0316," dated September 10, 1973

2. Letter from NRC to Indiana and Michigan Electric Company, "Supplement 7 to Safety Evaluation Report," dated December 23, 1977

3. Letter from T. G. Colburn, NRC, to E. E. Fitzpatrick, I&M, "Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. 68532/68533)," dated October 31, 1991

4. D. C. Cook Unit 2 License Amendment No. 22 , dated July 10, 1980 [Approved changes to surveillance and monitoring requirements for degraded voltage]

5. D. C. Cook Units 1 and 2 License Amendment Nos. 137 and 124, dated May 25, 1990 [Approved changes to allowable values for 4KV bus degraded voltage]

6. D. C. Cook Units 1 and 2 License Amendment Nos. 214 and 199, dated March 13, 1997 [Approved an increase in SG plugging limit]

7. Letter from J. F. Stang, NRC, to E. E. Fitzpatrick, I&M, "Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. M68532 and 68533)," dated April 23, 1992

8. Letter from S. A. Varga, NRC, to J. Dolan, I&M, "Safety Evaluation Regarding Environmental Qualification of Electric Equipment Important to Safety," dated January 11, 1985

<u>Table 3.3.2 Notes:</u>

1. Turbine Auxiliary Cooling Water (TACW) has been determined to have adequate margin to support operation at the proposed uprated core power level. However, similar to the D. C. Cook Unit 1 MUR power uprate assessment, TACW flow to the iso-phase bus duct cooling system and stator water coolers will be monitored and adjusted during post-modification system operation to accommodate additional heat generated at the uprated power level.

2 The environmental qualification of electrical equipment is based on the results of accident analyses which assumed core power levels that have been adjusted for a 2-percent calorimetric uncertainty.

3.3.3 Summary

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed MUR power uprate on (1) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformer/reserve auxiliary transformer, (2) emergency diesel generators, (3) station blackout, and (4) environmental qualification of electrical equipment. The NRC staff has determined that the results of licensee's analyses related to these areas will remain bounding following implementation of the proposed MUR power uprate, and that those results will continue to meet the requirements of 10 CFR Part 50, Appendix A, GDC-17, 10 CFR 50.63, and 10 CFR 50.49. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to electrical engineering.

3.4 Mechanical and Civil Engineering

3.4.1 Regulatory Evaluation

The NRC staff's review in the area of mechanical and civil engineering covers the structural and pressure boundary integrity of NSSS and balance-of-plant (BOP) systems and components (RIS 2002-03, Attachment 1, Section IV, Items 1.A, 1.B, and 1.D). The NRC staff's review focuses on the impact of the proposed MUR power uprate on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) reactor vessel (RV) and supports; (4) control rod drive mechanism; (5) SGs and supports; (6) reactor coolant pumps and supports; (7) pressurizer and supports; (8) reactor internals and core supports; and (9) safety-related valves. Technical areas covered by this review include stresses, cumulative usage factors, flow-induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs. The review is conducted to confirm that (1) the results of the analyses continue to meet allowable limits as defined in the American Society of Mechanical Engineers (ASME) code of record for the plant, (2) the safety-related valves will continue to perform acceptably, and (3) the safety-related valve programs will continue to be adequate. Guidance for the NRC staff's review of the topics within the mechanical and civil engineering area are contained in Chapters 3 and 5 of NUREG-0800.

3.4.2 Technical Evaluation

The NRC staff has reviewed the licensee's application as related to the mechanical and civil engineering areas discussed above and determined that existing analyses of record bound plant operation at the proposed uprated power level. The results of the NRC staff's review in the mechanical and civil engineering area are summarized in Table 3.4.2 below.

Table 3.4.2
Mechanical and Civil Engineering - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
RV Structural Evaluation	IV.1, IV.1.1 (pages 56, 57)	4.2.2.1 4.4	Y[Note 1] (References 1, 2)	Y	Acceptable

Table 3.4.2
Mechanical and Civil Engineering - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
Reactor Internals	IV.1.2 (page 57)	3.2.2 4.2.2.1	Y (References 1, 2)	Y	Acceptable
Piping and Supports	IV.2 (page 62)	4.2.2.7 4.2.2.9	Y (References 1, 2)	Y	Acceptable
Control Rod Drive Mechanisms	IV.3 (page 64)	3.2.3.1.4 3.2.3.2.2	Y (References 1, 2)	Y	Acceptable
Reactor Coolant Pumps and Motors	IV.4 (page 65)	4.2.2.5	Y (References 1, 2)	Y	Acceptable
SGs	IV.5.2, IV.5.4 (pages 69, 72)	4.2.2.4	Y (References 1, 2)	Y	Acceptable
Pressurizer	IV.6 (page 78)	4.2.2.2	Y (References 1, 2)	Y	Acceptable
NSSS Auxiliary Equipment	IV.7 (page 79)	4.2.2.3 4.2.2.8 Chapter 9	Y (References 1, 2)	Y	Acceptable
Balance of Plant					
Main Steam System	VI.2.1 (page 89-90)	10.2	Y (References 1, 2, 3)	Y	Acceptable
Steam Dump System	VI.2.1 (page 89-90)	7.3.2 10.2	Y[Note 2] (References 1, 2)	Y	Acceptable
Condensate and Feedwater System	VI.2.2 (page 90-91)	10.5.1	Y (References 1, 2)	Y	Acceptable
Auxiliary Feedwater System	VI.2.3 (page 91-92)	10.5.2	Y (References 1, 4)	Y	Acceptable
SG Blowdown System	VI.2.5 (page 92-93)	10.11	Y (References 1, 2)	Y	Acceptable
Programs					
High-Energy Line Break Program	VII.6.5 (page 106)	5.2.2.7 14.4.11.2	Y (References 5, 6)	Y	Acceptable
Motor-Operated Valve Program[Note 3]	VII.6.2 (page 103-104)	8.1.2	Y [Note 4, 5] (References 7, 8, 9)	Y	Acceptable
Air and Hydraulic Operated Valve Program	VII.6.3 (page 104)	n/a	Y [Note 6] (Reference 7)	Y	Acceptable

Table 3.4.2 References:

1. Safety Evaluation Report, “Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant – Units 1 and 2, Docket Nos. 50-315 and 50-0316,” dated September 10, 1973

2. Letter from NRC to Indiana and Michigan Electric Company, “Supplement 7 to Safety Evaluation Report,” dated December 23, 1977

3. D. C. Cook Units 1 and 2 License Amendment Nos. 182 and 167, dated September 9, 1994 [Approved an increase in MSSV setpoint tolerances]

4. D. C. Cook Units 1 and 2 License Amendment Nos. 214 and 199, dated March 13, 1997 [Approved an increase in SG plugging limit]

5. D. C. Cook Units 1 and 2 License Amendment Nos. 244 and 225,” dated April 25, 2000 [Approved modification to turbine-driven auxiliary feedwater pump room cooler plant]

6. D. C. Cook Units 1 and 2 License Amendment Nos. 249 and 230,” dated November 21, 2000 [Approved changes for high-energy line break methodology]

7. D. C. Cook Unit 1 License Amendment No. 273, dated December 20, 2002 [Approved Measurement Uncertainty Recapture Power Uprate]

8. Letter from M. W. Rencheck, I&M, to NRC Document Control Desk, “Donald C. Cook Nuclear Plant Units 1 and 2 Completion of Generic Letter (GL) 88-10 Motor-Operated Valve (MOV) Program Implementation and Description of Generic Letter 96-05 MOV Periodic Verification Program, [C1200-09],“ dated December 15, 2000

9. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, “Donald C. Cook Nuclear Plant, Units 1 and 2 – Closeout of Licensing Action for Generic Letter 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves (TAC Nos. M97037 and M97038)” dated August 8, 2001

Table 3.4.2 Notes:

1. The operating envelope (pressure-temperature (P-T)) evaluated for the D. C. Cook Unit 2 MUR power uprate is bounded by the envelope evaluated for fuel Cycle 8 (D. C. Cook License Amendment No. 134, dated August 27, 1990). Therefore, the RV structural analyses and evaluations that demonstrate compliance with applicable limits of Section III of the ASME *Boiler and Pressure Vessel Code* remain valid.

2. The licensee is in the process of conducting steam dump/margin-to-trip final analyses for D. C. Cook Unit 2. (See Attachment 5, “Regulatory Commitments,” of November 15, 2002, application.) The licensee will make changes/adjustments as necessary to ensure that valves have sufficient capacity prior to implementing the proposed 1.66-percent power uprate.

3. A description of the D. C. Cook MOV Program was provided to the NRC in a letter dated December 15, 2000 (Reference 9 of Table 3.4.2 above).

4. Impacts to the D. C. Cook MOV Program were addressed in the SE for D. C. Cook Unit 1 License Amendment No. 273 (Reference 6 of Table 3.4.2 above). This program is common to both D. C. Cook Unit 1 and Unit 2.

5. Reference 9 of Table 3.4.2 above is the NRC’s closeout document for the MOV Program (GL 96-05), which documents the acceptance of the D. C. Cook MOV Program, based on NRC review and/or inspection.

6. Impacts to the D. C. Cook Air and Hydraulic-Operated Valve Program were first addressed in the SE for D. C. Cook Unit 1 License Amendment No. 273 (Reference 6 of Table 3.4.2 above). This program is common to both D. C. Cook Unit 1 and Unit 2.

3.4.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, cumulative usage factors, flow induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs and has determined that the current analyses of record consider conditions that bound those which would follow implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the areas mechanical and civil engineering.

3.5 Dose Consequences Analysis

3.5.1 Regulatory Evaluation

The NRC staff's review covers the impact of the proposed MUR power uprate on the results of dose consequence analyses (RIS 2002-03, Attachment 1, Sections II and III). The review is conducted to verify that the results of the licensee's dose consequence analyses continue to meet the acceptance criteria in 10 CFR Part 100, 10 CFR 50.67, and/or 10 CFR Part 50, Appendix A, GDC-19, as applicable, following implementation of the proposed MUR power uprate.

3.5.2 Technical Evaluation

The NRC staff reviewed the impact of the proposed MUR power uprate changes on design-basis accident (DBA) radiological analyses, as documented in Chapter 14 of the D. C. Cook UFSAR. In its November 15, 2002, application, the licensee stated that the current radiological analyses of record for D. C. Cook Unit 2 were unaffected by the proposed MUR power uprate because they were performed assuming a nominal core power of 3588 MWt, which bounds the conditions for the proposed 1.66-percent power uprate. Using the current D. C. Cook UFSAR documentation in addition to information in the November 15, 2002, application, the NRC staff verified that the existing D. C. Cook Unit 2 UFSAR Chapter 14 radiological analyses source term and steam release assumptions, as appropriate, bound the proposed 1.66-percent power uprate conditions for analyses of the offsite radiological consequences of DBAs.

By D. C. Cook Units 1 and 2 License Amendment Nos. 258 and 241, dated November 13, 2001, and License Amendment Nos. 271 and 252, dated November 14, 2002, the NRC staff approved selective implementation of an alternative source term in accordance with 10 CFR 50.67. License Amendment Nos. 258 and 241 addressed the fuel handling accident only. License Amendment Nos. 271 and 252 addressed control room dose only. In its analyses, which were submitted in the applications for License Amendment Nos. 258, 241, 271, and 252, the licensee assumed a core power level of 102 percent of 3588 MWt (or 3660 MWt) for the revised analyses, which bounds the conditions for the proposed 1.66-percent power uprate for D. C. Cook Unit 2 for the fuel handling accident and control room doses. The NRC staff found the licensee's analyses to be acceptable, as stated in the SEs for License Amendment Nos. 258, 241, 271, and 252.

3.5.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on dose consequence analyses. As set forth above, the NRC staff has determined that the results of licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to dose consequence analyses.

3.6 Materials and Chemical Engineering

3.6.1 Regulatory Evaluation

The NRC staff's review in the area of materials and chemical engineering covers the effects that the proposed MUR power uprate would have on (1) the structural integrity evaluations for the RV, (2) SG tube integrity, and (3) erosion/corrosion programs (RIS 2002-03, Attachment 1, Section IV, Items 1.C through 1.F). The NRC staff's review in this area focuses on the impact of proposed MUR power uprate on (1) the P-T limits for the RV and reactor coolant pressure boundary, (2) evaluations for ensuring the integrity of the RV and reactor coolant pressure boundary against pressurized thermal shock (PTS), (3) evaluations for ensuring that the RV materials have sufficient levels of upper-shelf energy (USE), (4) surveillance capsule withdrawal schedules, (5) licensee programs for addressing SG tube degradation mechanisms, and (6) erosion/corrosion. This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50.55a; and 10 CFR Part 50, Appendices G and H, following implementation of the proposed MUR power uprate. Additional guidance for the NRC staff's review of the topics within the materials and chemical engineering area include the guidance contained in Chapters 4, 5, and 6 of NUREG-0800.

3.6.2 Technical Evaluation

The NRC staff has reviewed the licensee's application as related to the materials and chemical engineering areas discussed above and determined that, with the exception of the structural integrity evaluations for PTS and RV USE, the existing analyses of record bound the proposed operation of the plant at the uprated power level. The NRC staff's evaluation of the effects of the proposed MUR power uprate on the PTS and RV USE analyses is given in Section 3.6.2.1 of this SE. The results of the NRC staff's review for the remaining areas within the materials and chemical engineering scope are summarized in Table 3.6.2 below.

Table 3.6.2
Materials and Chemical Engineering - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
Component Integrity					
SG Structural Integrity Evaluation	IV.5.2, IV.5.3 (pages 69-72)	n/a[Note 1]	Y (References 1, 2, 3)	Y	Acceptable

Table 3.6.2
Materials and Chemical Engineering - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
SG Tube Vibration and Wear and Other Modes of Tube Degradation	IV.5.4 (pages 72-76)	n/a[Note 1]	Y (References 2, 3)	Y[Note 2]	Acceptable
Regulatory Guide 1.121 Analysis	IV.5.5 (pages 76-78)	n/a[Note 1]	N	Y[Note 3]	Acceptable (See Section 3.6.2.2 below)
Flow-Accelerated Corrosion	VII.6.4 (pages 104-106)	n/a	Y (Reference 4)	Y[Note 4]	Acceptable
Structural Integrity and Metallurgy					
10 CFR Part 50 Appendix G – P-T Limits	IV.1.1	4.2.5 4.2.6 4.4.1	Y[Note 5] (Reference 5)	Y[Note 5]	Acceptable
10 CFR Part 50 Appendix G - USE	IV.1.1	3.3.2.8 4.2.2.8	N	Y[Note 5]	Acceptable (See Section 3.6.2.1 below)
10 CFR 50.61 PTS Events	Enclosure 2, Section 5.2	3.3.2.8 4.2.2.8 4.4.2 14.3.7	N	Y[Note 5]	Acceptable (See Section 3.6.2.1 below)
10 CFR Part 50 Appendix H RPV Surveillance Program	IV.1.1	4.5.1.1	Y (Reference 1)	Y	Acceptable
Leak-Before-Break Analyses	IV.2.3	4.3.1 5.2.2.7 6.1 14.3.3.1 14.3.3.4 14.3.3.6	Y (References 6, 7)	Y	Acceptable
Structural Integrity of Control Rod Drive Mechanism Nozzles	IV.3	3.2.3.1.4 3.2.3.2.2 4.3.1 14.3.3	Y (References 1, 2, 6)	Y	Acceptable
Structural Integrity of RV Internals	IV.1.2	3.2.2 4.2.2.1	Y (References 1, 2, 6, 8)	Y	Acceptable
Structural Integrity of the Reactor Coolant Pump Flywheels	IV.4	4.2.2.5	Y (References 1, 2)	Y	Acceptable
Structural Integrity of Other Class 1 Reactor Coolant System Components	IV.1.2, IV.2.3, IV.3, IV.4	3.2.2 3.2.3.1.4 3.2.3.2.2 4.2.2.5, 4.3 14.3.3	Y (References 1, 2, 6, 7)	Y	Acceptable

Table 3.6.2 References:

1. Safety Evaluation Report, "Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant – Units 1 and 2, Docket Nos. 50-315 and 50-0316," dated September 10, 1973

2. Letter from NRC to Indiana and Michigan Electric Company, “Supplement 7 to Safety Evaluation Report,” dated December 23, 1977

3. D. C. Cook Unit 2 License Amendment No. 100, dated March 8, 1988 [Approved changes for the Steam Generator Repair Program]

4. D. C. Cook Unit 1 License Amendment No. 273, dated December 20, 2002 [Approved Measurement Uncertainty Recapture Power Uprate]

5. D. C. Cook Unit 2 License Amendment No. 255, dated March 20, 2003 [Approved revisions to P-T limits]

6. D. C. Cook Units 1 and 2 License Amendment Nos. 236 and 218, dated December 23, 1999 [Rod cluster control assembly insertion credit following a large-break LOCA (LBLOCA)]

7. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, “Donald C. Cook Nuclear Plant, Units 1 and 2 – Review of Leak-Before-Break for the Pressurizer Surge Line Piping as Provided by 10 CFR Part 50, Appendix A, GDC-4 (TAC Nos. MA7834 and MA7835),” dated November 8, 2000

8. D. C. Cook Units 1 and 2 License Amendment Nos. 148 and 134, dated August 27, 1990 [Approved the transition to Westinghouse 17x17 VANTAGE 5 fuel and the use of Westinghouse Licensing Topical Report WCAP-11397-P-A, “Revised Thermal Design Procedure,” dated April 1989]

Table 3.6.2 Notes:

1. The detailed SG component integrity analyses and evaluations are beyond the level of detail presented in the D. C. Cook UFSAR.

2. The D. C. Cook Unit 2 SG tube vibration and wear evaluation quantifies the results in terms of the fluidelastic stability ratio, tube amplitudes of vibration, and tube wear; whereas the D. C. Cook Unit 1 evaluations used the fretting wear damage parameter to quantify the results.

3. The D. C. Cook Unit 2 analyses consider a maximum level of SG tube plugging of 10 percent; whereas the D. C. Unit 1 analyses consider a 30-percent level of SG tube plugging.

4. The FAC Program and evaluation of that program for the MUR power uprates are common to both D. C. Cook units.

5. For D. C. Cook Unit 2, the proposed MUR uprate is based on new P-T curves, which were approved by D. C. Cook Unit 2 License Amendment No. 255, dated March 20, 2003 (Reference 6 of Table 3.6.2 above). The new P-T curves are supported by revised Unit 2 RV integrity analyses that used revised neutron fluence calculations, which follow the guidance in RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” The updated P-T curves used neutron fluence projections that correspond to 3800 MWt, and thus bound the proposed MUR power uprate.

3.6.2.1 Pressurized Thermal Shock and Upper-Shelf Energy Analyses

The licensee assessed the effect that the proposed MUR power uprate would have on the structural integrity assessments for the RV in Section IV.1.1 of the November 15, 2002, application. These structural integrity assessments included the assessment of RV materials relative to PTS and USE screening criteria in 10 CFR 50.61 and 10 CFR Part 50, Appendix G, respectively. The licensee concluded that the proposed 1.66-power uprate will not have a significant effect on the structural integrity evaluations for the D. C. Cook Unit 2 RV. For D. C. Cook Unit 2, the projected end-of-license (EOL) neutron fluences for the RV are based on 32 effective full power years (EFPYs) of operation and a core thermal power level of 3800 MWt.

The NRC staff performed an independent calculation of the material property values (i.e., RT_{PTS} values) for the RV beltline using the uprated neutron fluences for the RV in order to assess what effect the proposed uprated power conditions would have on the PTS evaluations for the plant and the validity of the licensee's conclusion. For the evaluation of PTS, the beltline of the D. C. Cook Unit 2 RV is limited by the amount of embrittlement that is projected to occur in RV intermediate shell plate 10-1 (material heat No. C5556-2) at EOL. The NRC staff projected the RT_{PTS} value for intermediate shell plate 10-1 to be 215 °F based on an uprated 32 EFPY neutron fluence of 1.625 x 10^{19} n/cm^2. This meets the screening criterion in 10 CFR 50.61 for RV base metal materials (i.e., $RT_{PTS} \leq 270°F$). Based on the above, the NRC staff finds that RV beltline materials for D. C. Cook Unit 2 will continue to have a sufficient safety margin against the impacts of PTS events and concludes that the uprated PTS assessment for the D. C. Cook Unit 2 RV is acceptable.

The NRC staff performed an independent calculation of the USE values for the RV beltline materials using the uprated neutron fluences for the 1/4T location RV at EOL. For the evaluation of USE, the beltline of the D. C. Cook Unit 2 RV is limited by the USE drop that is projected to occur in RV intermediate shell plate 10-2 (material heat No. C5521-2). The NRC staff projected the EOL USE value for this material to be 67 ft-lbs based on an uprated 32 EFPY 1/4T neutron fluence of 0.968 x 10^{19} n/cm^2. This meets the screening criterion in Appendix G to 10 CFR Part 50 of 50 ft-lbs for RV beltline materials in the irradiated condition. Based on the above, the NRC staff concludes that RV beltline materials for D. C. Cook Unit 2 will continue to comply with the USE requirements in Appendix G to 10 CFR Part 50.

3.6.2.2 Regulatory Guide 1.121 Analysis

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe condition of degradation in the tubes, beyond which tubes found defective by the established inservice inspection shall be removed from service. The level of acceptable degradation is referred to as the repair limit. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit an allowance for continued growth of the flaw and an allowance for eddy current measurement uncertainty.

The licensee performed an analysis to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. The licensee's analysis assumed a 10-percent SG tube plugging level, since this configuration envelopes the primary-to-secondary pressure gradients for the zero-plugging condition. The licensee concluded that the results of the RG 1.121 analysis are acceptable for the proposed 1.66-percent power uprate. The NRC staff finds the licensee's evaluation and reasoning to be acceptable because it follows the guidance in RG 1.121, which provides a conservative assessment of SG tube degradation.

3.6.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on RV integrity, SG tube integrity, and erosion/corrosion programs. The technical areas reviewed by the NRC staff are those discussed in Section 3.6.1 of this SE. Based on the above, the NRC staff concludes that the licensee has adequately addressed these impacts and has demonstrated that the plant will continue to meet the applicable requirements following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the materials and chemical engineering issues discussed above.

3.7 Human Factors

3.7.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions (RIS 2002-03, Attachment 1, Section VII, Items 1 through 4). The NRC staff's human factors evaluation is conducted to confirm that operator performance will not be adversely affected as a result of system changes necessary for the proposed MUR power uprate. The NRC staff's review covers the licensee's plans for addressing changes to operator actions, human-system interfaces, and procedures and training necessary for the proposed MUR power uprate. The NRC's acceptance criteria for human factors are based on 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR 55.59, and GDC-19.

3.7.2 Technical Evaluation

Items 1 through 4 in Section VII of Attachment 1 to RIS 2003-03 define the scope of the NRC staff's review for the human factors area. The licensee addressed these items in its November 15, 2003, application. Following is a summary of the NRC staff's evaluation related to the human factors area.

3.7.2.1 Operator Actions

The licensee indicated that the proposed MUR power uprate is not expected to have any significant effect on the manner in which the operators control the plant during normal operations or transient conditions. The licensee also indicated that all operator actions that were taken credit for in the safety analysis would still be valid following implementation of the proposed MUR power uprate. The NRC staff finds the implementation of the proposed MUR power uprate at D. C. Cook Unit 2 will not have an adverse affect on operator actions.

3.7.2.2 Emergency and Abnormal Operating Procedures

The licensee indicated that there are currently no Emergency Operating Procedures (EOPs) that reference use of the LEFM. Specific procedures within the EOP Program may need review and revision by the licensee based upon the proposed MUR power uprate plant parameters for thermal power, temperature, and pressure values. These changes were identified and will be implemented under the design change process to implement the proposed MUR power uprate. Specifically, values in the EOPs, the EOP Footnotes document, and the Abnormal Operating Procedures (AOPs) will be revised based upon the proposed 1.66-percent power uprate level. Any changes to values that are referenced in the EOPs or AOPs will be revised by the EOP/AOP Control Program to fully implement the proposed MUR power uprate. In addition, impacts to the D. C. Cook Emergency and Abnormal Operating Procedures were addressed in the SE for D. C. Cook License Amendment No. 273. This program is common to both D. C. Cook Unit 1 and Unit 2. Based on the above, the NRC staff finds that necessary procedures will be changed or updated prior to the implementation of the license and TSs changes associated with the proposed MUR power uprate. The NRC staff finds this acceptable.

3.7.2.3 Control Room Controls, Displays, and Alarms

The licensee stated that notification of the operators of the LEFM CheckPlus system condition will be through the plant process computer (PPC). Alarms and annotation of the LEFM system status will be through the computer display PPC. The alarm will use the existing Computer Priority Alarm. This alarm functions to alert the operators of PPC points being out of service, as well as a PPC malfunction. The annunciator position on the control boards would not change. There are no new controls for the operator to manipulate. Response to this computer alarm will be proceduralized. The Alarm Response Manual would be updated accordingly. The licensee indicated that reactor operators would be trained on the changes in the PPC, alarms associated with the LEFM, and the changes in the Alarm Response Manual in a manner consistent with the licensee's design modification process. Changes to control room controls, displays, and alarms, the control room plant simulator, and the operator training program will be developed as part of the implementation of the LEFM design change package. (See Attachment 5, "Regulatory Commitments," of the November 15, 2002, application). This will be finalized prior to implementing the proposed MUR power uprate. The NRC staff finds this acceptable.

3.7.2.4 Control Room Plant Reference Simulator

The D. C. Cook Nuclear Plant Simulator Certification was submitted in a letter from M. P. Alexich, I&M, to T. E. Murley, NRC, dated August 24, 1990, pursuant to 10 CFR 55.45(b)(5). The proposed MUR power uprate is not expected to have a significant effect on any simulated systems and the simulator is not expected to be modified. If changes to the simulator are necessary, the licensee indicated that changes to the simulator associated with the MUR power uprate would be treated in a manner consistent with any other plant modification, and would be tested and documented accordingly. The NRC staff finds this acceptable.

3.7.2.5 Operator Training Program

The installation of the LEFM and implementation of the proposed 1.66-percent MUR power uprate would necessitate procedure and training changes. Actions would be added to the appropriate operating procedures and an Administrative Technical Requirement would be developed for use in the event the LEFM system becomes unavailable. Operations training concerning the use of the LEFM, the associated procedures, and the Administrative Technical Requirement changes will be completed prior to implementation of the MUR power uprate. The NRC staff finds this acceptable.

3.7.3 Summary

The NRC staff has reviewed the licensee's planned actions related to the human factors area and has determined that licensee has adequately considered the impact of the proposed MUR power uprate on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators' performance is not adversely affected by the proposed MUR power uprate. Accordingly, the NRC staff concludes that the licensee will continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.120, and

10 CFR 55.59 following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the human factors aspects of required system changes.

3.8 Plant Systems

3.8.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, and (7) engineered safety feature (ESF) heating, ventilation, and air conditioning systems (NRC RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that the licensee's analyses bound plant operation at the proposed MUR power level and that the results of licensee analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Guidance for the NRC staff's review of reactor systems is contained in Chapters 3, 6, 9, 10, and 11 of NUREG-0800.

3.8.2 Technical Evaluation

The NRC staff has reviewed the licensee's application as related to the plant systems areas discussed above and has determined that for most areas, existing analyses of record bound plant operation at the proposed uprated power level. The results of the NRC staff's review in the plant systems area are summarized in Table 3.8.2 below. The licensee performed new analyses for post-LOCA containment hydrogen generation. The NRC staff's evaluation of these analyses is included in Section 3.8.2.1 below.

Table 3.8.2
Plant Systems - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
Post-LOCA Containment Hydrogen Generation	II.2.2 (page 40)	14.3.6	N[Note 1] (References 3, 4, 5, 6)	N	Acceptable (See Section 3.8.2.1 below)
Long-Term LOCA Mass and Energy Release Analysis	II.2.3.1 (page 41)	14.3.4.3.1.2	Y (References 2, 7)	Y	Acceptable
Short-Term LOCA Mass and Energy Release Analyses	II.2.3.2 (page 41)	14.3.4.5.1	Y (References 7, 8, 9)	Y	Acceptable
Fire Protection Systems					
Fire Protection Evaluation	VII.6.6 (page 106)	1.0.1	Y (References 9, 10, 11, 12, 13, 14, 15, 16)	Y	Acceptable

Table 3.8.2
Plant Systems - Summary of NRC Staff Review

Topic	Unit 2 MUR Application Section	UFSAR Section	Bounded by NRC-approved analysis	Similar to Unit 1 MUR	NRC Staff Conclusion
Power/Steam Systems					
Main Steam System and Steam Dump System	VI.2.1 (pages 89, 90)	7.3.2 7.3.3 10.2	N[Note 2] (References 8, 9, 26)	Y	Acceptable
Condensate and Feedwater Systems	VI.2.2 (pages 90, 91	10.5.1	Y (References 8, 9)	Y	Acceptable
Auxiliary Feedwater System and Condensate Storage System	VI.2.3 (pages 91, 92	10.5.2	Y (References 8, 27)	Y	Acceptable
Feedwater Heaters and Drains	VI.2.4 (page 92)	10.5.1	Y (References 8, 9)	Y	Acceptable
SG Blowdown System	VI.2.5 (pages 92, 93)	10.11	Y (References 8, 9)	Y	Acceptable
Cooling and Support Systems					
Component Cooling Water System	VI.3 (page 93)	9.5	Y[Note 3] (Reference 8)	Y	Acceptable
Essential Service Water System	VI.3.2 (page 93)	9.8.3	Y (References 8, 17, 18, 19, 20)	Y	Acceptable
Non-Essential Service Water	VI.3.3 (page 93)	9.8.3	Y (References 8, 21, 22)	Y	Acceptable
Turbine Auxiliary Cooling Water System	VI.3.4 (page 94)	10.7	Y (Reference 8)	Y	Acceptable
Emergency Diesel Generator Aftercooler, Lube Oil, and Jacket Cooling Water System	VI.3.5 (page 94)	8.4	Y (Reference 8)	Y	Acceptable
Circulating Water System	VI.3.6 (page 94)	10.6	Y (Reference 8)	Y	Acceptable
Spent Fuel Pool Cooling System	VI.3.7 (page 94)	9.4	Y (References 8, 23, 24, 25)	Y	Acceptable
Heating, Ventilation, and Air Conditioning Systems					
Auxiliary Building Ventilation Systems[Note 4]	VI.4 (Page 95)	9.9	Y (References 8, 9, 29, 31)	Y	Acceptable
Engineered Safety Features Ventilation System	VI.4 (Page 95)	9.9	Y (References 8, 9, 29, 31)	Y	Acceptable
Containment Ventilation System	VI.4 (Page 95)	5.5	Y (References 2, 8, 9, 28, 30, 32)	Y	Acceptable

Table 3.8.2 Plant Systems - Summary of NRC Staff Review					
Topic	**Unit 2 MUR Application Section**	**UFSAR Section**	**Bounded by NRC-approved analysis**	**Similar to Unit 1 MUR**	**NRC Staff Conclusion**
Auxiliary Feedwater Pump Room Coolers	VI.4 (Page 95)	9.8.3, 9.9.3, 14.4.9	Y (Reference 18)	Y	Acceptable
Control Room Ventilation System[Note 5]	VI.3.2 VII.5(iii) VII.6.1 VI.6.10	9.10	Y (References 33, 34)	Y	Acceptable

Table 3.8.2 References:

1. D. C. Cook Unit 2 License Amendment No. 135, dated September 18, 1990 [Allowed Unit 2 SG stop valve closure within 8 seconds]

2. D. C. Cook Units 1 and 2 License Amendment Nos. 234 and 217, dated December 13, 1999 [Approved containment sump modification, as evaluated in Westinghouse Licensing Topical Report WCAP-15302, “Donald C. Cook Nuclear Plant Units 1 and 2, Modifications to the Containment Systems, Westinghouse Safety Evaluation (SECL 99-076, Revision 3),” dated September 1999]

3. D. C. Cook Units 1 and 2 License Amendment Nos. 148 and 134, dated August 27, 1990 [Approved the transition to Westinghouse 17x17 VANTAGE 5 fuel and the use of Westinghouse Licensing Topical Report WCAP-11397-P-A, “Revised Thermal Design Procedure,” dated April 1989]

4. D. C. Cook Units 1 and 2 License Amendment Nos. 214 and 199, dated March 13, 1997 [Approved an increase in SG plugging limit]

5. D. C. Cook Unit 1 License Amendment No. 252, dated March 29, 2001 [Approved changes to TSs for spray additive tank (the analyses covered both units but only resulted in changes to Unit 1)]

6. Letter from R. L. Baer, NRC, to J. Tillinghast, I&M, “Order for Modification of License (Donald C, Cook Nuclear Plant Unit 2),” dated June 6, 1978 [Modifies TS limit for total nuclear peaking factor (F_Q)]

7. Supplement to Safety Evaluation Report, “Supplement No. 3 to Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316,” dated December 12, 1974

8. Safety Evaluation Report, “Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant - Units 1 and 2, Docket Nos. 50-315 and 50-316,” dated September 10, 1973

9. Letter from NRC to Indiana and Michigan Electric Company, “Supplement 7 to Safety Evaluation Report,” dated December 23, 1977

10. D. C. Cook Units 1 and 2 License Amendment Nos. 31 and 12, dated July 31, 1979 [Added license conditions for the Fire Protection Program]

11. Letter from S. A. Varga, NRC, to J. Dolan, I&M, “Safety Evaluation on Alternative Shutdown Capability,” dated November 22, 1983 [Complies with Sections III.G and III.L of Appendix R]

12. Letter from S. A. Varga, NRC, to J. Dolan, I&M, “Acceptance of Technical Exemptions from 10 CFR [Part] 50, Appendix R,” dated August 27, 1985

13. Letter from B. J. Youngblood, NRC, to J. Dolan, I&M, “Safety Evaluation Report Regarding Alternative Shutdown Procedures in the Event of Fire at D. C. Cook Units 1 and 2,” dated January 28, 1987

14. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, “Unrated Fire Hatches in Fire Area Boundaries (TAC Nos. 61690/61691),” dated June 17, 1988

15. Letter from R. S. Boyd, NRC, to J. Tillinghast, I&M, “Issuance of Facility Operating License No. DPR-74 (Donald C. Cook Nuclear Plant, Unit No. 2),” dated December 23, 1977

16. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, ”Revision to Technical Specification Bases Reflecting Change to Fire Suppression Backup Water Source (TAC Nos. M90177 & M90178),” dated December 14, 1994

17. D. C. Cook Units 1 and 2 License Amendment Nos. 164 and 149, dated April 22, 1992 [Approved changes to make TSs more consistent with ASME Code Requirements]

18. D. C. Cook Units 1 and 2 License Amendment Nos. 244 and 225,” dated April 25, 2000 [Approved modification to turbine-driven auxiliary feedwater pump room cooler]

19. D. C. Cook Units 1 and 2 License Amendment Nos. 253 and 235, dated August 3, 2001 [Added requirement for essential service water cross-tie to opposite unit]

20. D. C. Cook Units 1 and 2 License Amendment Nos. 270 and 251, dated September 9, 2002 [Approved changes to allow one-time extended allowed outage time for essential service water pump replacement]

21. D. C. Cook Units 1 and 2 License Amendment Nos. 59 and 42, dated September 9, 1982 [Approved TS changes to reflect replacement of containment isolation valves]

22. D. C. Cook Units 1 and 2 License Amendment Nos. 95 and 81, dated April 23, 1986 [Approved changes to containment isolation valve testing requirements]

23. D. C. Cook Units 1 and 2 License Amendment Nos. 32 and 13, dated October 16, 1979 [Approved increased storage capacity in spent fuel pool]

24. D. C. Cook Units 1 and 2 License Amendment Nos. 169 and 152, dated January 14, 1993 [Approved changes for spent fuel pool re-racking]

25. D. C. Cook Units 1 and 2 License Amendment Nos. 260 and 243, dated November 30, 2001 [Approved revision to “decay time” to allow start of core offload at 100 hours]

26. D. C. Cook Units 1 and 2 License Amendment Nos. 182 and 167, dated September 9, 1994 [Approved an increase in MSSV setpoint tolerances]

27. D. C. Cook Units 1 and 2 License Amendment Nos. 214 and 199, dated March 13, 1997 [Approved an increase in SG plugging limit]

28. D. C. Cook Units 1 and 2 License Amendment Nos. 66 and 47, dated December 8, 1982 [Allowed containment purging during operation and containment purge and vent modifications specified by the Three Mile Island Action Plan]

29. D. C. Cook Units 1 and 2 License Amendment Nos. 124 and 111, dated May 19, 1989 [Engineered Safety Features and Storage Pool Ventilation System]

30. D. C. Cook Units 1 and 2 License Amendment Nos. 195 and 181, dated June 23, 1995 [Containment Purge]

31. D. C. Cook Units 1 and 2 License Amendment Nos. 257 and 240, dated October 24, 2001 [Approved the use of American Society for Testing and Materials Standard D3808-1989 for charcoal testing in accordance with Generic Letter 99-02]

32. D. C. Cook Units 1 and 2 License Amendment Nos. 259 and 242, dated November 21, 2001 [Approved TS requirements to immediately (1) suspend operations involving core alterations, positive reactivity changes, and movement of irradiated fuel assemblies, (2) initiate actions to restore the required buses and return equipment to operable status, and (3) declare the associated required RHR loop(s) inoperable in Modes 5 and 6 with less than the specified minimum complement of AC or DC busses and equipment inoperable]

33. D. C. Cook Units 1 and 2 License Amendment Nos. 258 and 241, dated November 13, 2001 [Partial alternative source term]

34. D. C. Cook Units 1 and 2 License Amendment Nos. 271 and 252, dated November 14, 2002 [Alternative source term for control room habitability]

Table 3.8.2 Notes:

1. To support the proposed D. C. Cook Unit 2 MUR power uprate, the licensee performed an assessment to demonstrate that the post-LOCA hydrogen generation at the uprated power level remain within acceptance criteria (See Section II.2.2 of the licensee's November 15, 2002, application and Section 3.8.2.1 below for the NRC staff's evaluation). For Unit 1, the existing post-LOCA hydrogen analysis was based upon a core power of 3411 MWt, which bounds the proposed D. C. Cook Unit 1 MUR power uprate.

2. The licensee is in the process of conducting steam dump/margin-to-trip final analyses for D. C. Cook Unit 2. (See Attachment 5, "Regulatory Commitments," of November 15, 2002, application.) The licensee will make changes/adjustments as required to ensure that valves have sufficient capacity prior to implementing the proposed 1.66-percent power uprate.

3. The licensee re-performed the RHR cooldown analyses to support the proposed D. C. Cook Unit 2 MUR power uprate. The revised analyses, which considers a change to the plant's RTP only, demonstrates that the licensee will still be able to reach Mode 5 conditions within 36 hours on a single train of RHR, and the time to cool down to <140 °F with two trains of RHR available has increased from less than 20 hours to less than 23 hours. (See Section 3.2.2.1 of this SE)

4. The auxiliary building ventilation systems at D. C. Cook include the engineered safety features ventilation system, fuel handling area ventilation system, general ventilation systems, and general supply system.

5. The control room ventilation system was assessed as part of the on-site radiological dose consequences assessment, the heat load assessment for the essential service water system, and the environmental qualification of equipment analyses.

3.8.2.1 Post LOCA Containment Hydrogen Generation

The licensee determined that the analysis of record for post-LOCA hydrogen generation was performed for core thermal power of 3411 MWt. This analysis was performed to bound both units and is presented in Section 14.3.6 of the D. C. Cook Unit 1 UFSAR. The proposed uprated power level for D. C. Cook Unit 2 is 3468 MWt. Therefore, the existing analyses of record does not bound proposed operation of the plant at the uprated power level.

To support the power uprate application, the licensee performed assessments for the post-LOCA hydrogen generation analysis. The licensee's assessment covered operation up to 3588 MWt. Since the calculated hydrogen produced by radiolysis in the core and sump is a function of the ionizing radiation flux, the licensee assumed that the hydrogen produced by radiolysis is directly proportional to the core power level. The licensee assumed a one-to-one correlation and increased the hydrogen produced by radiolysis in the core and sump by 5 percent, which corresponds to an increase of 5-percent power. The licensee determined that the hydrogen generation from sources other than radiolysis would not be affected by the proposed power uprate.

The licensee's assessment was based upon the application of a conservative and bounding power increase of 5 percent compared to the proposed power increase of 1.66 percent. The licensee concluded that hydrogen production from all sources increases by only 1 percent during the first 24 hours, and by 2 percent at the end of 100 hours as a result of the increase in power level. Further, the calculations show that if recombiners are started at or before the time at which the containment hydrogen concentration reaches 3.5-percent volume, the resulting hydrogen concentrations remain below the lower flammability limit of 4.0 percent.

Similar assessments were also performed for containment subcompartment hydrogen concentrations. The licensee again increased the hydrogen produced by radiolysis in the core and sump by 5 percent, corresponding to an increase of 5-percent power. The licensee did not increase hydrogen generation sources other than radiolysis because it determined that hydrogen generation from such sources would not be affected by the proposed power uprate. The licensee's assessments for the hydrogen concentrations in containment subcompartments concluded that an increase in power up to 3588 MWt would result in an increase of 0.1 percent in the short-term peak subcompartment hydrogen concentration following a LBLOCA and an increase of 1.6 percent in the long-term (i.e., final analysis time of ~10 hours following a LBLOCA and ~14 hours following a small-break LOCA) peak subcompartment hydrogen concentrations. Further, the calculated values for the short-term and long-term hydrogen concentrations remain below the flammability limit of 4.0 percent.

The NRC staff reviewed the licensee's application related to post-LOCA hydrogen generation in containment, including the licensee's assumptions related to (1) the sources of hydrogen, (2) the one-to-one correlation between reactor power and hydrogen generation, and (3) the 5-percent increase in power. The NRC staff determined that (1) the licensee's assumptions are reasonable, (2) the assessments were performed in an acceptable manner to bound the proposed operation of the plant at the uprated power level, and (3) the resulting hydrogen concentrations remain below the lower flammability limit of 4.0 percent. The NRC staff concludes that the proposed MUR power uprate is acceptable with respect to post-LOCA containment hydrogen generation.

3.8.3 Summary

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, and (7) ESF heating, ventilation, and air conditioning systems. The NRC staff has determined that the results of licensee's analyses related to these areas would continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to plant systems.

4.0 LICENSE AND TECHNICAL SPECIFICATION CHANGES

(1) Change to Facility Operating License No. DPR-74

The licensee proposes to revise paragraph 2.C.(1) in Facility Operating License DPR-74 to authorize operation at a steady-state reactor core power level not in excess of 3468 MWt (100-percent power).

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

(2) Change to TS 1.3

The licensee proposes to revise the definition of "RATED THERMAL POWER" in TS 1.3 to reflect the increase from 3411 MWt to 3468 MWt.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

(3) Change to TS 3.5.2

The licensee proposes to revise the maximum allowed power level in TS 3.5.2, Action b, from 3250 MWt to 3304 MWt, to increase the maximum allowable core power level with a safety injection cross-tie valve closed.

The existing analysis of record supporting the maximum allowable core power level with a safety injection cross-tie valve closed (TS 3.5.2, Action b) was performed for a nominal power level of 3250 MWt and a power level uncertainty of 2 percent. The licensee's November 15, 2002, application justifies a reduction in the power level uncertainty from 2 percent to 0.34 percent. As a result, the licensee proposes to increase the maximum allowable core power level in TS 3.5.2, Action b, by 1.66 percent (i.e., the difference between the original assumption of 2-percent uncertainty and the proposed value of 0.34-percent uncertainty). The licensee's proposed change would result in an increase of the maximum allowable core power level in TS 3.5.2, Action b, from 3250 MWt to 3304 MWt. Based on (1) the NRC staff's acceptance of the new value of 0.34 percent for total power uncertainty (see Section 3.1 above), (2) the fact that the existing analysis of record accounted for a 2-percent uncertainty, and (3) the fact that this change merely recovers the difference between the 2 percent assumed in the analysis of record and the 0.34 percent accepted by the NRC staff in Section 3.1 of this SE, the NRC staff finds the proposed change acceptable.

(4) Changes to TS Table 3.7-1

The licensee proposes to revise TS Table 3.7-1 to reflect the maximum allowed power for operation with inoperable MSSVs. For D. C. Cook Unit 2, with one, two, or three MSSVs inoperable, the licensee proposes to change the maximum allowable power levels from 61.6 percent, 43.9 percent, and 26.2 percent to 60.4 percent, 43.0 percent, and 25.7 percent, respectively.

In TS Table 3.7-1, the licensee proposed the insertion of new values for the setpoints with inoperable MSSVs to be consistent with the proposed MUR power uprate. To calculate these values for the proposed uprated power level, the licensee used the conservative heat balance calculation described in TS Bases Section 3/4.7.1.1. The use of this conservative heat balance calculation to determine the new power range neutron flux high setpoints assures that the power level is sufficiently limited to accommodate the lower main steam system relief capacity with the corresponding number of MSSVs out of service while still maintaining protection against the limiting design-basis transient for MSSV capacity (i.e., the turbine trip transient). Therefore, the NRC staff finds the new values for maximum allowable power levels acceptable.

5.0 REGULATORY COMMITMENTS

To support the proposed D. C. Cook Unit 2 MUR power uprate, the licensee made the following commitments (as stated):

> I&M is installing an LEFM CheckPlus system at CNP [D. C. Cook Nuclear Plant] Unit 2 in anticipation of approval of this proposed amendment. Installation of this system will begin prior to the Unit 2 Cycle 14 refueling outage and will be completed after receipt of the requested license amendment. The design change for the installation will include instrumentation rescaling, UFSAR revision, maintenance and operational procedure impacts, training, monitoring iso-phase bus duct temperature, and implementation of the LEFM CheckPlus system out-of-service administrative technical requirements. The UFSAR revision for the Unit 2 MUR power uprate will be reflected in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).
>
> Prior to implementing this uprate, a reload safety evaluation will be performed to ensure that the core design bounds the uprated condition.
>
> Perform an analysis of the steam dump valve flow capacity at the uprated power level and implement changes/adjustments as required to ensure the valves have sufficient capacity prior to implementing the 1.66 percent power uprate.

The NRC staff considered the above commitments as part of its evaluation in Section 3.0 above and finds the commitments appropriate for the proposed MUR power uprate. The NRC staff has conditioned the implementation of the proposed MUR power uprate on completion of the above commitments.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 2805). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: List of Acronyms

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LIST OF ACRONYMS

AOP	Abnormal Operating Procedures
ASME	American Society of Mechanical Engineers
BOP	balance-of-plant
CFR	*Code of Federal Regulations*
DBA	design-basis accident
DNB	departure from nucleate boiling
ECCS	emergency core cooling system
EOL	end of license
EOP	emergency operating procedure
ESF	engineered safety feature
FAC	flow-accelerated corrosion
LEFM	leading edge flowmeter
LOCA	loss-of-coolant accident
MOV	motor-operated valve
MSSV	main steam safety valves
MUR	measurement uncertainty recapture
MWt	megawatts thermal
NRC	Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letters
NSSS	Nuclear Steam Supply System
PPC	plant process computer
P-T	pressure-temperature
PTS	pressurized thermal shock
RCCA	rod cluster control assembly
RCS	reactor coolant system
RHR	residual heat removal

RIS	Regulatory Issue Summary
RTP	rated thermal power
RV	reactor vessel
SE	safety evaluation
SG	steam generator
SGTR	steam generator tube rupture
UFSAR	Updated Final Safety Analysis Report
USE	upper-shelf energy