December 20, 2002

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT 273 REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MB5498)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 273 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1. The amendment consists of changes to the License and Technical Specifications in response to your application dated June 28, 2002, as supplemented October 15 (two separate letters), October 17, November 15, and December 6, 2002.

The amendment increases the licensed reactor core power level by 1.66 percent from 3250 megawatts thermal (MWt) to 3304 MWt. The power level increase is considered a measurement uncertainty recapture power uprate.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 273 to DPR-58
2. Safety Evaluation

cc w/encls: See next page

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KMcConnell	OGC	EMarinos	SPeters	JMedoff	
LRaghavan	ACRS	CHolden	PYChen	CKhan	

*Previously Concurred

OFFICE	PDIII-1/PM	PDIII-1/LA	SPLB/SC*	IEHB/SC*	OGC*	PDIII-1/SC	PDIII/D(A)	DLPM/D
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DATE	12/20/02	12/20/02	12/20/02	12/17/02	12/20/02	12/20/02	12/20/02	12/20/02

*The following technical branches provided safety evaluation input by memos dated**

OFFICE	EMCB	EMEB	EEIB	EEIB	EMCB	SRXB	SPSB
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DATE*	11/20/02	11/14/02	12/09/02	09/06/02	11/12/02	11/15/02	08/26/02

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 273
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated June 28, 2002, as supplemented October 15 (two separate letters), October 17, November 15, and December 6, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 273, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended to revise paragraph 2.C.(1) to reflect the increase in the reactor core power level. Paragraph 2.C.(1) is hereby amended to read as follows:

> The licensee is authorized to operate the Donald C. Cook Nuclear Plant, Unit No. 1, at steady state reactor core power levels not to exceed 3304 megawatts (thermal).

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance. Prior to implementation of the amendment, the licensee shall complete the following:

   A. Complete a formal engineering/reload safety evaluation of the effects of the power uprate on the Updated Final Safety Analysis Report (UFSAR) Sections 14.1.2, 14.1.8, and 14.1.10, as described in the licensee's June 28, 2002, application, and evaluated in the associated safety evaluation by the Office of Nuclear Reactor Regulation, dated December 20, 2002. The evaluation shall be reflected in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

   B. Complete an analysis of the steam dump valves 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 analysis and changes/adjustments to the steam dump valves shall be reflected in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

   C. Incorporate the secondary side pressure limitation of 679 psia into the UFSAR. This change shall be reflected in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

D. Submit a license amendment application to the NRC proposing new Unit 1 reactor coolant system pressure-temperature curves that reflect the limiting reactor vessel beltline material in accordance with 10 CFR Part 50, Appendix G.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/ by T. Marsh

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance: December 20, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 273

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of Facility Operating License No. DPR-58 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1-1	1-1
2-7	2-7
2-9	2-9
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 7-2	3/4 7-2
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 7-1	B 3/4 7-1

Donald C. Cook Nuclear Plant, Unit 1

Safety Evaluation for Amendment No. 273

Measurement Uncertainty Recapture Power Uprate

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RELATED TO AMENDMENT NO. 273 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By application dated June 28, 2002, as supplemented October 15 (two separate letters), October 17, November 15, and December 6, 2002, the Indiana Michigan Power Company (the licensee) requested an amendment to the Operating License and the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant (D. C. Cook), Unit 1. The 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 July 23, 2002 (67 FR 48219).

The proposed changes would increase the licensed reactor core power level by 1.66 percent from 3250 megawatts thermal (MWt) to 3304 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-58 to authorize operation at a steady-state reactor core power level not in excess of 3304 MWt (100 percent power).

2. The definition of RATED THERMAL POWER (RTP) in TS 1.3 to reflect the increase from 3250 MWt to 3304 MWt.

3. The notations for TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," to limit indicated T_{avg} at RTP to less than or equal to 574 °F (T' for Overtemperature Delta T (DT)) and 562.1 °F (T'' for Overpressure DP).

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 65.1 percent RTP to 63.8 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 45.5 percent (two inoperable MSSVs) and 27.4 percent (three inoperable MSSVs). The TS Bases for Limiting Condition for Operation (LCO) 3.7.1 would also be revised to reflect these changes.

5. The two notes on TS Figure 3.4-2, “Reactor Coolant System Pressure-Temperature Limits Versus 60°F/Hr Rate Criticality Limit and Hydrostatic Test Limit,” and the two notes on TS Figure 3.4-3, “Reactor Coolant System Pressure-Temperature Limits Versus Cooldown Rates,” to change the current limit of applicability from 32 effective full power years (EFPYs) to 18.6 EFPYs. The Bases for TS 3/4.4.9, “Pressure/Temperature Limits,” would also be revised to reflect these changes.

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 the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. The licensee has proposed to use a value of 1.004. 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 (Reference 1) and its supplement, Topical Report ER-157P (Reference 2). (The currently installed venturi flow meter will remain in place.) The licensee proposed to increase the power output of the plant by the margin created by the difference between the 1.02 multiplier used in the existing analyses of record and the 1.004 multiplier proposed as a result of the installation of the more accurate flowmeter. Since the analyses of record for LOCA and ECCS assumed a power level of 1.02 times the licensed power level, a 1.66-percent increase in power could be achieved without necessitating a reanalysis of these events. Other design-basis analyses are evaluated to ensure appropriate accounting of power level uncertainties.

3.0 EVALUATION

3.1 Instrumentation and Controls

3.1.1 Regulatory Evaluation

The NRC staff reviewed the area of instrumentation and controls to confirm that the licensee has applied Caldon Topical Report ER-80P and its supplement, Topical Report ER-157P. These reports evaluated the feedwater flow measurement device proposed to be used for improving the accuracy of the power measurement, in a manner consistent with the NRC staff's approval of the topical reports. The NRC staff also reviewed the power uncertainty calculations to ensure that they meet the requirement of Appendix K to 10 CFR Part 50 and demonstrate that the proposed uncertainty value of .34 percent correctly accounts for the uncertainties due to power level instrumentation errors.

3.1.2 Technical Evaluation

The generic bases for the proposed power uprate are provided in Caldon Topical Report ER-80P (Reference 1) and its supplement, 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. 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 NRC staff has reviewed the regulatory and technical analyses provided by the licensee in support of the proposed power uprate. The licensee anticipates significant improvement in the accuracy with which reactor core power level will be measured as a result of the installation of a new flowmeter in the common feedwater header upstream of the branches to individual feedwater lines. The new instrument is an ultrasonic, multi-path, transit-time LEFM manufactured by Caldon under the brand name "LEFM CheckPlus." Core power is inferred from feedwater flow (among other things), and improvement in feedwater flow measurement accuracy directly improves the power measurement. The licensee has submitted an uncertainty evaluation which evaluates 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, and will be reconfirmed during the commissioning process following installation. Therefore, the original 2-percent margin would be reduced to 0.34 percent, allowing for a power uprate of 1.66 percent (2 percent - 0.34 percent).

In the NRC safety evaluation reports approving the Caldon topical reports, the NRC requested that licensees address the following issues when applying for the approval of an MUR power uprate:

1. Maintenance and calibration procedures that will be implemented with the incorporation of the LEFM.

2. For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

3. The methodology used to calculate the uncertainty of the LEFM.

4. Where the ultrasonic meter (including the LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), licensees should provide additional justification for use.

The licensee provided the information concerning each of the above issues in its application and supplements. The NRC staff's evaluation of each of the licensee's responses is below.

3.1.2.1 Maintenance and Calibration Procedures

The licensee stated that implementation of the power uprate will include development of necessary procedures and documents required for operation, maintenance, calibration, testing, and training with the new system. Plant maintenance and calibration procedures will be revised to incorporate the instrument supplier's maintenance and calibration requirements prior to declaring the system operational and raising the core power above the current limit.

The LEFM provides data to the plant process computer system (PPC) by means of a data link. The PPC computes the core power as inferred from the LEFM data and the venturi-based data. A correction factor is computed for the venturi-based core power computation to bring it into conformance with the more accurate LEFM result. In the event of LEFM-related failure, this correction factor could be used along with the venturi-based data, subject to certain limitations, to produce a sufficiently accurate estimate of reactor power to permit continued operation above the pre-uprate RTP provided core power is maintained within 10 percent of the power level that corresponds to the most recently computed correction factor. If there is a change in core power in excess of 10 percent[1], or if the LEFM-based reactor power estimate is not restored, then core power will be reduced so that it is within the pre-uprate RTP within 48 hours of the failure.

If the plant experiences a power change greater than 10 percent during the 48-hour period in which the LEFM CheckPlus system is not available, then the permitted maximum power level must be reduced to 3250 MWt. This is to ensure that the calorimetric power level at which the plant operated accounts for potential differences in the way the venturi and LEFM calorimetric respond to a change in power. A power change greater than 10 percent may result in a change in the relationship between the venturi and LEFM calorimetric such that the correction factor calculated following the loss of LEFM may no longer be valid. The RTP will continue to be monitored by calibrated alternate instrumentation. Operators in the control room, monitoring core power via the calibrated alternate instruments, will assure that the plant continues to operate at or below the licensed RTP.

In addition to the process inputs provided by the LEFM CheckPlus system, the PPC program uses the following process inputs to calculate thermal power:

- steam pressure
- blowdown flow
- charging flow
- charging temperature

[1]The 10-percent power fluctuation limitation does not imply allowance to exceed 100 RTP.

- charging pressure
- letdown flow
- letdown temperature
- letdown pressure
- pressurizer pressure
- reactor coolant system (RCS) loop 4 cold leg temperature
- volume control tank temperature

Blowdown flow measurement is performed by a Caldon ultrasonic measurement system. The calibration of this ultrasonic measurement system is maintained using self-checking and self-adjusting methods. The value and status of the blowdown flow measurement are provided to the PPC. If the status of the blowdown flow measurement or the failure of the blowdown flow system indicates that the status is incorrect, this is reflected in the PPC calorimetric program and results in the status of the LEFM calorimetric values also indicating a "bad" status. Control of the ultrasonic measurement system is maintained by the licensee's change control process. Hardware control of the ultrasonic measurement system is provided by the licensee's design change control process, which conforms to 10 CFR Part 50, Appendix B, and control of the system software is provided by the licensee's software control process.

The remaining process inputs are obtained from analog instrumentation channels that are maintained and calibrated in accordance with required periodic calibration procedures. Configurations of the hardware associated with these process inputs is maintained in accordance with the licensee's change control process.

Instruments that affect the power calorimetric, including the LEFM inputs, are monitored by the licensee's System Engineering personnel in accordance with the provisions of the licensee's Corrective Action Program. Equipment problems for plant systems, including the LEFM CheckPlus equipment, fall under the site work control processes. Conditions that are adverse to quality are documented under the Corrective Action Program. Corrective action procedures, which ensure compliance with the requirements of 10 CFR Part 50, Appendix B, include instructions for notification of deficiencies and error reporting.

Calibration and maintenance are performed by the licensee's Instrumentation and Controls - Maintenance Department personnel using site procedures. Site procedures are developed using the vendor technical manuals for the applicable equipment. All work is performed in accordance with site work control procedures. Routine preventive maintenance procedures include physical inspections, power supply checks, backup battery replacements, and internal oscillator frequency verification. Corrective actions involving maintenance will be performed by the licensee's personnel qualified in accordance with the licensee's Instrumentation and Controls Training Program.

3.1.2.2 <u>Operational and Maintenance History of the Leading Edge Flowmeter Installation</u>

The licensee is currently in the process of installing the LEFM system in D. C. Cook Unit 1. Prior to putting the system into service, the licensee has committed to confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions in Caldon Topical Report ER-80P. The post modification testing procedures for the LEFM will contain a provision that the licensee confirm the above noted conditions. This will be accomplished in accordance with currently-existing licensee procedures.

Based on the licensee's commitment, the NRC staff finds that the licensee has adequately addressed the issue of operational and maintenance history of the installed LEFM system.

3.1.2.3 Methodology used to calculate the uncertainty of the LEFM system

The licensee computes calorimetric accuracy on the basis of thermal power sensitivity to variations in each of the component measurements, consistent with the Improved Thermal Design Procedure (ITDP) presented in Westinghouse Topical Report WCAP-8567 (Reference 3), and the overall statistical approach to combining uncertainties is in compliance with ANSI/ISA 67.04.01-2000 (Reference 4).

The licensee's uncertainty calculation evaluates the uncertainty in computed reactor core power that results from uncertainties in each of the parameters used in the computation. With the reactor operating at a reference condition near full power, a "snapshot" of the simultaneous values of core power and all associated parameters is recorded. A reference reactor core power computation is executed to document the values of the parameters and the corresponding core power. The computation is repeated for various values of each parameter. In these computations, the parameter in question is adjusted over the full range of its uncertainty band while all other parameters are held to their reference values. The uncertainty band for each parameter is either derived within the calculation or obtained from some cited reference. The maximum deviation of the computed core power from the reference value is noted for each parameter, regardless of which specific value produced the deviation or whether the deviation is positive or negative. This process establishes the maximum influence of the uncertainty in each parameter over the computed core power. The individual parameter influences are then combined to determine the overall core power uncertainty.

The foregoing calculation is performed using actual plant data obtained from a "snapshot" of operation near 100-percent power. The actual power level in the "snapshot" was slightly less than 100 percent, and the operating point of interest is the uprated power limit. To account for this variation in power level, the calculation was repeated with the feedwater flow value increased to yield a power level of 102 percent, thereby bounding the proposed 1.66-percent power uprate. All other parameters were held at the same nominal values and uncertainties as before. The change in uncertainty was on the order of 0.0003 percent, and so the "snapshot" based calculation is deemed applicable for the proposed power uprate.

The calculation is also repeated to evaluate the impact of a proposed future increase in a blowdown flow rate. Again, all other parameters and variances are held to the "snapshot" related values. A small increase in core power uncertainty is noted, but the resulting uncertainty remains well within the 0.34-percent limit upon which the proposed uprate is based.

The licensee cites the ITDP of Westinghouse Topical Report WCAP-8567 as one basis for the methodology employed to compute reactor core power uncertainty. ITDP was accepted by the NRC for use in departure from nucleate boiling ratio (DNBR) related calculations. Although the present calculation does not itself address DNBR directly and does not fully apply all provisions of Westinghouse Topical Report WCAP-8567, the NRC staff agrees that the basic concept employed regarding the "sensitivity" of computed core power to variations in measured plant process parameters is acceptable.

In computing the electromagnetic interference/radio frequency interference contribution to the uncertainty in feedwater pressure measurement, the licensee has incorporated the pressure transmitter manufacturer's assumed ambient field strength of 30 volts per meter over the frequency range of 20 to 1000 MHz. This is in excess of the 10v/m reference susceptibility level of Regulatory Guide (RG) 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," and much larger than the reasonably expected actual field strength, and therefore, the licensee's approach is acceptable. The uncertainty derivations for each of the process parameters included in the reactor power computation are included in the uncertainty evaluation directly or by reference.

The uncertainty evaluation concludes that the uncertainty in the reactor core thermal power evaluation, for both the present and the anticipated increased blowdown flow rate, is less than the ±0.34 percent of RTP required to support the proposed 1.66-percent power uprate.

Based on the above, the NRC staff finds that the licensee's uncertainty calculation used a methodology to calculate the uncertainty of the LEFM based on setpoint methodology previously reviewed and approved by the NRC staff, and is therefore acceptable.

3.1.2.4 Ultrasonic Meter Installation

The calibration factor for the instrument to be installed was established by means of testing of a full-scale model utilizing the installed geometry. Final acceptance will follow commissioning, which will confirm that actual performance (as-installed) will meet the established uncertainty bounds. Final commissioning will be in accordance with the manufacturer's recommendations as expressed in Appendix F of Caldon Topical Report ER-80P.

In as much as the NRC staff has already approved Topical Report TR ER-80P, the NRC staff finds that the licensee's responses sufficiently resolve the plant-specific concerns regarding maintenance and calibration of the LEFM system and other instrumentation affecting heat balance, hydraulic configuration of the installed LEFM, processes and contingencies for an inoperable LEFM, and methodology for calculating the LEFM and plant core power measurement uncertainties.

3.1.3 Summary

The NRC staff has reviewed the licensee's application regarding the manner in which the licensee proposes to apply Caldon Topical Report ER-80P and its supplement, Topical Report ER-157P. As set forth above, the NRC staff finds that the licensee's proposed application of the topical reports for D. C. Cook Unit 1 is consistent with the NRC staff's approval of these topical reports. In addition, the NRC staff has reviewed the licensee's power level uncertainty calculations and finds that the licensee adequately accounted for the uncertainties due to power level instrumentation error. Based on the above, the NRC staff concludes that the proposed power uprate is acceptable with respect to the instrumentation and controls area.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

The NRC staff's review of the area of reactor systems for the proposed MUR power uprate covers the following:

- accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level
- LOCA and LOCA-related events (including steam generator tube rupture (SGTR))
- non-LOCA analyses
- design transients
- accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level
- excessive heat removal due to feedwater system malfunctions loss of external electrical load - departure from nucleate boiling (DNB) case
- uncontrolled rod control cluster assembly (RCCA) bank withdrawal at power
- mechanical/structural/material component integrity and design steam generators (SGs)
- fuel evaluation
- system design
- nuclear steam supply system (NSSS) interface systems
- NSSS control systems
- changes to TSs, protection system settings, and emergency system settings

3.2.2 Technical Evaluation

3.2.2.1 Design Operating Parameters and Initial Conditions

The NSSS design parameters provide the RCS and secondary system conditions for use in the NSSS analyses and evaluations. The licensee presented four different plant cases at a core power level of 3315 MWt to bound the proposed uprate conditions. Two cases included varying the SG tube plugging levels from 0 percent to 30 percent while maintaining a T_{avg} of 553.7 °F. The other two cases similarly varied the tube plugging level, but maintained a T_{avg} of 575.4 °F. The NRC staff evaluated these parameters for these cases and found that they adequately represent the plant behavior at the specified conditions. Therefore, the NRC staff finds the reference NSSS design parameters to be acceptable.

3.2.2.2 Core Thermal Limits and OverTemperature and Overpower Delta-T Setpoints

The core thermal limits and the overtemperature and overpower DT setpoints (OTDT and OPDT) are essential inputs to the non-LOCA safety analyses. The licensee determined that it did not need to revise the OTDT and OPDT setpoint coefficients to increase the core power. however, the power uprate requires changes to the reference average temperatures. The proposed changes would restrict the OTDT average temperature (T') to 574 °F to ensure that the current F(DI) penalty limits remain bounding. The licensee will also restrict the OPDT reference average temperature (T'') to 562.1 °F to preserve the current OTDT and OPDT setpoints.

The proposed MUR power uprate would decrease the allowable T_{avg}. Since the licensee's proposed changes will account for this decrease and will ensure that their setpoints and core limits remain adequate, the NRC staff finds the changes to the OTDT and OPDT average reference temperatures acceptable. Additionally, the core thermal limits are inputs to the OTDT and OPDT setpoint calculations. These limits change with the increase in power; however, the changes are relatively linear with respect to changes in reactor coolant temperature, pressure, and thermal power. The licensee evaluated the current OTDT and OPDT setpoints and found that they continue to protect the core thermal limits for the uprated conditions.

Since the current OTDT and OPDT setpoints will continue to protect the core thermal limits at the uprated conditions, the NRC staff finds them acceptable for the proposed power uprate.

3.2.2.3 Accidents and Transients Analyses of Record that are Bounding

The current D. C. Cook Unit 1 large-break LOCA and small-break LOCA analyses assume 102 percent (3315 MWt) of the licensed power (3250 MWt) for the plant. The licensee has proposed to continue using the current large-break LOCA and small-break LOCA analyses as the licensing-basis analyses for D. C. Cook Unit 1. The proposed power uprate is based on the licensee's use of the Caldon LEFM technology to reduce the power measurement uncertainty to less than 0.34 percent from the previously assumed 2.0 percent. The remaining 1.66-percent margin is available for increasing the licensed power without changing the LOCA analyses' initial power assumptions. This modification is permitted by 10 CFR Part 50, Appendix K, Section 1.A.

The licensee's proposed uprated LOCA analysis input initial power assumption is consistent with 10 CFR Part 50, Appendix K, Section 1.A. Therefore, the NRC staff finds that the LOCA analyses presently approved for D. C. Cook Unit 1 continue to apply and are suitable for inclusion in plant licensing documentation.

3.2.2.3.1 Post-LOCA Long-Term Core Cooling

The regulation at 10 CFR 50.46(b)(5), "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors - Long Term Cooling," establishes the long-term cooling requirement following a LOCA. To satisfy this requirement, an adequate water volume and boron concentration must be provided to ensure the reactor core remains subcritical assuming all control rods are out. The licensee's current licensing position is that the reactor remains shut down by borated emergency core cooling system (ECCS) water residing in the RCS sump following the LOCA. However, when considering the potential effect of sump dilution during the post-LOCA cold leg injection recirculation cooling mode, the licensee credits control rod insertion together with the available sources of boron to offset the dilution. Additionally, the licensee states that current long-term core cooling analysis for D. C. Cook Unit 1 uses a nominal core power level of 3481 MWt.

Based on the post-LOCA, long-term, core cooling analysis being analyzed at a power level of 3481 MWt, and the licensee's analysis adequately addressing the effects of sump dilution, the NRC staff concludes that the water volumes and boric acid (H_3BO_3) concentrations of the ECCS water supply will remain acceptable for the proposed power uprate to 3304 MWt.

3.2.2.3.2 Hot Leg Switchover

A long-term cooling concern is boric acid accumulation potentially preventing adequate core cooling, because of boron precipitation.

As part of the D. C. Cook Unit 1 restart effort, the licensee analyzed the boron precipitation and hot leg switch-over issues. From its analyses, the licensee determined that in order to provide a reasonable amount of time for performance of the hot leg switchover evolution (between 2.5 hours and 7.5 hours after rod insertion), it needed to credit control rod insertion. The NRC staff reviewed and approved the licensee's use of the methodology to credit the negative reactivity provided by control rod insertion into the reactor core following any design-basis LOCA, during realignment from a cold leg recirculation to a hot leg recirculation configuration (Reference 5) for D. C. Cook Units 1 and 2. Additionally, since the licensee uses a core power level of 3481 MWt, the licensee concluded that its analysis bounds the proposed power level of 3304 MWt. Given that the control rod insertion provides an adequate time for performance of the switchover, and given that the current analysis remains bounding, the licensee determined that the current Emergency Operating Procedures (EOPs) remain valid.

Since the current hot leg switchover analysis bounds the uprated conditions, and since the current EOPs remain valid, the NRC staff finds the current D. C. Cook Unit 1 hot leg switchover analysis acceptable for the proposed power uprate to 3304 MWt.

3.2.2.3.3 Steam Generator Tube Rupture - Thermal-Hydraulic Analysis

For an SGTR, the thermal-hydraulic analysis calculates the primary-to-secondary break flow and the steam released to the environment. For the offsite dose consequences, the NRC approved the Unit 1 SGTR analysis in D. C. Cook License Amendment No. 126 (Reference 6). For the onsite consequences, the NRC approved the analysis in D. C. Cook License Amendment Nos. 271 (Unit 1) and 252 (Unit 2) (Reference 7). These analyses considered core power levels up to 3588 MWt, therefore, they bound the proposed uprated power level of 3304 MWt.

In addition, to demonstrate that they continue to meet their design basis for SG overfill following an SGTR, the licensee performed a supplemental overfill analysis. Using the NRC-approved LOFTTR2 computer code, the licensee assumed a nominal NSSS power of 3262 MWt and showed that it had an adequate margin to overfill. When analyzing overfill for a 2-percent uprate, the licensee showed that it had a slightly higher margin to overfill. Even though the higher power level will cause a slightly higher break flow, these effects are offset by the lower initial water mass in the SG at the higher power level. In summary, the increase in power level will have a negligible effect on the margin to overfill analysis.

Since the power uprate has a negligible effect on the margin to overfill, and since the thermal-hydraulic conditions of the uprate are bounded by currently approved analyses, the NRC staff finds the SGTR accident analyses acceptable for the proposed power uprate to 3304 MWt.

3.2.2.4 Non-LOCA Analyses

The licensee performed all the analyses described in this section using NRC approved methodologies and as described below, appropriate input parameters. Accordingly, the NRC staff has determined that the results of these analyses may be compared to established acceptance criteria to determine whether the proposed action is acceptable. Individual accident analysis are discussed below.

3.2.2.4.1 Single Reactor Coolant Pump Locked-Rotor Accident

A reactor coolant pump (RCP) locked-rotor accident results from the instantaneous seizure of an RCP rotor. The flow through the affected reactor coolant loop rapidly reduces, and the reactor trips on a low reactor coolant flow signal. The sudden decrease in core coolant flow while the reactor is powered results in a decreased core heat transfer, which may cause fuel damage.

The licensee performed an evaluation of the accident for the uprated conditions to determine if the number of fuel rods that exceed the DNBR limit remain below those assumed in the D. C. Cook Unit 1 dose analysis. The licensee's evaluation showed that the existing statepoints for the accident remain valid, except for the increase in the nominal core heat flux. Even with the increased nominal heat flux, the licensee concluded that the fuel rods do not experience DNB. Since the rods will not experience DNB at the uprated conditions, the accident continues to meet the DNB acceptance criteria.

The licensee did not reanalyze the accident to ensure that it would continue to meet the maximum RCS pressure, clad temperature, and zirconium-water criteria. The licensee did not perform this analysis because the current model uses a 2 percent power measurement uncertainty, which bounds the power uprate of 1.66 percent.

Based on the licensee's analysis, the NRC staff determined that D. C. Cook Unit 1 will continue to meet the DNB acceptance criteria, and will continue to meet the RCS pressure, clad temperature, and zirconium-water criteria at the proposed uprated conditions. Since the plant will continue to meet the Standard Review Plan (SRP) acceptance criteria, the NRC staff finds these analyses acceptable for the proposed power uprate to 3304 MWt.

3.2.2.4.2 Loss of External Electrical Load - Overpressure Analysis

A loss of external electrical load event occurs when an electrical disturbance causes the loss of a significant portion of the generator load. Upon a full loss of load, the reactor protection system automatically initiates a reactor trip. For the overpressure analysis, the licensee evaluated this transient without pressurizer pressure control, assuming one case with minimum and one case with maximum reactivity feedback.

With the power level assumption of 102 percent of 3250 MWt (3315 MWt), the results of the analysis indicate that the SRP acceptance criteria continue to be met (i.e., the peak RCS pressure remains within 110 percent of design limits). Additionally, since the licensee performed the analysis using an NRC-approved methodology for a power level of 3315 MWt, the NRC staff finds that it bounds the proposed power level of 3304 MWt. Therefore, the NRC staff finds the analysis acceptable for the proposed 1.66-percent power uprate.

3.2.2.4.3 Loss of Normal Feedwater Flow and Loss of All AC Power

A loss of normal feedwater event reduces the capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped, or if an alternate supply of feedwater were not supplied to the plant, core damage could occur. Currently, the D. C. Cook Unit 1 loss of normal feedwater analysis uses the LOFTRAN computer code to model a core power level of 3457 MWt. This power level bounds the proposed uprate level of 3304 MWt with an 0.34 percent uncertainty. Since the current analysis bounds the power uprate, the NRC staff find the analysis acceptable.

The loss of all alternating current (ac) power to the station auxiliaries results in a loss of all power to auxiliary systems, including the RCPs, condensate pumps, etc. Upon the loss of power, core cooling and removal of residual heat is accomplished by natural circulation in the reactor coolant loops, aided by auxiliary feedwater, SG pressure-operated relief valves (PORVs), and safety valves on the secondary side of the plant.

For this transient, the licensee used the LOFTRAN computer code with input assumptions that ensure conservative results. The acceptance criteria for this transient include preventing the minimum DNBR from going below the limit value, preventing the RCS pressure from going above 110 percent of the design value, and preventing pressurizer overfill.

The licensee's current analyses model the transient at a core power level of 3457 MWt. This power level bounds the proposed uprate level of 3304 MWt with a 0.34 percent uncertainty. Additionally, these analyses show that the above acceptance criteria continue to be met. Since the licensee's analyses show that the acceptance criteria will continue to be met at the proposed uprated conditions, and since the analyses bound the proposed power level of 3304 MWt, the NRC staff finds them acceptable for the proposed 1.66-percent MUR power uprate.

3.2.2.4.4 Rupture of a Control Rod Drive Mechanism Housing

A control rod drive mechanism (CRDM) pressure housing rupture may result in the ejection of an RCCA and drive shaft to their fully withdrawn position. The consequences of this failure include a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage.

For the hot full power (HFP) condition, the licensee evaluated their accident at a power level of 102 percent of 3250 MWt. This power level bounds the proposed uprate level of 3304 MWt with a 0.34 percent uncertainty. Since the current analysis power level bounds that of the proposed power uprate, the NRC staff finds the analysis acceptable for the proposed 1.66-percent MUR power uprate.

3.2.2.4.5 RCCA Misalignment and RCCA Drop

The RCCA misalignment accidents include a dropped RCCA, a dropped RCCA bank, and a statically misaligned RCCA. The scenarios considered most severe misaligned RCCA transient occurs when one RCCA is fully inserted. RCCA drops occur when the drive mechanism loses power. The drop causes a decrease in core power, specifically near the dropped rod, and an increase in the hot channel factors for the remaining rods. As the rod control system tries to restore power to the initial power level, the automatic rod withdrawal will further decrease the safety margin of the fuel. The action of the automatic rod control will cause a power overshoot, which sets this condition as the limiting case for the accident.

The licensee analyzed their existing accident statepoints for continued applicability at the proposed uprated power conditions and determined that they continue to remain valid. The licensee also evaluated the increase in nominal heat flux and confirmed that the DNB design basis continues to be met.

Since the postulated accidents will continue to meet the SRP acceptance criteria at the proposed uprated conditions, the NRC staff finds them acceptable for the proposed 1.66-percent MUR power uprate.

3.2.2.4.6 Partial and Complete Loss of Forced Reactor Coolant Flow

A mechanical or electrical failure in one or more RCPs or a fault in the power supply to these pumps may cause a partial or complete loss of forced coolant flow. If the reactor is powered at the time of the incident, the loss of coolant flow causes a rapid increase in coolant temperature. This increase could result in DNB. The licensee evaluated its existing accident statepoints for continued applicability at the proposed uprated power conditions and determined that the nominal core heat flux increases, but the other statepoints remain valid. Nevertheless, with the greater heat flux, the licensee's evaluation shows that the SRP DNB design basis continues to be met. Since the accident statepoints will continue to meet the SRP acceptance criteria at the proposed uprated conditions, the NRC staff finds it acceptable for the proposed 1.66 percent MUR power uprate.

3.2.2.4.7 Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

An uncontrolled RCCA bank withdrawal accident may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. However, for D. C. Cook Unit 1, reactivity feedback terminates the initial power increase resulting from the rod withdrawal. Thus, power increases until its peak and rapidly decreases by the time the rods begin to drop.

The licensee evaluated this accident scenario for the proposed power uprate and determined that because of the small increase in reactor power, the rod drop times remain essentially unaffected. In addition, with the exception of the nominal core heat flux increasing, the other statepoints remain valid. Nevertheless, with the greater heat flux, the licensee's evaluation shows that the SRP DNB design basis continues to be met.

Since the licensee evaluated this accident scenario at the proposed uprated conditions, and since the accident continues to meet the SRP acceptance criteria, the NRC staff finds the evaluation acceptable for the power uprate to 3304 MWt.

3.2.2.4.8 Chemical Volume and Control System

The licensee evaluated the chemical volume and control system (CVCS) malfunction (boron dilution) event for the proposed 1.66-percent power uprate and determined that the uprate would have an insignificant impact on the reactor trip time assumed in the event. Since the reactor trip time (both time to trip and time for control rod insertions) would remain essentially unchanged, the plant will continue to meet the CVCS malfunction acceptance criteria. Additionally, because the reactor is not at full power for plant modes other than power operation (Mode 1), the licensee determined that the power uprate will not impact the analyses for Modes 2 through 6 (startup, hot standby, hot shutdown, cold shutdown, and refueling). Therefore, for Modes 2 through 6, the accident will continue to meet the SRP acceptance criteria.

Since the licensee evaluated this event at the proposed uprated conditions, and since the accident continues to meet the SRP acceptance criteria, the NRC staff finds the evaluation acceptable for the proposed power uprate to 3304 MWt.

3.2.2.4.9 Excessive Heat Removal Due to Feedwater System Malfunctions

A feedwater system malfunction occurs when relatively cool feedwater is supplied to the SGs. This action causes excess heat removal by the secondary side, which increases core power above full power. This accident could occur if a failure in the feedwater control system leads to the simultaneous full opening of the feedwater control valves. However, the protection afforded by the overpower-overtemperature protection of the RCS (nuclear overpower and DT trips) prevents a power increase that could lead to a violation of the DNBR limits.

The licensee stated that the power uprate will not affect this event (full opening of the feedwater control valves) for the hot zero power (HZP) case, but the uprate will affect the event at HFP. Therefore, the licensee analyzed this event at HFP, this analysis for the HFP case is discussed in Section 3.2.2.7.1, and the NRC staff found the licensee's analysis acceptable. Since the power uprate does not affect this event at HZP, and since the NRC staff found it acceptable for HFP, the NRC staff continues to find the licensee's evaluation acceptable for the power uprate to 3304 MWt.

Another feedwater system malfunction is the malfunction that reduces feedwater temperature. The licensee evaluated this accident and determined that the feedwater enthalpy reduction results in a transient that is very similar to, and bounded by, that of an excessive load increase transient. Because the excessive load increase transient bounds this event, the licensee determined that it does not need to be reanalyzed for the MUR power uprate. The NRC staff agrees with the licensee's assessment that the reduction in feedwater temperature incident does not need to be reanalyzed for the proposed MUR power uprate.

3.2.2.4.10 Excessive Load Increase Incident

An excessive load increase incident occurs when a rapid increase in steam flow causes a power mismatch between the reactor core power and the SG load demand. The RCS accommodates a 10-percent step-load increase or a 5-percent-per-minute ramp-load increase between 15-percent and 100-percent power. However, loading rates exceeding these values may result in a reactor trip initiated by the reactor protection system. The following reactor trips protect the RCS during this accident: overtemperature DT, overpower DT, power range high neutron flux, and low pressurizer pressure.

The licensee evaluated this accident by conservatively comparing bounding plant conditions for deviations in core power, average coolant temperature, and RCS pressure to conditions necessary to exceed the core thermal limits. Because it had sufficient margin to the core thermal operating limits, the licensee concluded that the minimum DNBR remains above the limiting value for all cases.

Because the DNBR values for this accident remain below their limit values, the transient will continue to meet the SRP acceptance criteria at a power level of 3304 MWt. Therefore, the NRC staff finds the licensee's evaluation acceptable for the proposed power level of 3304 MWt.

3.2.2.4.11 Rupture of a Steam Pipe - Core Response Analysis

The rupture of a steam pipe accident models an uncontrolled steam release from an SG, which includes steam pipe breaks and valve malfunctions. The most limiting steam pipe accidents occur when the reactor is in a hot shutdown condition. With the RCS in this condition the steam release will cool the RCS. Since the RCS has a negative moderator temperature coefficient, this cooling may cause the core to become critical and return to power, possibly causing fuel damage. The safety injection (SI) system eventually terminates this accident by supplying boric acid to shut down the core.

Because the most limiting case of this accident occurs at hot shutdown (no load) conditions, and because the SI system terminates the accident independent of power level, the core response portion of the steam pipe rupture accident remains independent of power level. Since the core response of this accident is not influenced by power level, the NRC staff finds the core response acceptable for the licensee's proposed power uprate to 3304 MWt.

3.2.2.4.12 Rupture of a Control Rod Drive Mechanism Housing MODE 3

A CRDM pressure housing rupture may result in the ejection of an RCCA and drive shaft to their fully withdrawn position. The consequences of this failure include a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage.

For this analysis, the licensee used the NRC-approved TWINKLE, FACTRAN, THINC, and LOFTRAN computer codes. Also, the licensee analyzed the RCCA ejection at the beginning and end of the core life for both HFP and HZP conditions, in order to bound the fuel cycle and expected operating conditions.

For the HFP conditions, the analyses explicitly model a 2-percent power measurement uncertainty. This 2-percent uncertainty bounds the proposed 1.66-percent uprate. Therefore, the NRC staff finds the current analyses acceptable for the HFP condition at 3304 MWt.

Alternately, for the HZP cases, the licensee determined that the proposed power uprate would not affect the analysis since it is performed at shutdown conditions. Since the proposed power uprate has no effect on the HZP accident analyses, the NRC staff finds them acceptable for the uprate to 3304 MWt.

3.2.2.4.13 Anticipated Transients Without Scram

The licensee analyzed this event for the proposed 1.66-percent (3304 MWt) power uprate, as required by 10 CFR 50.62, "Requirements for reduction of risk from Anticipated Transients Without Scram (ATWS) events for light-water-cooled nuclear power plants." The ATWS rule for Westinghouse plants requires that the plant implement a system diverse from the reactor trip system to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS. The system is called the ATWS mitigation system action circuitry (AMSAC). D. C. Cook Unit 1 meets the requirements of 10 CFR 50.62(c) because an AMSAC system has been installed and is operable.

Additionally, to demonstrate that the proposed power uprate will not result in a transient peak RCS vessel pressure above the American Society of Mechanical Engineers (ASME) stress level "C" limit of 3200 psig, the licensee relied upon the Westinghouse generic ATWS analysis (Reference 8). The licensee determined that the ATWS analyses that are applicable to D. C. Cook Unit 1 are those for the Westinghouse 4-loop pressurized-water reactor (PWR) with Model 51 SGs.

The model for the limiting loss of load ATWS event shows that the peak RCS pressure reaches only 2974 psia. This pressure results in a margin of 226 psi to the peak RCS limit of 3200 psia. However, the proposed power uprate and the generic model conditions have several differences, including a lower reactor power level, lower auxiliary feedwater capacity, and a more positive moderator temperature coefficient. Accounting for these variations at the proposed uprated conditions, the licensee determined that the peak pressure would increase to 3063 psia.

Since the D. C. Cook Unit 1 ATWS peak pressures remain below the limit of 3200 psig, the NRC staff finds the licensee's assessment acceptable to support the proposed power uprate to 3304 MWt.

3.2.2.4.14 Station Blackout

In its coping analysis for a station blackout (SBO) event, the licensee performed its decay heat removal inventory requirements assuming a 2-percent power uncertainty. Since this analysis continues to bound the proposed power level of 3304 MWt with a 0.34 percent uncertainty, the NRC staff finds it acceptable for the proposed power uprate.

3.2.2.5 Design Transients

To support reduced temperature and pressure operation at D. C. Cook Unit 1, the licensee performed analytical work for a 3600 MWt rerating. Even though the NRC staff did not specifically review this work for a power uprate, the NRC staff has reviewed and approved this work for operation of the plant at reduced temperature and pressure conditions (Reference 6). The NRC staff's review indicated that the specified operating parameters and accidents were acceptable as proposed. These design parameters included RCS flow, RCS pressure, T_{hot}, T_{avg}, SG outlet temperature, steam temperature, steam pressure, and feedwater temperature. In addition, the licensee accounted for design limits on primary-to-secondary pressure differential by adjusting the pressurizer pressure setpoint to 2100 psia when operating at full-power steam pressures at or below 679 psia. Otherwise, the licensee can put the setpoint to 2250 psia. The NRC reviewed and approved these operating pressures in D. C. Cook Unit 1 License Amendment No. 126 (Reference 6).

As part of an effort to increase the SG tube plugging limits to 30 percent, the licensee revised the D. C. Cook loss of load and loss of power design transients to reflect a 3-percent pressurizer safety valve tolerance. The NRC staff found these changes acceptable in D. C. Cook License Amendment Nos. 214 (Unit 1) and 199 (Unit 2) (Reference 9).

Another change to the SGs involved the D. C. Cook Unit 1 SG replacement. The licensee replaced the Westinghouse Model 51 SGs with Babcock and Wilcox International (BWI) SGs. However, since the design-basis specifications for transients for the BWI SGs remained the same as the previous SGs, the licensee concluded that the transient behavior will not change.

Since the NRC staff previously found the above parameters acceptable, and since the parameters bound those for the proposed 1.66-percent MUR power uprate design transients, the NRC staff finds the transients acceptable for a power uprate to 3304 MWt.

3.2.2.6 Auxiliary Equipment Design Transients

The licensee reviewed the NSSS auxiliary equipment design transients and determined that the proposed MUR power uprate could potentially affect the temperature transients that are impacted by the full-load NSSS design temperatures. Those transients currently assume a full-load NSSS worst-case T_{cold} of 560 °F and T_{hot} of 630 °F. These values bound those that would be achieved because of the proposed 1.66-percent MUR power uprate. Therefore, the currently assumed transients would bound those of the proposed MUR power uprate.

Since the current auxiliary equipment design transients bounds those of the proposed power uprate, the NRC staff finds them acceptable for a power uprate to 3304 MWt.

3.2.2.7 Accidents and Transients Analyses

3.2.2.7.1 Feedwater System Malfunctions (full power case)

The NRC staff reviewed the feedwater system malfunction accident scenario and found it acceptable for the HZP cases and the temperature reduction cases. However, the licensee reanalyzed this accident for the HFP cases where a failure in the feedwater control system leads to the simultaneous full opening of the feedwater control valves. This action would cause excess heat removal by the secondary side and increase core power above full power.

The licensee evaluated this accident scenario for the proposed uprated conditions using the LOFTRAN computer code, the NRC-approved Westinghouse reload safety evaluation methodology described in Westinghouse Topical Report WCAP-92772-P-A (Reference 10), and the NRC-approved DNB methodology described in Westinghouse Topical Report WCAP-11397-P-A (Reference 11). The licensee also used assumptions that ensure conservative results for the feedwater system malfunction.

The results of the analyses for the overpressure and DNB cases indicate that the SRP acceptance criteria continue to be met (i.e., the minimum DNBR remains above the limit value, and the peak RCS pressure remains within 110 percent of the design limits). Additionally, since the licensee performed these analyses using NRC-approved codes and methodologies for the proposed uprated conditions, the NRC staff finds them acceptable for the proposed 1.66-percent MUR power uprate.

3.2.2.7.2 Loss of External Electrical Load - DNB Case

A loss of external electrical load event occurs when an electrical disturbance causes the loss of a significant portion of the generator load. Upon a full loss of load, the reactor protection system automatically initiates a reactor trip. The licensee evaluated this transient to ensure that no core damage occurs (DNB analysis). For the DNB analysis, the licensee assumed that the reactor used manual rod control and automatic pressurizer pressure control, with two cases for reactivity feedback (one maximum and one minimum).

To support this analysis at the proposed uprated conditions, the licensee used the LOFTRAN computer code, the reload safety evaluation methodology described in Westinghouse Topical Report WCAP-92772-P-A, and the DNB methodology described in Westinghouse Topical Report WCAP-11397-P-A. The licensee also used assumptions that ensure conservative results for the loss of electrical load.

The results of the analyses for the DNB cases indicate that the SRP acceptance criteria continue to be met (i.e., the minimum DNBR remains above the limit value). Additionally, since the licensee performed these analyses using NRC-approved codes and methodologies for the proposed uprated conditions, the NRC staff finds them acceptable for the proposed 1.66-percent MUR power uprate.

3.2.2.7.2 Uncontrolled RCCA Bank Withdrawal at Power

Similar to the RCCA withdrawal from subcritical, an uncontrolled RCCA withdrawal at power accident can be caused by a malfunction of the reactor control or rod control systems. This withdrawal will also uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The licensee analyzed this accident for reactivity insertion rates from 0.6 pcm/sec to 100 pcm/sec for power levels of 10 percent, 60 percent, and 100 percent of nominal power to ensure that it meets the acceptance criteria.

For this accident analysis, the licensee used the LOFTRAN computer code, the reload safety evaluation methodology described in Westinghouse Topical Report WCAP-92772-P-A, and the DNB methodology described in Westinghouse Topical Report WCAP-11397-P-A. The licensee also used assumptions that ensure conservative results for the uncontrolled RCCA bank withdrawal.

The results of the licensee's analysis indicate that the SRP acceptance criteria continue to be met (i.e., the minimum DNBR remains above the limit value, the fuel centerline temperatures do not exceed the melting point, and the primary and secondary pressures do not exceed 110 percent of design pressure). Since this analysis continues to meet the SRP acceptance criteria, the NRC staff finds it acceptable for the proposed 1.66-percent MUR power uprate.

3.2.2.8 <u>Steam Generators</u>

Recently, Westinghouse identified issues with SG water level setpoint uncertainties. One problem deals 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.

The licensee recently replaced the Westinghouse SGs with Babcock and Wilcox Model 51R SGs. These SGs differ from the Westinghouse SGs. These differences include an improved internal feedwater distribution system, different separator unit, use of a lattice grid support plate, and improved tubing material. Additionally, the Model 51R SG's narrow range level region differs significantly from that of the Westinghouse SGs. Therefore, the licensee concluded that the level instrumentation problems identified in the Westinghouse SGs do not affect the current D. C. Cook Babcock and Wilcox SGs. Based on the above, the NRC staff finds the SGs level instrumentation acceptable for the proposed power uprate.

3.2.2.9 <u>Fuel Evaluation</u>

3.2.2.9.1 <u>Nuclear Design</u>

For the nuclear design issues associated with the power uprate, the licensee evaluated two cases. Since the licensee plans to uprate D. C. Cook Unit 1 mid-cycle, the first case involved the mid-cycle power uprate of D. C. Cook Unit 1 during Cycle 18, and the second case involved the subsequent cycles. During the mid-cycle uprate, the licensee plans to lower the HFP inlet temperature of the core by 0.4 °F so that the HFP vessel average moderator temperature will remain at 556 °F. In addition, the licensee plans to keep the RCS pressure at 2100 psia. These conditions will help minimize the fuel thermal-hydraulic impact.

Alternately, for the subsequent cycle operation, the licensee used NRC-approved methodologies to analyze an increase in the system pressure from 2100 to 2250 psia and a moderator temperature increase from 556 °F to 574 °F. The licensee determined that these changes, in conjunction with changes to the feed fuel enrichment, will similarly only result in small changes to the axial and radial power distribution and the critical boron concentration.

Based upon its analysis, the licensee concluded that the changes to the power distributions caused by the power uprate will be small in comparison to the typical cycle-to-cycle variability associated with reload. The licensee concluded that the changes to the boron concentrations, reactivity coefficients, shutdown margin, and other safety analysis inputs will also be small. In addition, during the standard reload design process, future cycles are analyzed to confirm that all of the applicable limits are met, and any differences in key parameters are routinely addressed.

Since the power uprate will only cause small changes to the nuclear design based on the use of previously approved methodologies, and since future changes to the nuclear design are evaluated in the standard reload design process, the NRC staff finds the nuclear design acceptable for the proposed power uprate to 3304 MWt.

3.2.2.9.2 Fuel Rod Design

The licensee evaluated the fuel rod design criteria at the proposed uprated conditions for Cycle 18 and also evaluated a representative future cycle. The licensee performed these analyses using NRC-approved models and methods, and the results of the evaluation indicate that the rods would be expected to meet all of the design criteria at the proposed uprated conditions. Additionally, the licensee reports the results of the analyses in the Reload Safety Evaluation Report as part of the normal design process.

Since the licensee performed its analyses using NRC-approved models and methods, and since the fuel rods will meet all of the design criteria at the proposed uprated conditions, the NRC staff finds them acceptable for the proposed uprate to 3304 MWt.

3.2.2.9.3 Core Thermal-Hydraulic Design

For the D. C. Cook Unit 1 DNB analyses, the licensee kept the DNBR design limits and safety analysis limits unchanged from the previous limits. The licensee also kept the DNBR portion of the core limits and the axial offset limits unchanged to minimize the impact on the OTDT and OPDT protection setpoints. This decision is conservative because, with the reduced measurement uncertainty, the licensee could have lowered the DNBR limits.

To ensure that it meets these limits at the proposed uprated power conditions, the licensee performed the thermal-hydraulic design analyses and evaluations with a power level increase of 1.7 percent. Since the licensee used the improved thermal design procedure, the licensee is only required to perform these analyses using nominal values (i.e., a 1.66-percent power uprate).

The licensee's analyses assuming a 1.7-percent increase bound the proposed power uprate of 1.66 percent, therefore, the NRC staff finds the D. C. Cook Unit 1 thermal-hydraulic analyses acceptable for the proposed power uprate to 3304 MWt.

3.2.2.9.4 Fuel Structural Evaluation

The licensee also evaluated the 15x15 optimized fuel assembly design to determine the impact of the power uprate on the fuel assembly structural integrity. The licensee determined that the original plate motions remain applicable at the proposed uprated conditions, and since the uprate would have an insignificant impact on the operating and transient loads, there is no adverse effect on the fuel assembly functional requirements. From these determinations, the licensee concluded that the fuel assembly structural integrity is not affected, and the seismic and LOCA evaluations for the fuel assembly designs remain applicable.

Since the structural integrity is not affected by the power uprate, and since the seismic and LOCA evaluations of the fuel assembly designs remain applicable, the NRC staff finds the fuel assembly structural integrity acceptable for a power level of 3304 MWt.

3.2.2.10 System Design

3.2.2.10.1 NSSS Interface Systems

3.2.2.10.1.1 Residual Heat Removal System

Various D. C. Cook Unit 1 TSs require that the plant be capable of being placed in cold shutdown within 36 hours. In addition, the current licensing basis requires that under normal operating conditions, the residual heat removal (RHR) system be capable of reducing RCS temperature to 140 ^{o}F within 20 hours following a reactor shutdown. The licensee performed these analyses assuming a core power of 3411 MWt. The licensee's analyses showed that for a single train cooldown, the TS requirement of 36 hours is met, and for a dual train cooldown, the licensing-basis requirement of 20 hours is met.

Since the analyses at 3411 MWt bound the proposed power level of 3304 MWt, the NRC staff finds the RHR system acceptable for operation at the 3304 MWt power level.

3.2.2.10.1.2 Emergency Core Cooling System

The licensee stated that it performed the current D. C. Cook Unit 1 long-term core cooling analyses for a power level of 3481 MWt. Because of this power assumption, the licensee concluded that the proposed power uprate would not affect the decay heat already considered for the ECCS system. Consequently, the uprate will not affect the performance of the ECCS.

Since the current ECCS analysis bounds the proposed uprated conditions, the NRC staff finds it acceptable for the proposed uprated power level of 3304 MWt.

3.2.2.10.1.3 NSSS Control Systems

Upon evaluation of the effects of the power uprate on the NSSS control systems, the licensee determined that the limiting Condition I transients are the ramp-load increase (1 percent per minute between 20-percent and 100-percent power), ramp-load decrease (5 percent per minute between 100 and 20 percent power without steam dump), and 10-percent load decrease (200 percent per minute). The licensee's analyses showed that for the proposed uprated conditions, these transients will not cause a spurious reactor trip or an engineered safety feature (ESF) actuation.

In addition, the licensee analyzed the turbine and reactor trip transients initiated from 100-percent power with steam dump actuation at the proposed uprated conditions. From its analysis, the licensee concluded that the plant/turbine trip controller mode of steam dump operation will continue to perform its function.

Since the above transients will continue to meet the design bases at the proposed uprated conditions, the NRC staff finds them acceptable for the proposed 1.66-percent power uprate.

However, for the 40-percent load rejection transient (100-percent to 60-percent power at 200 percent per minute without steam dump), the licensee's current accident analysis shows that for higher T_{avg} and a higher core burnup, this transient may cause a reactor trip. Additionally, the reduced steam dump margin caused by the power uprate will tend to worsen the situation. As part of the proposed MUR power uprate request, the licensee has committed to reanalyze the steam dump capacity prior to implementation of the proposed MUR power uprate to ensure that it has adequate margin. If the analysis requires modifications be made to the steam dump system, the modifications will also be made prior to implementation of the proposed MUR power uprate. The licensee will revise the Updated Final Safety Analysis Report (UFSAR) to reflect the results of the analysis and any modifications that are warranted.

Based on the licensee's commitment to maintain the margin to trip for the 40-percent load rejection transient, which is confirmed by a licensee condition, the NRC staff finds it acceptable for the power uprate to 3304 MWt.

3.2.2.10.1.4 NSSS Pressure Control Component Sizing

The licensee evaluated the sizing of the pressure control components to determine if the capacity remained acceptable at the proposed uprated conditions. These components included the pressurizer heaters, pressurizer spray, and pressurizer PORVs.

Since the heatup time of the reactor from cold shutdown to hot standby is not impacted by power level, the licensee concluded that the pressurizer heater capacity is not affected. Alternately, for pressurizer spray, the licensee determined that pressurizer sprays could accommodate a design NSSS power level of 3600 MWt, assuming a smaller spray flow than design values, without challenging the pressurizer PORVs.

Since the pressure control components can accommodate a power level of 3600 MWt without challenging the PORVs, the NRC staff finds the sizing of the PORVs acceptable for a power level of 3304 MWt.

3.2.2.11 Low Temperature Overpressure Protection System

The low temperature overpressure protection system (LTOP) system operates when the plant is in a shutdown condition. Therefore, the proposed power uprate would not impact the response of the LTOP system.

3.2.2.12 Changes to Technical Specifications

The licensee proposed to modify TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," to limit the indicated T_{avg} for the OTDT and OPDT (T' and T'') to less than or equal to 574 °F and 562.1 °F, respectively. The NRC staff reviewed and found these changes acceptable in Section 3.2.2.2 of this safety evaluation.

In TS Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation," the licensee proposed the insertion of new values for the setpoints with inoperable steamline safety valves to be consistent with the proposed power uprate. For D. C. Cook Unit 1, with one, two, and three steamline safety valves inoperable, the licensee proposed to change the maximum allowable power levels from 65.1 percent, 46.5 percent, and 28.0 percent to 63.8 percent, 45.5 percent, and 27.4 percent, respectively. 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.

Since the licensee used a conservative heat balance calculation to determine the new power range neutron flux high setpoints, the NRC staff finds them acceptable for the proposed power uprate to 3304 MWt.

3.2.3 Summary

The NRC staff has reviewed the licensee's analyses to support operations of D. C. Cook Unit 1 at a maximum core power level of 3304 MWt. The NRC staff finds that the licensee has satisfactorily addressed the areas discussed above, the supporting safety analyses were performed using NRC-approved methods, 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.

In addition, to support the proposed uprate, the licensee made the following commitments:

> Prior to implementing this uprate, a engineering/reload safety evaluation will be performed to ensure that the core design bounds the uprated condition. The UFSAR will also be updated to reflect the safety evaluation.
>
> 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-power uprate. The UFSAR will be updated to reflect the analysis and/or commitments

Both of these analyses will be required by license condition, the NRC staff finds the proposed MUR 1.66-percent power uprate acceptable with respect reactor systems performance. Therefore, the NRC staff concludes that the supporting analyses are acceptable for the proposed power uprate to 3304 MWt.

3.3 Electrical Systems

3.3.1 Regulatory Evaluation

General Design Criteria (GDC) 17, "Electric Power Systems," of Appendix A to 10 CFR Part 50 requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that the containment integrity and other vital functions are maintained in the event of postulated accidents.

The regulation at 10 CFR 50.63 requires that all nuclear power plants have the capability to withstand a loss of all AC power for an established period of time, and to recover therefrom. The plant was previously evaluated for SBO.

The regulation at 10 CFR 50.49, “Environmental Qualification of Electric Equipment important to Safety for Nuclear Power Plants,” requires licensees to establish programs to qualify electric equipment important to safety. Under the rule, each licensee must (1) prepare and maintain a record of qualification to document that each item of equipment subject to the rule is qualified for its application, and (2) meets its specified performance requirements when subjected to the environmental conditions predicted to be present when it must perform its safety function up to the end of qualified life.

As described by the licensee’s application dated June 28, 2002, the main generator is rated at 1280 million volt amperes (MVAs) at a 0.9 power factor. The station output generated at 26 kV is fed through an isolated phase bus to the primary windings of the main transformer where it is stepped up to 345 kV voltage. The unit auxiliary transformer (UAT) supplies power to balance-of-plant (BOP) systems under normal operating conditions. Upon a trip of the main generator, the station auxiliaries are automatically transferred to the preferred offsite power source through the reserve auxiliary transformer to assure continued power to equipment when the main generator is off-line. The station service transformers provide power to Class 1E power systems. The station distribution system consists of various auxiliary electrical systems to provide electrical power during all modes of operation and shutdown conditions.

3.3.2 Technical Evaluation

Following is the evaluation of the grid stability, main generator, transformers, emergency diesel generators (EDGs), SBO, and EQ:

3.3.2.1 Grid Stability

The present output of the main generator at 100-percent RTP is 1077 Megawatt Electric (MWe), which is equivalent to 1197 MVA at a power factor of 0.9. This is below its maximum rating of 1280 MVA of the main generator. The licensee has evaluated the impact of power uprate on the grid stability and determined that it will have no impact. The impedance values assumed for main generator will not be changed by the power uprate. Therefore, the proposed 1.66-percent power uprate will not impact the grid stability analysis.

The NRC staff reviewed the licensee’s submittal and concluded that the impact of the power uprate on grid stability is insignificant. Therefore, the plant continues to meet GDC-17 for grid stability with the proposed MUR power uprate.

3.3.2.2 Main Generator

The main generator is rated at 1280 MVA at 0.9 power factor. A power uprate of 1.66 percent of the current generator output of 1077 MWe yields a maximum net increase in generator output to 1095 MWe at 0.9 power factor. This is equivalent to 1217 MVA, which is below the generator nameplate rating of 1280 MVA. The main generator and associated cooling equipment are designed to accept the maximum generator output at the proposed uprated conditions. There is no impact of the power uprate on the protective relay settings of the main generator.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1217 MVA is below the maximum main generator design rating of 1280 MVA and, therefore, operating the main generator at the proposed uprated power conditions is acceptable.

3.3.2.3 Main Transformer

The main transformer rating of 1300 MVA provides a margin of 20 MVA above the maximum rating of 1280 MVA of the main generator. A 1.66-percent power uprate will raise the main generator output to 1217 MVA which is below the main transformer rating of 1300 MVA.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1.66 percent is below the maximum main transformer design rating of 1300 MVA and, therefore, operating the main transformer at the proposed uprated power conditions is acceptable.

3.3.2.4 Isophase Bus

The Isophase bus duct connects the main generator to the primary windings of the main transformer and the UAT. The isolated phase bus duct and associated cooling equipment are designed to accept the maximum generator output and will continue to support plant operations at uprated conditions.

The NRC staff reviewed the licensee's application and concluded that the impact of the proposed 1.66-percent power uprate is below the design rating of the isophase bus and, therefore, operating the isophase bus at the proposed uprated power conditions is acceptable.

3.3.2.5 Unit Auxiliary Transformer/Reserve Auxiliary Transformer

The UAT is rated at 18/24/30 MVA, 26/4.16 kV. The reserve auxiliary transformer (RAT) is rated at 18/24/30 MVA, 34.5/4.16 kV. The UAT supplies power to BOP systems under normal operating conditions. Upon a trip of the main generator, the station auxiliaries are automatically transferred to the preferred offsite power source through the RAT to assure continued power to equipment when the main generator is offline. The UAT/RAT design ratings bound any expected bus loading increases that result from increased system flow rates in the secondary systems.

The NRC staff reviewed the licensee's application and concludes that the increase in house loads resulting from the proposed 1.66-percent power uprate is below the maximum UAT/RAT design rating and, therefore, operating the UAT/RAT at the proposed uprated power conditions is acceptable.

3.3.2.6 Emergency Diesel Generators

The EDGs supply the source of power following a loss of offsite power or degraded voltage conditions. The EDGs automatically supply AC power to the Class 1E buses in order to provide motive and control power to equipment required for safe shutdown of the plant and mitigation and control of accidents. Existing accident analyses bound the increase in reactor core decay heat. The licensee has evaluated the heat removal systems, including the RHR, component cooling water (CCW), and emergency service water system pumps and determined that they would have an insignificant effect on system flow due to the proposed power uprate, and thus, on motor loads. The BOP and NSSS system and component performance reviews determined that there is no need to increase EDG loading as a result of the proposed power uprate.

The NRC staff's review determined that the proposed power uprate would not affect the loading on the EDG and, therefore, the licensee will continue to meet GDC-17 requirements with the proposed MUR power uprate.

3.3.2.7 Station Blackout

The methodology and the assumptions associated with the SBO analysis with regard to equipment operability are unchanged with the power uprate. The condensate inventory required for decay heat removal is bounded by the original 102-percent analysis. There are no changes to direct current (DC)-powered components or inverter fed ac-powered components; therefore, the Class 1E battery capacity is not impacted. No changes to the instrument air system or components fed by the instrument air are associated with the proposed power uprate. The proposed power uprate would not impact the current containment isolation evaluations. In addition, the proposed power uprate would not impact the RCS inventory evaluations, RCP seal leak rates, the normal TS leak rates from the RCS, or letdown isolation capabilities. Therefore, the ability of the plant to respond to an SBO will not be altered due to the proposed power uprate.

The NRC staff reviewed the licensee's application and concluded that the proposed power uprate would not adversely affect the ability of the plant to mitigate a postulated SBO event and the plant continues to meet the requirements of 10 CFR 50.63 and is therefore acceptable.

3.3.2.8 Environmental Qualification of Electrical Equipment

The licensee analyzed the impact of the power uprate on the environmental qualification of the electrical equipment and determined that all the current D. C. Cook Unit 1 analyses continue to support EQ qualification levels up to a maximum error-adjusted power level of 3315 MWt.

The NRC staff reviewed the licensee's submittal and determined that qualification data for a power level of 3315 MWt are bounding for the proposed uprate to 3304 MWt; therefore, no changes to the EQ program are required for this power uprate, and the plant continues to meet the requirements of 10 CFR 50.49.

3.3.3 Summary

The results of the NRC staff's evaluations as discussed above, show that the increase in core thermal power from the proposed MUR power uprate will have negligible impact on grid stability, SBO, or the EQ of electrical components. Therefore, the NRC staff finds the proposed power uprate to 3304 MWt acceptable with respect to electrical systems performance.

3.4 Civil and Engineering Mechanics

3.4.1 Regulatory Evaluation

The NRC staff reviewed the proposed MUR power uprate amendment at D. C. Cook Unit 1, as it relates to the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and the BOP systems and components. The NRC staff reviewed the impact of the proposed power uprate on the NSSS components and BOP systems and components; the Motor-Operated Valve (MOV) Program; the Air and Hydraulic-Operated Valve (AHOV) Program; the High-Energy Line Break (HELB) Program; and the mechanical piping design.

3.4.2 Technical Evaluation

3.4.2.1 Reactor Vessel Structural Evaluation

The impact of the proposed MUR power uprate on the D. C. Cook Unit 1 reactor vessel was evaluated by the licensee. The licensee indicated that there is no change to any design inputs that were previously considered in the reactor vessel evaluations for the Rerating Program, which was approved for D. C. Cook Unit 1 in License Amendment No. 126 (Reference 6). Therefore, the NRC staff finds that the proposed power uprate to 3304 MWt does not change the results contained in the D. C. Cook Unit 1 reactor vessel analytical report.

The licensee concluded that the D. C. Cook Unit 1 reactor vessel continues to satisfy the applicable requirements of Section III (Nuclear Vessels) of the ASME *Boiler and Pressure Vessel Code* (B&PV Code), 1965 edition through winter 1966 addenda, in accordance with the reactor vessel design requirements. Therefore, the NRC staff concurs with the licensee's conclusion that the current design of the reactor vessel continues to be in compliance with licensing-basis codes and standards for the proposed power uprate.

3.4.2.2 Reactor Internals

The licensee indicated that it has performed analyses to demonstrate that the reactor internals can perform their intended design functions at the proposed 1.66-percent uprated conditions.

The licensee performed an evaluation to determine the hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly would remain seated and stable for all conditions. The licensee stated that the evaluation results indicate that the downward force remains essentially unchanged, indicating that the reactor internals would remain seated and stable for the proposed 1.66-percent power uprate.

The licensee stated that the proposed 1.66-percent power uprate would not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to reevaluate the structural effects from the seismic operating-basis earthquake, safe-shutdown earthquake loads, and the LOCA hydraulic and dynamic loads.

The licensee also stated that with regard to flow-induced vibration, the lowest vessel/core inlet coolant temperature would remain unchanged. The corresponding vessel outlet coolant temperature would increase by 1.4 °F. This temperature change causes a change in water density that has a negligible impact on the vibratory response of the reactor internals. The design power capability parameters for the current design basis and the proposed MUR uprate would essentially remain the same. Therefore, the licensee concluded that there is no significant impact on the performance of the reactor internals with regard to flow-induced vibration.

The licensee stated that evaluations were performed to demonstrate that the structural integrity of the reactor components is not adversely affected by the proposed 1.66-percent power uprate. The licensee indicated that the reactor internals components subjected to heat generation effects (either directly or indirectly) are the upper core plate, the lower core plate, and the baffle-barrel region. The licensee also indicated that for all reactor internal components, except the lower core plate and the upper core plate, the stresses and cumulative fatigue usage factors for D. C. Cook Unit 1 would be unaffected by the proposed 1.66-percent power uprate because the previous analyses remain bounding.

The licensee stated that due to the lower core plate's proximity to the core, it is subjected to the effects of heat generation. The heat generation rates in the lower core plate due to gamma heating can cause a significant temperature increase in this component. A structural evaluation was performed to demonstrate that the structural integrity of the lower core plate is not adversely affected by the revised design conditions. The cumulative fatigue usage factor of the lower core plate, including the effects of the increase in the heat generation rates, is small (0.237), and the lower core plate is structurally adequate for the proposed 1.66-percent power uprate.

The licensee further stated that the thermally-induced displacements of the baffle-former bolts for the proposed 1.66-percent power uprate relative to the original design conditions were calculated for a bounding range of conditions. The results demonstrated that the proposed 1.66-percent power uprate conditions have smaller thermally-induced bolt displacement than the original design conditions. Therefore, the baffle-barrel region thermal and structural analysis results are still bounding for the revised design conditions associated with the power uprate.

The maximum stress contributor in the upper core plate is the membrane stress resulting from the average temperature difference between the center portion of the upper core plate and the rim. The increased stress from the increased gamma heating was determined as a function of the heat generation rate increment. The fluid temperature effect due to the proposed 1.66-percent power uprate is small. The results show that the structural integrity of the upper core plate is maintained for the proposed 1.66-percent power uprate. The cumulative fatigue usage factor of the upper-core plate caused by the increase in the heat generation rates remains less than unity and the plate is structurally adequate for the proposed 1.66-percent power uprate.

As a result of these evaluations, the licensee concluded that the reactor internal components at D. C. Cook Unit 1 will be structurally adequate and can perform their intended design functions at the proposed 1.66-percent power uprate conditions.

Based on the above evaluation, the NRC staff finds that the reactor internal components at D. C. Cook Unit 1 will be structurally adequate for the proposed MUR power uprate.

3.4.2.3 Piping and Supports

For NSSS piping, the licensee stated that parameters associated with the power uprate were reviewed for impact on the existing analyses of the reactor coolant loop (RCL) piping and the pressurizer surge line including the effects of thermal stratification. The proposed power uprate NSSS performance parameters are bounded by the NSSS performance parameters from Westinghouse Topical Report WCAP-11902, Supplement 1 (Reference 13), and the existing design-basis piping analyses are still applicable for the proposed power uprate. The equipment nozzle and support loads, and the piping stresses are not affected by the proposed power uprate. The existing RCL LOCA analysis and RCL analysis with compartment pressures due to the main steam and feedwater breaks are not affected by the proposed power uprate. Since the existing piping analysis is still applicable, there are no changes in the SG or reactor coolant loop displacements, or the primary equipment nozzle and support loads due to the power uprate.

The licensee also stated that the operating temperature window of the RCL due to the power uprate is bounded by the existing range in operating temperature. With the continued applicability of the existing design transients, the impact of the power uprate is on NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," evaluation of the auxiliary spray piping, and NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," evaluation of the pressurizer surge line piping is judged as insignificant. With the continued applicability of the design transients and the insignificant changes due to the thermal analysis, the impact of the power uprate on the Auxiliary Class 1 branch nozzle displacements from the deadweight, thermal, seismic, and LOCA analyses is negligible.

The licensee further stated that the SG, RCP, reactor vessel, and pressurizer supports have been qualified for piping and component loads resulting from the Babcock and Wilcox International Replacement Steam Generator (BWIRSG) Program. The RCS supports were shown to meet the allowable stresses for all loading combinations for the D. C. Cook Unit 1 BWIRSG program loads. Since the proposed MUR power uprate would not significantly change the loads exerted on the support structures, the license concluded that the supports will continue to be qualified for the proposed 1.66-percent power uprate.

For other mechanical piping design, the licensee stated that the maximum operating pressures and temperatures will not change as a result of the proposed 1.66-percent power uprate. Therefore, existing code piping analyses are not affected by the proposed power uprate and will have no effect on qualification or adequacy of piping components. Thus, the licensee concluded that no changes are required to the mechanical piping design and code piping analyses as a result of the proposed power uprate.

On the basis of its review of the licensee’s submittals, the NRC staff concurs with the licensee’s conclusion that the existing NSSS piping and supports, primary equipment nozzles, primary equipment supports, and auxiliary lines connecting to the primary loop piping will remain in conformance with the design-basis criteria, as defined in the D. C. Cook UFSAR, and are therefore acceptable for the proposed MUR power uprate.

3.4.2.4 Control Rod Drive Mechanisms

The licensee stated that the only NSSS design parameters considered in the CRDM evaluation are hot leg temperatures and RCS pressures. There is no change to the maximum operating reactor coolant pressure of 2250 psia (which bounds operation at 2100 psia). Higher temperatures are more limiting for the CRDM structural design qualification because they result in a decrease in the margin to the allowable design stress limits. The maximum T_{hot} in the NSSS design parameters for any case determined from the MUR Power Uprate Program is 609.1 °F. Furthermore, the possible RCS operating pressure values continue to remain at either 2100 psia or 2250 psia for the proposed power uprate.

The licensee further stated that the D. C. Cook Unit 1 CRDMs were evaluated as part of the Rerating Program (Westinghouse Topical Report WCAP-11902 and Westinghouse Topical Report WCAP-11902, Supplement 1) for temperatures higher than the maximum temperature of 609.1 °F associated with the proposed power uprate. Therefore, the licensee concluded that the evaluations performed remain bounding and applicable to the proposed power uprate.

Based on the above, the NRC staff concurs with the licensee's conclusion that the current design of CRDMs continues to be in compliance with licensing-basis codes and standards under conditions expected from the proposed MUR power uprate.

3.4.2.5 Reactor Coolant Pumps and Motors

The licensee stated that the RCPs are located between the SG outlet and the reactor vessel inlet in the RCL. The maximum vessel inlet (RCP outlet) temperature is 541.7 °F for the proposed power uprate conditions. The licensee also stated that this temperature is lower than the vessel inlet temperature of 547 °F used in the previous 3425 MWt rerate evaluation, and, therefore, represents a less limiting condition. The revised pressure changes (ΔPs) and temperature changes (ΔTs) are less than those previously evaluated and are bounded for the proposed power uprate.

For the RCP motor, the licensee indicated that the limiting design parameter of the RCP motor is the horsepower loading at continuous hot and cold operation. The new hot load of 6458 horsepower (hp) for the revised operating conditions was evaluated, as it exceeds the 6000 hp nameplate rating, and found to be acceptable. The new cold load of 8057 hp for the revised operating conditions was also evaluated, as it exceeds the 7500 hp cold loop nameplate rating, and found to be acceptable. The starting temperature rise for the rotor cage winding was calculated for starting the motor under cold loop conditions with 80-percent voltage and reverse flow due to the other RCPs running at full speed. The results show that the temperatures of the rotor bars and the resistance rings will reach 230.8 °C and 38.82 °C, respectively. These temperatures do not exceed the design limits of 300 °C for the bars and 50 °C for the resistance rings. Therefore, the motor can safely start and accelerate under the worst-case conditions associated with the proposed MUR power uprate. The loads on the motor thrust bearings were also determined for the proposed uprated conditions and determined to be acceptable.

Based upon the above evaluation, NRC staff concurs with the licensee's conclusion that the RCPs, when operating at the proposed uprated conditions, would remain in compliance with the requirements of the codes and standards under which D. C. Cook Unit 1 is currently licensed.

3.4.2.6 Steam Generators

The licensee's structural evaluation focused on the critical SG components as determined by the stress ratios and fatigue usage. The evaluations were performed to confirm the acceptability of the critical primary and secondary side components when subjected to the proposed uprated operation conditions defined by the NSSS design parameters, and the applicable design transients.

The licensee stated that the Westinghouse Model 51 design SGs originally installed in D. C. Cook Unit 1 were designed and analyzed to the specifications provided in the original plant design specifications for a 3264 MWt NSSS power rating. In the year 2000, portions of the original Westinghouse Model 51 SGs were replaced with vertical shell and U-tube heat exchangers with integral moisture separating equipment. This combination was designated the BWI Series 51 replacement SG (RSG). The RSGs were designed and analyzed for both the 3264 MWt NSSS thermal power conditions and a 3600 MWt uprated NSSS thermal power condition. A comparison of the applicable MUR power uprate design transient set to the set of values evaluated for the RSG 3600 MWt operating condition was performed.

The licensee concluded that the results of the structural analyses performed on the BWI Series 51 SGs show that all SG components continue to meet ASME B&PV Code, Section III, 1989 edition, limits for the proposed 1.66-percent uprate with the RCS pressure at 2100 psia. The primary-to-secondary pressure differential remains below the design value of 1600 psid.

With regard to the flow-induced vibration, the licensee stated that the impact of the proposed power uprate on the SG tubes was evaluated based on the current design-basis analysis and included the changes in the thermal-hydraulic characteristics of the secondary-side of the SG resulting from the uprate. The effects of these changes on the fluid-elastic instability ratio and amplitudes of tube vibration due to both vortex shedding and turbulence were addressed. In addition, the potential effect of the proposed 1.66-percent power uprate on future tube wear was considered.

The licensee concluded that the analysis of the RSGs indicates that significant levels of tube vibration will not occur from either the fluid-elastic, vortex shedding, or turbulent mechanisms as a result of the proposed 1.66-percent power uprate. The projected level of tube wear as a result of vibration would be expected to remain small, and will not result in unacceptable wear.

On the basis of its review described above, the NRC staff concludes that the licensee has demonstrated the maximum stresses for the limiting SG components are within the code-allowable limits and are therefore acceptable.

3.4.2.7 Pressurizer

The licensee indicated that any changes in T_{hot} and T_{cold} are small, and are bounded by the existing pressurizer stress analysis performed for the D. C. Cook Unit 1 SG Tube Plugging (SGTP) Program conducted in 1995. No changes were made to the design transients that are applicable to the pressurizer. Therefore, the current design transients are still applicable. Additionally, there are no changes to the pressurizer nozzle loads as a result of the proposed power uprate. The licensee concluded that the revised parameters would not have any impact on the pressurizer stress and fatigue analysis and that the current evaluations remain valid.

Therefore, the licensee concluded that the pressurizer components meet the stress/fatigue analysis requirements of the ASME B&PV Code, Section III, 1965 edition through 1996 winter addenda, for plant operation at the MUR uprate conditions. The NRC staff finds that the existing design-basis analyses discussed above bound the proposed MUR power uprate, and the NRC staff agrees with the licensee's conclusion.

3.4.2.8 NSSS Auxiliary Equipment

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. The licensee performed an evaluation to determine the potential effect that the revised design conditions will have on these equipment. Only the safety injection accumulators and boron injection tanks have transients associated with them. None of the transients associated with these tanks are impacted by the proposed power uprate, therefore these tanks are not affected by the proposed MUR power uprate. Additionally, the proposed MUR power uprate would have no affect on the pressurizer relief tank or the volume control tank.

The licensee has evaluated the revised design conditions with respect to the impact on the auxiliary heat exchangers, valves, pumps, and tanks. The licensee concluded that the auxiliary equipment continues to meet the design pressure and temperature requirements, as well as the fatigue usage factors and allowable limits for which the equipment is designed. The NRC staff finds that the existing design-basis analyses, discussed above, bound the proposed MUR power uprate, and concurs with the licensee's conclusion.

3.4.2.9 Balance of Plant

The licensee has reviewed the following BOP fluid systems to assess compliance with the Westinghouse NSSS/BOP interface guidelines for the proposed power uprate:

- main steam system
- steam dump system
- condensate and feedwater system
- auxiliary feedwater system
- SG blowdown system

Various interface systems were reviewed to provide interface information that could be used in the BOP analyses.

The licensee stated that the evaluation of the interface systems indicates that, except for the steam dump valves, the design of these systems bounds operation at the proposed uprated core power level of 3304 MWt, and components within these systems are bounded by previous analyses.

The licensee stated that piping, valves, tanks, and turbines of the main steam (MS) system were evaluated to determine the overall system capability due to the power uprate. The MS system has sufficient capacity to accommodate the anticipated steam flow increase and reduced full-load operating steam pressure impacts from the proposed 1.66-percent power uprate.

The design of the steam dump valves does not meet the current licensing-basis specification to provide 40-percent steam dump capacity. These valves are currently gagged to limit valve travel to 2.75-inches. The licensee has evaluated the capability of the steam dump valves to satisfy the licensee's design-basis function for the proposed 1.66-percent power uprate and preliminarily concludes that the steam dump valves, at their current travel stop positions, have sufficient flow capacity in the current configuration for the proposed 1.66-percent power uprate. The licensee further indicated that, prior to implementation of the proposed MUR power uprate, a final steam dump valve flow capacity analysis will be completed which will determine the appropriate steam dump travel stop position. If the final steam dump valve flow capacity analysis is not adequate, then the steam dump valves' travel stop position will be changed to ensure the valves have sufficient capacity to meet the 40-percent steam dump criterion prior to implementing the proposed 1.66-percent power uprate. With this commitment by the licensee, which is confirmed in a license condition, the NRC staff considers the steam dump system acceptable for the power uprate.

3.4.2.10 High-Energy Line Break Program

The licensee stated that it reviewed the HELB Program for D. C. Cook Unit 1 in support of the proposed power uprate. The licensee determined that no HELB Program changes are required to be implemented as a result of the proposed power uprate. The activities, elements, and philosophy that currently constitute the HELB Program are not affected by the proposed power uprate. In accordance with the licensee's design change process, the design change package for installing the LEFM CheckPlus system will be evaluated against the HELB Program requirements, as required in the licensee's plant modification process. No new piping is added, no postulated break locations are changed, and no changes are made to the assumed blowdown from any currently-postulated breaks; therefore, there is no impact on the current D. C. Cook Unit 1 HELB analysis.

The licensee further stated that the proposed 1.66-percent power uprate is bounded by the existing HELB analysis-of-record. These analyses are consistent with the provisions of Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirement," and the D. C. Cook Unit 1 current licensing basis, as indicated in Unit 1 License Amendment No. 249. No specific TS or operating procedure changes were identified by the HELB Program review for the proposed power uprate. Thus, the licensee concluded that no changes are required to the HELB Program. The NRC staff finds that the existing design-basis analyses discussed above bound the proposed MUR power uprate, and the NRC staff concurs with the licensee's conclusion.

3.4.2.11 Motor-Operated Valve Program

The licensee stated that it reviewed the design-basis calculations for feedwater, main steam, and plant cooling systems (i.e., CCW, essential service water, nonessential service water) to determine potential impacts on the MOV Program. The limiting (bounding) differential pressures were based on system capacities and setpoints (e.g., SG safety valve setpoints, pump shutoff head), which will not change due to the proposed 1.66-percent power uprate.

The licensee concluded that the proposed 1.66-percent power uprate does not challenge the capability of valves in the MOV program to satisfy their design functions and that no changes are required to the MOV program as a result of the proposed power uprate. The NRC staff finds that the existing design-basis analyses discussed above bound the proposed MUR power uprate, and the NRC staff concurs with the licensee's conclusion.

3.4.2.12 Air and Hydraulic Operated Valve Program

The licensee stated that a review of heat balances that reflect the effect of the proposed 1.66-percent power uprate on system design function parameters indicates that there is no impact on the ability of air and hydraulic-operated valves (AHOVs) to perform their design function in the systems affected by the proposed MUR power uprate. No additional AHOVs were identified as being impacted by the proposed power uprate. As a result, the licensee concluded that no changes will be required to the AHOV Program due to the proposed MUR power uprate. The NRC staff finds that the existing design-basis analyses discussed above bound the proposed MUR power uprate, and the NRC staff concurs with the licensee's conclusion.

3.4.3 Summary

The NRC staff reviewed the proposed MUR power uprate at D. C. Cook Unit 1 as it relates to the effects on the structural and pressure boundary integrity of the NSSS and the BOP systems and components. Results of the NRC staff evaluation show that the NSSS and BOP piping, components, and supports operating at the proposed 1.66-percent power uprate are bounded by the existing analyses, and continue to satisfy the licensing codes of record and the original design basis. Therefore, the NRC staff finds the proposed 1.66-percent power uprate acceptable with respect to the performance NSSS and BOP piping, systems, and components.

3.5 Dose Consequences Analysis

3.5.1 Regulatory Evaluation

A revision to 10 CFR Part 50, Appendix K, effective July 31, 2000, allowed licensees to use a power uncertainty less than 2 percent in design-basis LOCA analyses, based on the use of state-of-the art feedwater flow measurement devices that provide for a more accurate calculation of power. Appendix K did not originally require the power measurement uncertainty be determined, but instead required a 2-percent margin. The revision allows licensees to justify a smaller margin for power measurement uncertainty based on power level instrumentation error.

As with any license amendment, the licensee must show that the plant continues to meet dose limit criteria given in 10 CFR Part 100 or 10 CFR 50.67, as applicable, 10 CFR Part 50, Appendix A, GDC-19. The NRC staff uses applicable sections of Standard Review Plan (NUREG-0800) Chapter 15 for design-basis accidents (DBAs) to review such amendments. Licensees requesting MUR power uprates should identify existing DBA analyses of record which bound plant operation at the proposed power level. For any DBA for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.5.2 Technical Evaluation

The NRC staff reviewed the impact of the proposed changes on DBA radiological analyses, as documented in Chapter 14 of the D. C. Cook UFSAR. In its application, the licensee stated that the current radiological analyses of record for D. C. Cook Unit 1 were unaffected by the proposed 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 June 28, 2002, application, the NRC staff verified that the existing D. C. Cook Unit 1 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.

The NRC staff has reviewed and approved D. C. Cook License Amendment Nos. 271 (Unit 1) and 252 (Unit 2), dated November 14, 2002. These amendments made a selective implementation of an alternative source term in accordance with 10 CFR 50.67. The licensee assumed a core power level of 102 percent of 3588 MWt or 3660 MWt for the revised analysis, which bounds the conditions for the proposed 1.66-percent power uprate for D. C. Cook Unit 1. The NRC staff found the analysis to be acceptable, as stated in the safety evaluation associated with License Amendment Nos. 271 and 252.

3.5.4 Summary

Based upon the above discussion, the NRC staff concludes that for a 1.66-percent increase to the rated core power level, the radiological consequences of DBAs would continue to be bounded by the doses estimated in previously performed and accepted analyses. The NRC staff finds that there is reasonable assurance that radiological consequences of DBAs meet the dose limits given in 10 CFR Part 100, 10 CFR Part 50, and GDC-19, and conform to applicable dose acceptance criteria given in Standard Review Plan Chapter 15. Therefore, the NRC staff finds that the proposed MUR power uprate is acceptable with respect to the radiological consequences of DBAs.

3.6 Component Integrity

3.6.1 Regulatory Evaluation

The NRC staff used the following documents in performing its evaluation of the licensee's proposed 1.66-percent MUR power uprate: (1) NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," (2) Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," and (3) Electrical Power Research Institute (EPRI) report NSAC-202L-R2, "Recommendation for Effective Flow-Accelerated Corrosion Program."

3.6.2 Technical Evaluation

3.6.2.1 Structural Integrity Evaluation

The licensee's structural evaluation focused on critical SG components, as determined by stress ratios and fatigue usage. The primary side components that were evaluated included the SG tubes, SG tube plugs, tube-to-tubesheet weld, tubesheet-to-shell junction, and divider plate.

The secondary side components included the feedwater nozzle, secondary manway bolts and shell penetration, and the steam nozzle. The evaluations were performed to confirm the acceptability of the critical primary and secondary side components when subjected to the proposed uprated conditions and the applicable design transients.

The licensee concluded that the results of the analyses show that all SG components continue to meet ASME B&PV Code, Section III, limits for the proposed 1.66-percent power uprate conditions with the RCS pressure at 2100 psia. The primary-to-secondary pressure differential remains below the design value of 1600 psid. For operation with the RCS at 2250 psia, the primary-to-secondary pressure differential remains below the design value of 1600 psid, provided the secondary side steam pressure is limited to 679 psia. In order to address this limitation, the licensee committed to documenting this limitation in the D. C. Cook Engineering Control Procedure, “Precautions, Limitations, and Setpoints - Unit 1.” In addition, the 679 psia limitation will also be incorporated into the UFSAR prior to implementation of the proposed license amendment.

The NRC staff finds that 10 CFR 50.59 will provide appropriate control over secondary side steam pressure, and therefore, the NRC staff concludes that the proposed 1.66-percent power uprate will not have significant impact on SG structural integrity.

3.6.2.2 Tube Vibration and Wear and Other Modes of Tube Degradation

The potential effects of the proposed MUR power uprate on SG tube wear and other modes of SG tube degradation (e.g., axial and/or circumferential cracking, etc.) were evaluated by the licensee.

The evaluation of the effects of the proposed 1.66-percent power uprate on SG tube wear were based on the current design basis analysis and included changes in the thermal-hydraulic characteristics of the secondary-side of the SG resulting from the power uprate. The licensee indicated that fretting wear damage parameter would increase as a result of the power uprate; however, it would still remain below the previously established limit, which was determined to be acceptable. Therefore, the licensee determined that although there would be an increase in the level of tube wear due to the power uprate, the projected level of tube wear would be expected to remain small and within pre-established limits.

The licensee evaluated the effects of the power uprate on modes of SG tube degradation other than wear. The licensee concluded that the proposed 1.66-percent power uprate is not expected to have a significant impact on corrosive SG tube degradation and that no new modes of SG tube degradation are introduced as a result of the power uprate.

Because none of the potential degradation mechanisms are significantly affected by the proposed 1.66-percent power uprate conditions, the licensee concluded that the required frequency of inspection is also not significantly affected by the planned power uprate.

The NRC staff agrees with the licensee’s analysis set forth above, and concludes, therefore, that the proposed power uprate conditions will not significantly affect SG tube degradation.

3.6.2.2 Regulatory Guide 1.121 Analysis

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 30-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.

3.6.2.4 Flow-Accelerated Corrosion

Flow-accelerated corrosion (FAC) is a corrosion mechanism causing wall thinning of high energy pipes in the power conversion system which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety systems, the licensee has a program for predicting, inspecting, and repairing or replacing the components whose wall thinning exceeds the values required for their safe operation. The program uses the EPRI-developed CHECWORKS computer code for predicting thinning of the walls in the components subjected to FAC. In its application, the licensee indicated that a review of heat balances that reflect the effect of the proposed power uprate on FAC-related parameters indicates that no additional systems, piping, or other components need to be added to the FAC Program. The FAC Program will direct changes to the CHECWORKS input parameters due to changes to system operating pressures, temperatures, and flowrates. Based on review of the revised heat balance calculation. the licensee concluded that these changes are minimal and the power uprate is not expected to significantly affect the current wear predictions of the CHECWORKS software. The NRC staff considers this licensee's action adequate for ensuring integrity of the high-energy pipes.

3.6.3 Summary

The NRC staff has evaluated the licensee's programs for managing the effects of a 1.66-percent MUR power uprate on the corrosion wear rates due to FAC and on the performance of the SGs at D. C. Cook Unit 1. On the basis of its evaluation, the NRC staff finds that in both cases, the effects of the proposed MUR power uprate would be very small and the licensee has procedures in place to account for them. In addition, the NRC staff concludes that the power uprate condition will not adversely impact the piping systems managed by the FAC program. Also, the NRC staff concludes that the power uprate will not have a significant impact on its SG tube structural and leakage integrity. Therefore, the NRC staff finds that the proposed MUR power uprate is acceptable with respect to component integrity.

3.7 Structural Integrity and Metallurgy Section

3.7.1 Regulatory Evaluation

This section identifies those NRC regulations, RGs, NRC staff positions, and other guidance documents used by the NRC staff in the review of the applicable sections of the licensee's application for the proposed 1.66-percent MUR power uprate.

3.7.1.1 10 CFR Part 50 Appendix G - Pressure-Temperature Limits

The Commission's requirements for generating pressure and temperature (P-T) limit curves are given in Appendix G to 10 CFR Part 50. Section IV.A.2. of 10 CFR Part 50, Appendix G, requires that the P-T limits for the reactor vessel (RV) of a light-water reactor be at least as conservative as the P-T limit generation methods of Appendix G to Section XI of the ASME B&PV Code. For materials in the beltline region of the vessel, the rule requires the calculations of P-T limits to take into account the effects of neutron irradiation on the reference temperatures for nil ductility (i.e., RT_{NDT} values) for the materials used to fabricate the RV and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RV material surveillance program. Additional guidance for the NRC staff's review of RV P-T limit curves is provided in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," SRP Section 5.3.2, "Pressure-Temperature Limits," and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." The NRC's regulatory requirements related to the establishment of a facility's RV surveillance capsule program and withdrawal schedule are given in Appendix H to 10 CFR Part 50, which also references the guidance in American Society for Materials and Testing Standard Practice E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." SRP Section 5.3.1, "Reactor Vessel Materials," also applies.

3.7.1.2 10 CFR Part 50 Appendix G - Upper Shelf Energy

The Commission's requirements for upper-shelf energy (USE) are given in Appendix G to 10 CFR Part 50. Section IV.A.2. of 10 CFR Part 50, Appendix G, requires that the RV beltline materials have a minimum USE value of 75 ft-lb in the unirradiated condition, and maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analytical engineering analyses (i.e., through equivalent margins analyses), lower values of USE would provide acceptable margins of safety against fractures equivalent to those required by Appendix G of Section XI to the ASME B&PV Code. For materials in the beltline region of the vessel, the rule requires USE calculations to account for the effects of neutron irradiation on materials and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RV material surveillance program. Additional guidance for the NRC staff's review of USE analyses is provided in RG 1.99, Revision 2, SRP Section 5.3.2, and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." Appendix K to Section XI of the ASME B&PV Code and RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb.," may also be used as guidance when USE equivalent margins analyses are required.

3.7.1.3 10 CFR 50.61 - Pressurized Thermal Shock Events

The Commission's requirements for protecting the RVs of PWRs against pressurized thermal shock (PTS) events are stated in 10 CFR 50.61. The rule requires RV materials made of carbon or low-alloy steel materials to meet a maximum screening criterion for nil-ductility reference temperatures (i.e., RT_{PTS} values). The rule's screening criteria are 270 °F for axial weld materials and base metal materials (i.e., plates or forging materials) and 300 °F for circumferential weld materials. The rule provides methods for calculating these RT_{PTS} values. For materials in the beltline region of the vessel, the rule requires the calculations to take into account the effects of neutron irradiation on the RT_{PTS} values for the materials and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RV material surveillance program.

3.7.1.4 Leak-Before-Break Analyses

Application of leak-before-break (LBB) is used to address the elimination of dynamic effects associated with pipe rupture from a facility's licensing basis. This consideration is related to the requirements in GDC 4, or, for those plants licensed prior to the development of Appendix A to 10 CFR Part 50, similar requirements which were imposed during the proceeding on the facility's operating license. In evaluating the technical basis for a licensee's application of LBB, the NRC staff references NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," and draft Standard Review Plan Section 3.6.3, "Leak Before Break Evaluation Procedures."

3.7.1.5 Structural Integrity of CRDM Nozzles

Maintenance of the leakage and structural integrity of the CRDM nozzles is directly related to requirements in Appendix A to 10 CFR Part 50, GDC 14, or, for those plants licensed prior to the development of Appendix A to 10 CFR Part 50, similar requirements which were imposed during the proceeding on the facility's operating license. Guidance regarding an acceptable evaluation of the potential for primary water stress-corrosion cracking (and the performance of appropriate inservice inspections related to the evaluation) was provided in NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," and the NRC staff's acceptance of responses from the industry and individual licensees to this bulletin.

3.7.1.6 Structural Integrity of Reactor Vessel Internals

Maintenance of the structural integrity of the reactor pressure vessel (RPV) internals is required in order to demonstrate that the functional requirements of the RPV internals are met. These functional requirements include core support and ECCS performance aspects. As such, the structural integrity of the RPV internals is linked to regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry. Additional guidance regarding the evaluation of the structural integrity of RPV internals may be found in SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," ASME B&PV Code, Sections III and XI, or other standards which were used in the NRC's review of the original licensing basis of a particular facility.

3.7.1.7 Structural Integrity of Reactor Coolant Pump Flywheels

Maintenance of the structural integrity of RCP flywheels is important to assure the RCP flywheels can maintain a continuous coast down from 120 percent of the design rotor speed for the RCP impellers and to preclude the potential for missile generation as a result of their failure. The evaluation of RCP flywheel integrity is related to Appendix A to 10 CFR Part 50, GDC-1, and GDC-4, or, for those plants licensed prior to the development of Appendix A to 10 CFR Part 50, similar requirements which were imposed during the proceeding on the facility's operating license.

3.7.2 Technical Evaluation

3.7.2.1 Projected Neutron Fluence Values and P-T Limits

The licensee's request for a 1.66-percent MUR power uprate at the end of Cycle 18 would increase the end-of-license vessel neutron fluence. The licensee estimated that the current 32 EFPY P-T limits needed to be redesignated for 28.4 EFPYs. The NRC staff indicated that fluence calculations for P-T curve adjustment should follow the guidance in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." In addition, the licensee stated that it had discovered a new limiting element (a peripheral weld with Heat No. 1P3571) with a chemistry factor of 218.6 °F. The licensee estimated that the combined effect of the new critical element and the revised fluence lowered the P-T limits to 18.62 EFPYs.

In its June 28, 2002, application, the licensee requested TS changes to modify the vessel P-T applicability limits. The TS changes became necessary due to the increased vessel irradiation rate associated with the proposed MUR power uprate and the identification of a new limiting element in the vessel beltline. In its June 28, 2002, application, the licensee proposed to adjust the P-T limit curves from 32 EFPYs of operation to 28.4 EFPYs. However, the identification by the licensee of a new limiting element in the vessel beltline, combined with the results of revised fluence calculations, required a readjustment of the P-T applicability limits. Therefore, in its supplemental letter dated October 15, 2002, the licensee revised the change proposed in the June 28, 2002, application.

The NRC staff's review is limited to the vessel fluence calculation and its application in the adjustment of the P-T curves. The fluence calculation methodology was described in the licensee's supplemental letter dated October 15, 2002. In addition, the projected neutron sources reflect the proposed power uprate to be implemented for Cycle 18. The calculation results were compared to the results of capsules T, X, Y, and U, which further validated the methodology. Based on its review, the NRC staff finds that the methodology adheres to the guidance of RG 1.190, and therefore, the results of the fluence calculation are acceptable.

The current P-T curves were calculated for a peak vessel fluence of 1.41×10^{19} n/cm^2 for 32 EFPYs based on the analysis and measurement results of capsule U (Reference 12). The revised 32 EFPY fluence is 1.587×10^{19} n/cm^2. The new controlling element has a chemistry factor of 218.6 °F. The licensee back-calculated the applicability of the current P-T curves with the new fluence. The result is 18.62 EFPYs. The NRC staff performed an independent calculation and concurs with the licensee's numerical value.

Figures 3.4-2 and 3.4-3 of TS 3.4, “Reactor Coolant System Pressure,” would be modified to reflect the revised P-T curve applicability limits. The NRC staff finds these TSs changes acceptable.

Based on the considerations discussed above, the NRC staff concludes that changes to the notes on TS Figures 3.4-2 and 3.4-3 are acceptable for the established period of applicability at the proposed MUR power uprate conditions. In addition, a condition has been added to the license to require the licensee to recalculate the PT curves prior to implementation of this license amendment.

3.7.2.2 Effect on the USE and PTS Assessments

Using the limiting uprated neutron fluence value (i.e., 0.953×10^{19} n/cm^2) for the 1/4T location of the D. C. Cook Unit 1 RV, the NRC staff performed independent calculations of the end-of-licence USE values for the RV beltline materials. The limiting USE material (i.e., the circumferential weld fabricated from heat 1P3571) in the D. C. Cook Unit 1 RV has an end-of-licence USE of 60.9 ft-lb. Based on these results, the NRC staff concludes that the D. C. Cook Unit 1 beltline materials will continue to have adequate values of USE and ductility under the proposed MUR power uprate conditions.

The NRC staff performed an independent assessment of the impact of the proposed uprated neutron fluence values on the PTS assessment for D. C. Cook Unit 1 in order to confirm that the RT_{PTS} values for the beltline materials in the RV would comply with the requirements of 10 CFR 50.61. Using the limiting uprated neutron fluence value (i.e., 1.587×10^{19} n/cm^2) for D. C. Cook Unit 1, the NRC staff calculated an RT_{PTS} value of 232 °F for the limiting circumferential weld material (material heat 1P3571). This meets the NRC staff’s screening criterion of 300 °F for RV circumferential welds materials. The NRC staff calculated an RT_{PTS} value of 222 °F for the limiting axial weld material (material heat 13252/12008). This meets the NRC staff’s screening criterion of 270 °F for RV base metal and axial weld materials. Based on these results, the NRC staff concludes that the D. C. Cook Unit 1 beltline materials will continue to have adequate protection against PTS events under the proposed MUR power uprate conditions.

3.7.2.3 Structural Integrity of Other Class 1 Reactor Coolant System Components

The licensee previously reanalyzed the design transients and accidents for D. C. Cook Unit 1 as part of a license amendment request for operating under reduced RCS temperature and pressure conditions (i.e., reduced temperature and pressure rerate program). The NRC staff approved the reduced temperature and pressure rerate program in D. C. Cook License Amendment No. 126, dated June 9, 1989.

In its safety assessment for the proposed 1.66-percent MUR power uprate, the licensee compared the key plant temperature and pressure parameters for the RCS under the proposed uprated power conditions to those previously approved for the reduced temperature and pressure rerate program. The table below provides the key RCS power, temperature, and pressure parameters for both the proposed 1.66-percent MUR power uprate and the reduced temperature and pressure rerate program.

The licensee's safety assessment for License Amendment No. 126 included an evaluation of the reduced temperature and pressure conditions on the stress intensity and fatigue usage factors for the RV, CRDM housings, and RV internals. In the NRC staff's safety evaluation associated with License Amendment No. 126, the NRC staff concluded that the stress intensities for the RV, CRDM housings, and RV internals would continue to be acceptable under the reduced temperature and pressure rerate program.

In its June 28, 2002, application, the licensee indicated that (1) the vibrational and stress levels would not change for the RV, RV internals, and CRDM housing components under the proposed uprated power conditions, and (2) the proposed uprated conditions would not impact the loadings previously analyzed in the licencee's LBB analysis for the reactor coolant piping.

Key Operating Parameter Values for the D. C. Cook Unit 1 RCS Under the 1.66% Power Uprate and the Reduced Temperature and Pressure Rerate Program While Operating at a High T_{avg} for the RCS

Parameter	1.66% Power Uprate	Reduced T-P Rerate Program
Reactor Thermal Power, MWt	3315	3588
NSSS Power, MWt	3327	3600
Thermal Design Flow, Loop gallons per minute (gpm)	83200	88500
Reactor Coolant Pressure, psia	2100-2250	2100-2250
T_{hot}, °F	609.1	615.2
T_{avg}, °F	575.4	581.3
SG Outlet Temperature, °F	541.5	547.1
Steam Pressure, psia	765	820
Feedwater Temperature, °F	437.4	449

For operation of D. C. Cook Unit 1 at its maximum T_{avg} (i.e., at the high T_{avg}), the table above indicates that the operating power levels, pressures, and temperatures for the D. C. Cook Unit 1 RCS under the proposed 1.66-percent power uprate are bounded by those that were analyzed and approved by the NRC staff for the reduced pressure and temperature rerated design. Since the operating power levels, temperatures, and pressures for the RCS under the proposed 1.66-percent power uprate are bounded by those previously approved by the NRC staff for the reduced pressure and temperature rerate program, the NRC staff concludes that the loadings and stress intensity levels for the RV internals, CRDM housings and nozzles, RCS piping, and RCP flywheels are bounded by those previously approved by the NRC staff in D. C. Cook License Amendment No. 126 . Therefore, the NRC also concludes that the proposed 1.66-percent power uprate will not affect the structural integrity analyses previously performed by the licensee for these components (i.e., limit load and stress-corrosion crack growth analyses for RV internals and CRDM housings and nozzles, LBB analysis for the RCS main loop piping, and thermal fatigue crack growth analysis for the RCP flywheels).

3.7.3 Summary

The NRC staff has reviewed the licensee's application to increase the rated core thermal power for D. C. Cook Unit 1 by 1.66-percent and has evaluated the impact that the proposed uprated power conditions will have on the structural integrity assessments for the RV, RV internals, CRDM housings and nozzles, RCS piping, and RCP flywheels. The NRC staff has determined that, with the exception of the P-T limits, the proposed license amendment will not significantly impact the following structural integrity assessments for the RCS: (1) PTS assessment for the RV, (2) USE assessment for the RV, (3) structural integrity assessments for the RV internals, RCS outlet nozzle safe-end, and CRDM nozzles and housings, (4) LBB analysis for the RCS main loop piping, and (5) thermal fatigue crack growth analysis for the RCP flywheels.

The NRC staff has determined that the proposed 1.66-percent power uprate would not impact the P-T limits for the RCS, as currently approved in the D. C. Cook Unit 1 TSs. To address this impact, the licensee has proposed to reduced the expiration of the current P-T limits from 32 EFPYs to 18.6 EFPYs and has also included the proposed change to the P-T limits as part of its June 28, 2002, application. The NRC staff has reviewed the licensee's proposed reduction in the expiration term for the current P-T limit curves and has determined the proposed change appropriately takes into account both the increase in the neutron fluence level for the RV as a result of the uprate and a change in the limiting material for the D. C. Cook Unit 1 RV. Based on these changes, the NRC staff confirmed that 18.6 EFPYs is an acceptable expiration term for current P-T limit curves in the TSs. Therefore, the NRC staff concludes that the licensee has appropriately addressed the impact that the proposed 1.66-percent power uprate will have on the structural integrity for the RCS components.

3.8 Human Performance

3.8.1 Regulatory Evaluation

The human performance evaluation is to ensure that operator performance would not be adversely affected as a result of changes needed for the proposed 1.66-percent MUR power uprate. The following were used to review the proposed action: (1) SRP Chapter 13.2.1, "Reactor Operator Training," (2) SRP Chapter 13.5.2.1, "Operating and Emergency Operating Procedures," and (3) SRP Chapter 18, "Human Factors Engineering."

3.8.2 Technical Evaluation

3.8.2.1 Operator Actions

The licensee indicated that the proposed MUR power uprate is not expected to have any significant affect 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 1 will not have an adverse affect on operator actions.

3.8.2.2 Emergency and Abnormal Operating Procedures

The licensee indicated that there are currently no EOPs that reference use of the LEFM. Specific procedures within the EOP program may require review and revision based upon the proposed 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 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 levels. 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. 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.

3.8.2.3 Control Room Controls, Displays and Alarms

The licensee stated that the notification of the operators of the LEFM CheckPlus system condition will be through the 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 (ARM) 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 ARM in a manner consistent with the design modification process. This will be finalized prior to implementing the proposed MUR power uprate. The NRC staff finds this acceptable.

3.8.2.4 Control Room Plant Reference Simulator

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.

3.8.2.5 Operator Training Program

The installation of the LEFM and implementation of the proposed 1.66-percent MUR power uprate would require procedure and training changes. Actions would be added to the appropriate operating procedures and the development of an Administrative Technical Requirement (ATR) in the event the LEFM system becomes unavailable. Operations training concerning the use of the LEFM, the associated procedures, and the ATR changes will be completed prior to implementation of the MUR power uprate. All this information will be updated in a manner consistent with other plant modifications and license amendments. The NRC staff finds this acceptable.

3.8.3 Summary

The NRC staff finds that the licensee has satisfactorily addressed the areas discussed above, and that the proposed 1.66-percent MUR power uprate should not adversely affect human performance. Therefore, the NRC staff finds the proposed 1.66-percent power uprate acceptable with respect to human performance.

3.9 Containment/Fire Protection Systems

3.9.1 Containment Performance Evaluation

3.9.1.1 Feedwater and Steamline Break Mass and Energy Releases

The licensing-basis safety analyses related to the feedwater and steamline break mass and energy releases were evaluated to determine the effect of a 1.66-percent power uprate. The evaluation determined that the NSSS design parameters for the proposed MUR power uprate remain unchanged or are bounded by the current safety analyses.

The evaluations performed include:

Short-Term Feedwater Line Break Mass and Energy Releases
Short-Term Steamline Break Mass and Energy Releases Inside Containment
Long-Term Steamline Break Mass and Energy Releases Inside Containment

These evaluations also include the SG enclosure and fan/accumulator room subcompartment analyses and long-term containment analyses, which were demonstrated to be unaffected by the proposed MUR power uprate.

The NRC staff's reviewed the licensee's evaluations and found that each evaluation was bounded by current evaluations which were previously reviewed and approved by the NRC staff. Therefore the NRC staff finds that the proposed MUR power uprate will not effect the feedwater or steamline break mass and energy releases.

3.9.1.2 Post-LOCA Containment Hydrogen Generation

The D. C. Cook Unit 1 post-LOCA containment hydrogen generation analysis of record calculates hydrogen generation rates using a core thermal power of 3411 MWt. Therefore, the current analyses for post-LOCA containment hydrogen generation bounds the proposed MUR power uprate. The NRC staff finds this acceptable.

3.9.1.3 Long-Term LOCA Mass and Energy Release Analysis

The current analyses of record was performed at an assumed Unit 1 core power level of 3481 MWt. The NRC staff previously reviewed and approved these analyses. Therefore, the current licensing basis remains bounding for the proposed 1.66-percent power uprate. The NRC staff finds this acceptable.

3.9.1.4 Short-Term LOCA Mass and Energy Release Analyses

The analyses are conducted in two phases. In the first phase, the mass and energy releases from the postulated break are determined prior to evaluating the containment response. In the second phase, the analyses involve evaluating the subcompartment containment response to the releases.

Since the critical portion of this event lasts for less than 3 seconds, the effect of reactor power is not significant. The analyses inputs having the potential to change due to the proposed 1.66-percent power uprate are the initial RCS fluid temperatures. The critical flow correlation used in the mass and energy releases for these analyses provide an increase in the mass and energy release for a lower fluid temperature. For the current analyses of record, an RCS hot leg (vessel outlet temperature) of 579.1 ºF (minus 5 ºF for uncertainty (574.1 ºF)) was used and a cold leg (vessel/core inlet temperature) of 511.7 ºF (minus 5 ºF for uncertainty (506.7 ºF)) was used. Both were conservatively bounded low for short-term considerations. The proposed power uprate values of 588.2 ºF for the hot leg temperature and 519.2 ºF for the cold leg temperature are both bounded by the analyses of record. Therefore, the current licensing basis remains bounding for the proposed 1.66-percent power uprate. The NRC staff has previously reviewed and approved these analyses. Therefore, the current licensing basis remains bounding for the proposed 1.66-percent power uprate. The NRC staff finds this acceptable.

3.9.2 Fire Protection Systems

3.9.2.1 Regulatory Basis

The Fire Protection Program at D. C. Cook is based on NRC criteria, National Fire Protection Association standards, Institute of Electrical and Electronic Engineers standards, and other industry codes. The staff has previously found that the Fire Protection Program at D. C. Cook Unit 1 complies with the requirements of 10 CFR 50.48 and 10 CFR Part 50, Appendix R.

The objective of the Fire Protection Program is to minimize both the probability and consequences of postulated fires. The probability and consequences of such fires are minimized by a combination of design features, procedural controls, and personnel training, including a well-trained fire brigade.

3.9.3.2 Fire Protection Evaluation

The activities and elements currently in place to implement the Fire Protection Program are not affected by the proposed power uprate. The post-fire safe shutdown aspect of the Fire Protection Program is in place to meet the requirements of 10 CFR Part 50, Appendix R. The addition of the LEFM CheckPlus system will not change the circuit separation nor adversely impact any systems credited for an Appendix R safe shutdown. No new cables for credited components will be added or deleted. The safe shutdown analysis methodology and acceptance criteria previously developed to demonstrate compliance with 10 CFR Part 50, Appendix R, would remain unchanged.

No changes are required to the Fire Protection or Appendix R/Safe Shutdown Programs as a result of the proposed MUR power uprate. In accordance with the licensee’s design change process, the design change package for installing the LEFM CheckPlus system will be evaluated against provisions of the Fire Protection/Appendix R Program, as set forth in the D. C. Cook Unit 1 plant modification process.

Therefore, the NRC staff finds the proposed power uprate acceptable with respect to fire protection requirements.

3.9.4 Summary

Based on the review of the licensee's analysis as set forth above, the NRC staff finds that the results are reasonable, conservative, and therefore acceptable with respect to the containment and fire protection systems for the proposed MUR power uprate.

4.0 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.

5.0 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 change the surveillance requirements. The NRC 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 (67 FR 48219). 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.

6.0 CONCLUSION

The NRC 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.

7.0 REFERENCES

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

3. Westinghouse Licensing Topical Report WCAP-8567-P-A, (Proprietary), "Improved Thermal Design Procedure," July 1975.

4. American National Standards Institute/Instrument Society of America ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation," February 29, 2000.

5. Letter from J. F. Stang, NRC to R. P. Powers, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments [234 (Unit 1) and 217 (Unit 2)] RE: SBLOCA [Small-Break LOCA] Scenarios," December 13, 1999.

6. Letter from J. F. Stang, NRC, to M. P. Alexich, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit 1 - Issuance of Amendment [126 (Unit 1)] RE: Reduced Temperature and Pressure Conditions," June 9, 1989.

7. Letter from J. F. Stang, NRC, to A. C. Bakken III, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments [271 (Unit 1) and 252 (Unit 2)] RE: Control Room Habitability and Generic Letter 99-02 Requirements," November 14, 2002.

8. Westinghouse Letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.

9. J. B. Hickman, NRC, to E. E. Fitzpatrick, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments [214 (Unit 1) and 199 (Unit 2)] RE: Increased Steam Generator Plugging Limit," March 13, 1997.

10. Westinghouse Licensing Topical Reports WCAP-9272-P-A (Proprietary) and WCAP-9273-NP-A (Non-proprietary) "Westinghouse Reload Safety Evaluation Methodology," July 1985.

11. Westinghouse Licensing Topical Reports WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-proprietary), "Revised Thermal Design Procedure," April 1989.

12. Westinghouse Licensing Topical Report WCAP-12483, “Analysis of Capsule U from the American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program,” January 1990.

13. Westinghouse Licensing Topical Report WCAP-19902, Supplement 1, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988.

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