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Cook Nuclear Plant
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INDIANA
MICHIGAN
POWER

June 28, 2002
AEP:NRC:2900
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License Amendment Request for Appendix K Measurement Uncertainty Recapture – Power Uprate Request

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, proposes to amend Facility Operating License (OL) DPR-58, including Appendix A, Technical Specifications (TS). CNP Unit 1 is presently licensed for a core power rating of 3250 megawatts thermal (MWt). Based on the implementation of more accurate feedwater flow measurement instrumentation and power calorimetric uncertainty values, approval is sought to increase the licensed core power by 1.66 percent, to 3304 MWt.

The feedwater flow measurement system to be installed at CNP Unit 1 is a Leading Edge Flow Meter (LEFM™) CheckPlus™ ultrasonic multi-path transit time flowmeter. The design of this advanced flow measurement system was submitted by the manufacturer, Caldon Incorporated, in topical reports that were reviewed and approved by the Nuclear Regulatory Commission (NRC).

Enclosure 1 provides an oath and affirmation affidavit statement. Enclosure 2 provides a detailed description and safety analysis to support the proposed changes, including the 10 CFR 50.92(c) evaluation, which concludes that no significant hazard is involved, and the environmental assessment. Attachment 1 provides marked-up OL/TS pages for CNP Unit 1. Attachment 2 provides the proposed OL/TS pages with the changes incorporated. Attachment 3 provides

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the information delineated in Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," to establish the appropriate scope, structure, and level of detail for this Appendix K Measurement Uncertainty Recapture (MUR). Attachment 4 contains a list of new regulatory commitments made in this letter.

I&M recognizes that various WCAPs that are part of the CNP Unit 1 licensing basis may have included explicit references to their use of "102% of licensed core power levels." These WCAPs will not be revised to reflect this requested power uprate, because it is understood that the statements provided in these WCAPs refer to the previously-required 2 percent Appendix K margin and the currently licensed power level.

I&M requests approval of this request by October 1, 2002, to support a mid-cycle power uprate. I&M requests a 60-day implementation period.

No previous submittals affect OL/TS pages that are submitted in this request. If any future submittals affect these pages, I&M will coordinate changes to the pages with the NRC Project Manager to ensure proper OL/TS page control when the associated license amendment requests are approved.

Should you have any questions, please contact Mr. Gordon P. Arent, Manager of Regulatory Affairs, at (616) 697-5553.

Sincerely,

J. E. Pollock
Site Vice President

NH/jen

Enclosures:

1. Notarized oath and affirmation statement
2. Evaluation of proposed changes to Facility Operating License DPR-58

Attachments:

1. Marked-Up Proposed Operating License/Technical Specification Changes
2. Proposed Operating License/Technical Specification Pages
3. Summary of Measurement Uncertainty Recapture Evaluations Following Guidance Provided in Regulatory Issue Summary 2002-03
4. Regulatory Commitments

c: K. D. Curry
J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

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

Indiana Michigan Power Company

J E Pollock

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 28 DAY OF JUNE, 2002

Jennifer L Kernosky
Notary Public

My Commission Expires 5/26/05

JENNIFER L KERNOSKY
Notary Public, Berrien County, Michigan
My Commission Expires May 26, 2005

Appendix K Measurement Uncertainty Recapture
Power Uprate Request

1.0 Description

Indiana Michigan Power Company (I&M) proposes to amend Facility Operating License (OL) DPR-58, including Appendix A, Technical Specifications (TS), for Donald C. Cook Nuclear Plant (CNP) Unit 1. CNP Unit 1 is presently licensed for a core power rating of 3250 megawatts thermal (MWt). Through the use of more accurate feedwater flow measurement instrumentation, approval is sought to increase the licensed core power by 1.66 percent, to 3304 MWt.

2.0 Proposed Changes

The proposed license amendment would revise the CNP Unit 1 OL and TS to increase licensed power level to 3304 MWt, or 1.66 percent greater than the current level of 3250 MWt. The proposed changes, which are indicated on the marked-up pages in Attachment 1, are described below:

1. Paragraph 2.C.(1) in Facility Operating License DPR-58 is revised 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 is revised 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," are revised to limit Indicated T_{avg} at RTP (T' for Overtemperature ΔT; T'' for Overpressure ΔP) to less than or equal to 574°F and 562.1°F, respectively.

4. TS Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation," is revised to reflect the maximum allowed power for operation with inoperable main steam safety valves (MSSVs). With one inoperable MSSV per loop, the power reduction is 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 is 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 are revised to reflect these changes.

5. TS Figure 3.4-2 "Reactor Coolant System Pressure – Temperature Limits Versus 60°F/hr Rate Criticality Limit and Hydrostatic Test Limit," and TS Figure 3.4-3 "Reactor Coolant System Pressure – Temperature Limits Versus Cooldown Rates," are revised to reflect the new limit of applicability of 28.4 EFPY versus 32 EFPY. The TS Bases for LCO 3/4.4.9,

"Reactor Coolant System – Pressure/Temperature Limits," are revised to reflect this change.

6. TS Table 4.4-5, "Reactor Vessel Material Irradiation Surveillance Schedule," is revised by changing the removal interval for Capsule S from "32 EFPY" to "Standby." The TS Bases for LCO 3/4.4.9, "Reactor Coolant System – Pressure/Temperature Limits," are revised to reflect this change.

3.0 Background

CNP Unit 1 is presently licensed for a core power rating of 3250 MWt. Based on the implementation of more accurate feedwater flow measurement instrumentation and associated power calorimetric uncertainty values, approval is sought to increase the licensed core power by 1.66 percent, to 3304 MWt.

The 1.66 percent core power uprate for CNP Unit 1 (Measurement Uncertainty Recapture (MUR) Uprate Program) is based on recapturing measurement uncertainty currently included in the analytical margin. The analytical margin was originally required for emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in Title 10 of the Code of Federal Regulations (CFR) Part 50 (10 CFR 50), Appendix K, "ECCS Evaluation Models." In June 2000, the Nuclear Regulatory Commission (NRC) approved a change to the 10 CFR 50, Appendix K, requirements to provide licensees with the option of maintaining the 2 percent power margin between the licensed core power level and the assumed core power level for ECCS evaluations, or apply a reduced margin to the ECCS evaluations. The proposed alternative to recapture margin for ECCS evaluation has been demonstrated to account for uncertainties due to a reduction in power level instrumentation error.

I&M is currently installing a more accurate feedwater flow measurement system manufactured by Caldon, Incorporated (Caldon). The NRC has approved Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ flow measurement systems, similar to the system to be installed at CNP Unit 1, in a Safety Evaluation Report dated December 20, 2001 (Reference 7). The Caldon instrumentation provides the capability to determine core power level with a power measurement uncertainty of approximately 0.31 percent. Based on the use of the Caldon instrumentation and CNP Unit 1 power calorimetric uncertainty values, including retention of a 0.03 percent design margin beyond the uncertainty of 0.31 percent, I&M proposes to use a reduced margin for ECCS evaluation pursuant to the revised requirements of 10 CFR 50, Appendix K, to achieve an increase of 1.66 percent in the licensed core power level using current NRC-approved methodologies.

The impact of the MUR Uprate Program has been evaluated on the nuclear steam supply system (NSSS) and balance of plant (BOP) systems, components, and safety analyses. Attachment 3 summarizes these evaluations, analyses, and conclusions, andprovides the information delineated in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (Reference 1) to facilitate NRC review of Appendix K MUR power uprate license amendment requests.

4.0 Technical Analysis

I&M has evaluated the impact of the proposed power uprate on NSSS systems and components, BOP systems, and safety analyses.

Attachment 3 summarizes the results of the comprehensive engineering review performed to evaluate the increase in the licensed core power from 3250 MWt to 3304 MWt. Results of this analysis are provided in a format consistent with the regulatory guidance provided in RIS 2002-03.

The evaluation for the CNP Unit 1 MUR Uprate Program was implemented consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant" (Reference 2). The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects, including the broad categories that must be addressed. These include the NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel, as well as the interfaces between the NSSS and BOP systems. The methodology includes the use of well-defined analysis input assumptions and parameter values, currently approved analytical techniques, and currently applicable licensing criteria and standards. The results of I&M's analyses and evaluations demonstrate that applicable acceptance criteria will continue to be met following the implementation of the proposed 1.66 percent power uprate.

I&M has reviewed the proposed MUR Uprate Program to determine if these changes will result in an increase in the plant's risk profile. This review found that the installation of the LEFM CheckPlus system in the feedwater system would not affect the CNP Probabilistic Risk Assessment (PRA) model, because flow instrumentation is below the level of detail of the plant's PRA model. Setpoint rescaling required for implementation of the 1.66 percent power increase will not impact the risk profile. Furthermore, the MUR uprated core thermal power (3304 MWt) is bounded by the rated thermal power assumed in the PRA Success Criteria (Reference 6). Therefore, the proposed MUR Uprate Project will not affect the CNP Unit 1 risk profile.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

I&M has evaluated whether a significant hazards consideration is involved with the proposed amendment to increase the licensed core power level from 3250 MWt to 3304 MWt through improved feedwater flow measurement accuracy by using more accurate ultrasonic flow measurement instrumentation. The I&M evaluation has been performed by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No

Probability of Occurrence of an Accident Previously Evaluated -

In support of this measurement uncertainty recapture power uprate, a comprehensive evaluation was performed for nuclear steam supply system (NSSS) and balance of plant (BOP) components and analyses that could be affected by this change. A power calorimetric uncertainty calculation was performed, and the effect of increasing plant power by 1.66 percent on the plant's design and licensing basis was evaluated. The result of these evaluations is that all plant components will continue to be capable of performing their design function at an uprated core power of 3304 megawatts thermal (MWt). In addition, an evaluation of the accident analyses demonstrates that applicable analysis acceptance criteria continue to be met. No accident initiators are affected by this uprate and no challenges to any plant safety barriers are created by this change.

Consequences of an Accident Previously Evaluated -

This change does not affect the release paths, the frequency of release, or the source term for release for any accidents previously evaluated in the Updated Final Safety Analysis Report. Structures, systems, and components (SSC) required to mitigate transients remain capable of performing their design functions, and thus were found acceptable. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at a core power of 3304 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. The installation of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system has been analyzed, and failures of this system will have no adverse effect on any safety-related system or any SSCs required for transient mitigation. SSCs previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system.

This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at a core power level of 3304 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at a core power of 3304 MWt. Credible malfunctions continue to be bounded by the current accident analysis of record or re-analysis demonstrates that applicable acceptance criteria continue to be met.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No

The margins of safety associated with this Measurement Uncertainty Recapture Uprate Program are those pertaining to core power. This includes those associated with the fuel cladding, Reactor Coolant System (RCS) pressure boundary, and containment barriers. A comprehensive engineering review was performed to evaluate the 1.66 percent increase in the licensed core power from 3250 MWt to 3304 MWt. The 1.66 percent increase required that revised NSSS design thermal and hydraulic parameters be established, which then served as the basis for all of the NSSS analyses and evaluations. This engineering review concluded that no design transient modifications are required to accommodate the revised NSSS design conditions. NSSS systems and components were evaluated and it was concluded that the NSSS equipment has sufficient margin to accommodate the 1.66 percent power uprate. NSSS accident analyses were either evaluated or revised for the 1.66 percent power uprate. In all cases the evaluations and re-analyses demonstrate that the applicable analyses acceptance criteria continue to be met. As such, the margins of safety continue to be bounded by the current analyses of record for this change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

5.2 Applicable Regulatory Requirements/Criteria

The proposed MUR Uprate Program has been reviewed to ensure compliance with all applicable regulatory requirements and criteria. Specifically, the requirements of 10 CFR 50.36, 50.46, 50.48, 50.49, 50.61, 50.62, 50.63, 50.71(e), 10 CFR 50, Appendix K, and the CNP Unit 1 OL, including Appendix A, "Technical Specifications," were reviewed. The TS changes that are summarized in Section 2, "Proposed Changes," of this enclosure are required to support plant operation following implementation of the MUR Uprate Program. These changes meet the four criteria for TS LCOs specified in 10 CFR 50.36(c)(2)(ii), while allowing continued TS compliance at the uprated conditions. This MUR Uprate Program will be implemented in accordance with the revised requirements of 10 CFR 50, Appendix K, which allow recapture of measurement uncertainty that was previously included in the analytical margin. The re-allocation of measurement uncertainty will not change the analysis or reporting requirements or the acceptance criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." Changes to the UFSAR that result from the MUR Uprate Program will be submitted to the NRC as specified in 10 CFR 50.71(e). Finally, the NRC's basis for approval of the analyses performed to demonstrate compliance with NRC regulations promulgated after issuance of the OL (i.e., fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63)) were reviewed to ensure I&M's approved methodology for complying with these regulations would not be impacted by the MUR Uprate Program. Based on the impact reviews that were conducted in support of the MUR Uprate Program, and the OL/TS changes that were identified as required for implementation of this change, it is concluded that I&M's compliance with applicable regulatory requirements will be maintained.

The WCAP-10263 methodology has been successfully used as the basis for power uprate projects for several Westinghouse pressurized water reactor (PWR) units, which have also implemented Caldon LEFM systems. These include Watts Bar Nuclear Plant, Unit 1 (Reference 3), Comanche Peak Steam Electric Station, Units 1 and 2 (Reference 4), and Beaver Valley Power Station, Units 1 and 2 (Reference 5). The scope and level of detail of this CNP Unit 1 license amendment request are commensurate with that provided in the approved amendments for the referenced plants.

I&M has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

Based upon the determinations that the acceptance criteria of WCAP-10263 are met by the proposed power uprate and that there are no significant hazards considerations associated with the proposed uprate, I&M concludes that the proposed change will not endanger the health and safety of the public. Similar amendment requests have been accepted for other nuclear power plants.

In conclusion, based on the considerations discussed above, (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.

6.0 Environmental Considerations

The environmental review, pursuant to 10 CFR 51.22(b), determined that no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment. In accordance with the guidance provided in RIS 2002-03, the environmental considerations pertaining to this license amendment request are addressed in Attachment 3, Section VII.5, "Environmental Review."

7.0 References

1. Letter from W. D. Beckner, NRC, "NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

2. WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," dated January 1983

3. Letter from R. E. Martin, NRC, to J. A. Scalice, Tennessee Valley Authority, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendments Regarding Increase of Reactor Power to 3459 Megawatts Thermal (TAC No. MA9152)," dated January 19, 2001

4. Letter from D. H. Jaffe, NRC, to C. L. Terry, TXU Electric, "Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 – Issuance of Amendments Re: Increase in Allowable Thermal Power to 3458 MWt and Deletion of Texas Municipal Power Agency from the Operating Licenses (TAC Nos. MB1625 and MB1626)," dated October 12, 2001

5. Letter from L. J. Burkhart, NRC, to L. W. Myers, FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves (TAC Nos. MB0996, MB0997, and MB2557)," dated September 24, 2001

6. Letter from E. E. Fitzpatrick, I&M, to T. E. Murley, NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Individual Plant Examination Submittal Response to Generic Letter 88-20," dated May 1, 1992

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

ATTACHMENT 1 to AEP:NRC:2900

FACILITY OPERATING LICENSE AND
TECHNICAL SPECIFICATION PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 1

Operating License Page 2 of 5

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B 3/4 7-1

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any by-product, source and special nuclear material as sealed neutron sources for reactor start-up, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any by-product, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such by-product and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

* 2.C(1) Maximum Power Level

Amendment No. 18

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 ~~3250~~3304 megawatts (thermal).

* (2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment 269 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

* The following Amendments have been issued to paragraph 2.C(2):
Nos. 12, 13, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 28, 29, 30, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100, 101, 102, 103, 104, 105, 106, 107,108,109, 110, 111, 112, 113, 114, 115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 132, 133, 134, 135, 136, 137, 138, 139, 140, 141, 142, 143, 144, 145, 146, 147, 148, 149, 150, 151, 152, 153, 154, 155, 156, 157, 158, 159, 160, 161, 162, 163, 164, 165, 166, 167, 168, 169, 170, 171, 172, 173, 174, 175, 176, 177, 178, 179, 180, 181, 182, 183, 184, 185, 186, 187, 188, 189, 190, 191, 192, 193, 194, 195, 196, 197, 198, 199, 200, 201, 202, 203, 204, 205, 206, 207, 208, 209, 210, 211, 212, 213, 214, 215, 216, 217, 218, 219, 220, 221, 222, 223, 224, 225, 226, 228, 229, 230, 231, 232, 233, 234, 235, 236, 237, 238, 239, 240, 241, 242, 243, 244, 245, 246, 247, 248, 249, 250, 251, 252, 253, 254, 257, 259, 260, 261, 262, 263, 264, 266, 267, 268, and 269.

* Amendment No. 18 superceded Amendment No. 14.

1.0 DEFINITIONS

DEFINED TERMS -

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3250~~ 3304 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2 \left[\frac{1+\tau_1 s}{1+\tau_2 s}\right] \; (T-T') + K_3 \, (P-P') - f_1 \, (\Delta I)]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER (≤~~576.3~~ 574.0 °F)
	P	=	Pressurizer pressure, psig
	P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} $\tau_1 =$ 22 secs. $\tau_2 =$ 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2(\Delta I)\right]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ ~~563.0~~ 562.1 °F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constants utilized in the rate lag controller for T_{avg} $\tau_3 =$ 10 secs.
	S	=	Laplace transform operator
	$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

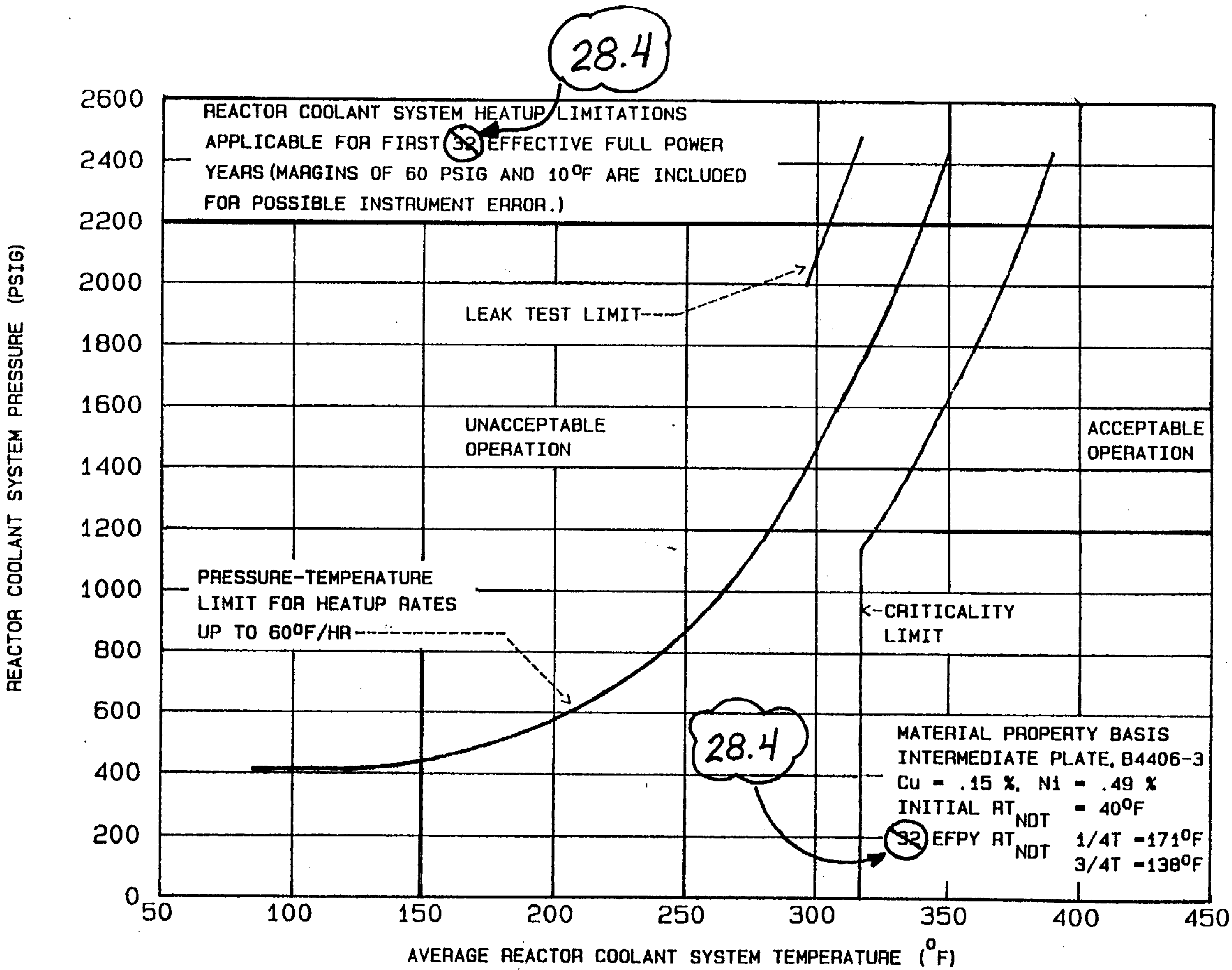

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE – TEMPERATURE LIMITS VERSUS 60°F/HR RATE CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

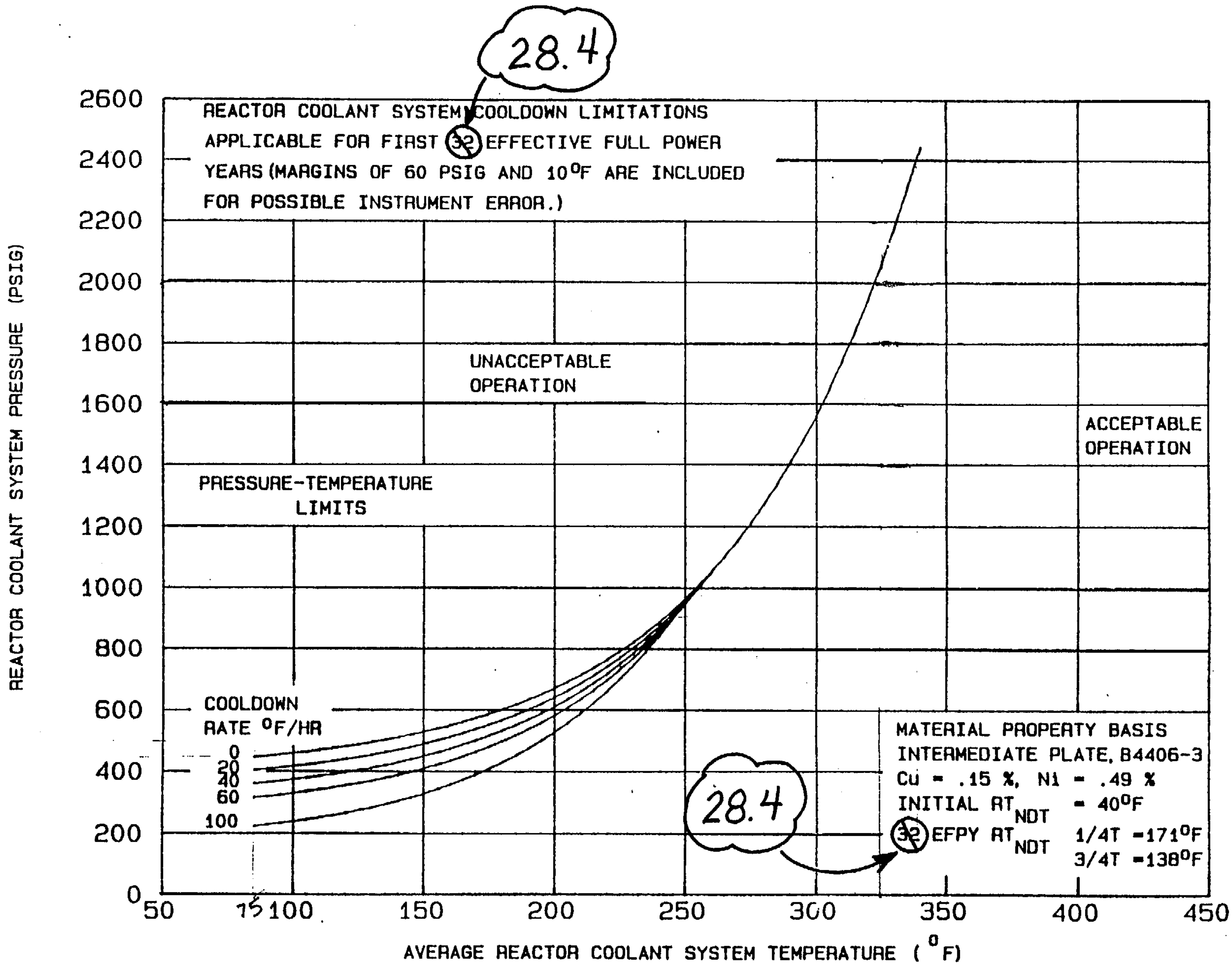

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS COOLDOWM RATES

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
Capsule T	1.25 EFPY
Capsule X	3 EFPY
Capsule Y	5 EFPY
Capsule U	9 EFPY
Capsule S	~~32 EFPY~~ Standby
Capsules V, W, Z	Standby

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~65.1~~ 63.8
2	~~46.5~~ 45.5
3	~~28.0~~ 27.4

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in thevessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of ~~32~~ 28.4 EFPY.

Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, and the copper and nickel content of the material must be predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include the adjusted RT_{NDT} at the end of ~~32~~ 28.4 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The ~~32~~ 28.4 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shell plate, B4406-3, being the limiting material with a copper and nickel content of .15% and .49%, respectively.

2. ~~The fluence values contained in Table 6-14 of Westinghouse's WCAP-12483 report, "Analysis of Capsule U From the American Electric Power Company D.C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program," dated January 1990.~~
The applicability date of the heatup and cooldown curves was revised by the D. C. Cook Unit 1 Measurement Uncertainty Recapture Uprate Program to reflect the increased fluence values over those found in Table 6-14 of WCAP-12483.

3. Figure 1, NRC Regulatory Guide 1.99, Revision 2

The shift in RT_{NDT} of the reactor vessel material has been established by removing and evaluating the material surveillance capsules installed near the inside wall of the reactor vessel in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed ~~until Capsule S is to be removed after 32 EFPY (EOL)~~. Capsule S, V, W, and Z will remain in the reactor vessel, and will be removed to address industry reactor embrittlement concerns, if required.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constituted an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The safety valve is OPERABLE with a lift setting of ±3% about the nominal value. However, the safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance. The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is approximately ~~121~~ 118 percent of the total secondary steam flow of ~~14,120,000~~ 14,540,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

$$Hi\Phi = (100/Q)\frac{(4\, w_s h_{fg})}{K}$$

where:

$Hi\Phi$ = Safety Analysis power range high neutron flux setpoint in percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt

K = Conversion factor, 947.82 $\frac{(Btu/Sec)}{Mwt}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate in Btu/lbm

4 = Number of loops in plant

The values calculated from this algorithm are then adjusted lower for use in Technical Specification 3.7.1.1 to account for instrument and channel uncertainties by 9%. This reduces the maximum plant operating power level so that it is lower than the reactor protection system setpoint by an appropriate operating margin.

ATTACHMENT 2 to AEP:NRC:2900

PROPOSED FACILITY OPERATING LICENSE AND
TECHNICAL SPECIFICATION PAGES

REVISED PAGES
UNIT 1

Operating License Page 2 of 5

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B 3/4 7-1

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any by-product, source and special nuclear material as sealed neutron sources for reactor start-up, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any by-product, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such by-product and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

* 2.C(1) Maximum Power Level

Amendment No. 18

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).

* (2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment 269 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

* The following Amendments have been issued to paragraph 2.C(2):
Nos. 12, 13, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 28, 29, 30, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100, 101, 102, 103, 104, 105, 106, 107,108,109, 110, 111, 112, 113, 114, 115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 132, 133, 134, 135, 136, 137, 138, 139, 140, 141, 142, 143, 144, 145, 146, 147, 148, 149, 150, 151, 152, 153, 154, 155, 156, 157, 158, 159, 160, 161, 162, 163, 164, 165, 166, 167, 168, 169, 170, 171, 172, 173, 174, 175, 176, 177, 178, 179, 180, 181, 182, 183, 184, 185, 186, 187, 188, 189, 190, 191, 192, 193, 194, 195, 196, 197, 198, 199, 200, 201, 202, 203, 204, 205, 206, 207, 208, 209, 210, 211, 212, 213, 214, 215, 216, 217, 218, 219, 220, 221, 222, 223, 224, 225, 226, 228, 229, 230, 231, 232, 233, 234, 235, 236, 237, 238, 239, 240, 241, 242, 243, 244, 245, 246, 247, 248, 249, 250, 251, 252, 253, 254, 257, 259, 260, 261, 262, 263, 264, 266, 267, 268, and 269.

* Amendment No. 18 superceded Amendment No. 14.

1.0 DEFINITIONS

DEFINED TERMS -

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3304 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2 \left[\frac{1+\tau_1 s}{1+\tau_2 s}\right] (T-T') + K_3 (P-P') - f_1 (\Delta I)]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T′	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 574.0 °F)
	P	=	Pressurizer pressure, psig
	P′	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1= 22 secs. τ_2 = 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2(\Delta I)\right]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 562.1 °F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constants utilized in the rate lag controller for T_{avg} $\tau_3 =$ 10 secs.
	S	=	Laplace transform operator
	$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

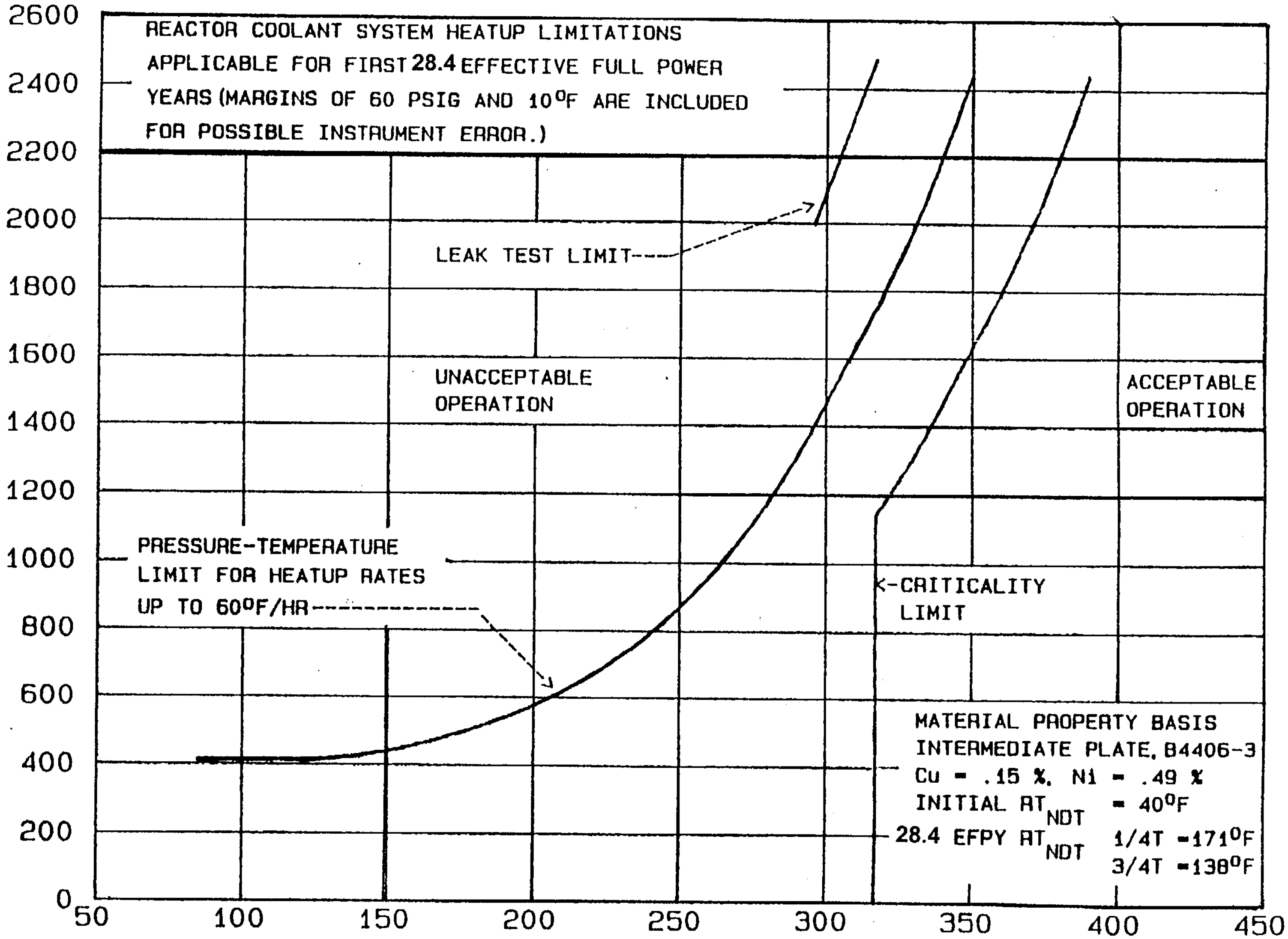

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS 60°F/HR RATE
CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

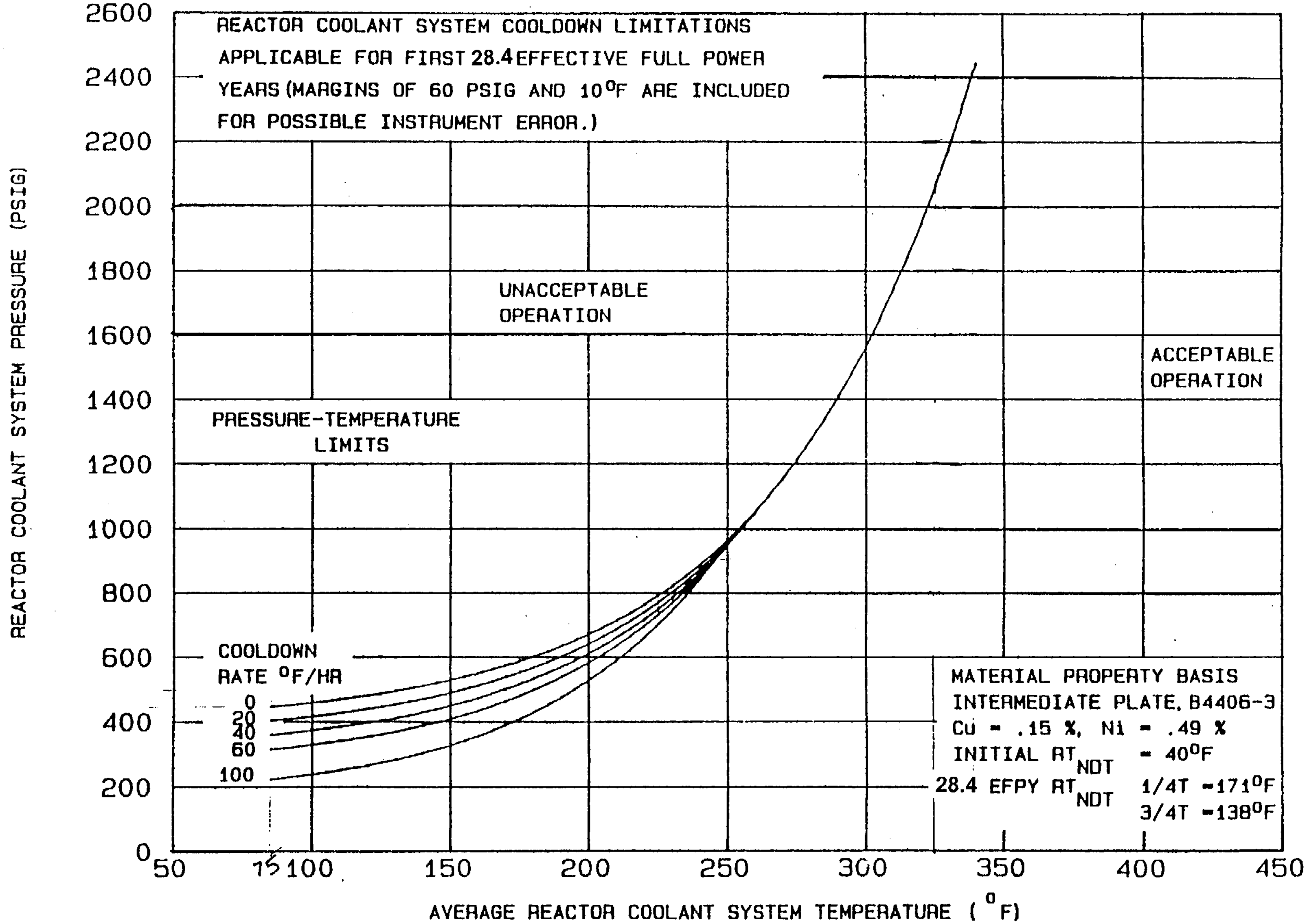

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS COOLDOWN RATES

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
Capsule T	1.25 EFPY
Capsule X	3 EFPY
Capsule Y	5 EFPY
Capsule U	9 EFPY
Capsule S	Standby
Capsules V, W, Z	Standby

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	63.8
2	45.5
3	27.4

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in thevessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 28.4 EFPY.

Reactor operation and resultant fast neutron ($E > 1$ Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, and the copper and nickel content of the material must be predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include the adjusted RT_{NDT} at the end of 28.4 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 28.4 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shell plate, B4406-3, being the limiting material with a copper and nickel content of .15% and .49%, respectively.

2. The applicability date of the heatup and cooldown curves was revised by the D. C. Cook Unit 1 Measurement Uncertainty Recapture Uprate Program to reflect the increased fluence values over those found in Table 6-14 of WCAP-12483.

3. Figure 1, NRC Regulatory Guide 1.99, Revision 2

The shift in RT_{NDT} of the reactor vessel material has been established by removing and evaluating the material surveillance capsules installed near the inside wall of the reactor vessel in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed. Capsule S, V, W, and Z will remain in the reactor vessel, and will be removed to address industry reactor embrittlement concerns, if required.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constituted an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The safety valve is OPERABLE with a lift setting of ±3% about the nominal value. However, the safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance. The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is approximately 118 percent of the total secondary steam flow of 14,540,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

$$Hi\Phi = (100/Q)\frac{(4\, w_s h_{fg})}{K}$$

where:

$Hi\Phi$ = Safety Analysis power range high neutron flux setpoint in percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt

K = Conversion factor, 947.82 $\frac{(\text{Btu/Sec})}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate in Btu/lbm

4 = Number of loops in plant

The values calculated from this algorithm are then adjusted lower for use in Technical Specification 3.7.1.1 to account for instrument and channel uncertainties by 9%. This reduces the maximum plant operating power level so that it is lower than the reactor protection system setpoint by an appropriate operating margin.

ATTACHMENT 3 to AEP:NRC:2900

SUMMARY OF
MEASUREMENT UNCERTAINTY RECAPTURE EVALUATION
FOLLOWING GUIDANCE PROVIDED IN
REGULATORY ISSUE SUMMARY 2002-03

TABLE OF CONTENTS

LIST OF TABLES

LIST OF ACRONYMS

AC	alternating current
AFWS	auxiliary feedwater system
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
AOP	abnormal operating procedure
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BEF	best-estimate flow
BIT	boron injection tank
BOP	balance of plant
BVPS	Beaver Valley Power Station
BWI	Babcock and Wilcox International
CCW	component cooling water
CFR	Code of Federal Regulations
CNP	Donald C. Cook Nuclear Plant
CRDM	control rod drive mechanism
CTS	containment spray system
CST	condensate storage tank
CVCS	chemical and volume control system
CW	circulating water
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EFPY	effective full-power year
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ERG	emergency response guideline
ESF	engineered safety feature
FCV	feedwater control valves
FFM	feedwater flow measurement
FIV	feedwater isolation valve
GPM	gallons per minute
HFP	hot-full power
HZP	hot-zero power
I&C	instrumentation and control
I&M	Indiana Michigan Power Company

LBB	leak-before-break
LBLOCA	large-break loss-of-coolant accident
LEFM	leading edge flow meter
LOCA	loss-of-coolant accident
LTOP	low temperature overpressure protection
MSIV	main steam isolation valve
MS	main steam
MSSV	main steam safety valve
MUR	Measurement Uncertainty Recapture
MWt	megawatts thermal
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OFA	optimized fuel assembly
OL	Operating License
OPΔT	overpower delta T
OSG	original steam generator
OTΔT	overtemperature delta T
PORV	power-operated relief valve
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RIS	Regulatory Issue Summary
RPS	reactor protection system
RSG	replacement steam generator
RTD	resistance temperature detector
RTDP	revised thermal design procedure
RWFS	rod withdrawal from subcritical
SBLOCA	small-break loss-of-coolant accident
SER	Safety Evaluation Report
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SSE	safe shutdown earthquake
T/H	thermal and hydraulic
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
USE	upper shelf energy

Summary of Measurement Uncertainty Recapture Evaluator Following Guidance Provided in RIS 2002-03

Introduction

Indiana Michigan Power Company (I&M) proposes to amend Facility Operating License (OL) DPR-58, including Appendix A, Technical Specifications (TS), for Donald C. Cook Nuclear Plant (CNP) Unit 1. CNP Unit 1 is presently licensed for a core power rating of 3250 megawatts thermal (MWt). Through the use of more accurate feedwater flow measurement instrumentation, approval is sought to increase the licensed core power by 1.66 percent, to 3304 MWt. The proposed 1.66 percent power uprate is based on eliminating unnecessary analytical margin originally required of emergency core cooling system (ECCS) evaluation models developed in accordance with the requirements set forth in 10 CFR 50, Appendix K, "ECCS Evaluation Models."

In June 2000, the Nuclear Regulatory Commission (NRC) approved a change to the 10 CFR 50, Appendix K, requirements to provide licensees with the option of maintaining the 2 percent power margin between the licensed power level and the assumed power level for ECCS evaluations, or apply a reduced margin to the ECCS evaluations. The proposed alternative to recapture margin for ECCS evaluation has been demonstrated to account for uncertainties due to a reduction in power level instrumentation error. I&M is currently installing Caldon, Incorporated (Caldon) Leading Edge Flow Meter (LEFM™) CheckPlus™ instrumentation with an installed power measurement uncertainty of less than 0.31 percent. Based on the implementation of the LEFM CheckPlus instrumentation and CNP specific power calorimetric uncertainties, I&M proposes to reduce the licensed power uncertainty required by 10 CFR 50, Appendix K, and increase the CNP Unit 1 licensed power level by 1.66 percent using NRC-approved methodologies.

I&M has evaluated the impact of the proposed power uprate on nuclear steam supply system (NSSS) systems and components, balance of plant (BOP) systems, safety analyses, and programs. The results of I&M's analyses and evaluations, which demonstrate that applicable acceptance criteria will continue to be met, are summarized in this attachment. Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (Reference 1) was used to establish the appropriate scope, structure, and level of detail presented in this license amendment request.

Approach for Increasing the Plant Power Level

The CNP Unit 1 Measurement Uncertainty Recapture (MUR) Uprate Program has been developed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant" (Reference 2). This methodology has been successfully used as the basis for power uprate projects for several Westinghouse pressurized water reactor (PWR) units, including Watts Bar Nuclear Plant, Unit 1 (Reference 3), Comanche Peak Steam Electric Station, Units 1 and 2 (Reference 4), and Beaver Valley Power Station, Units 1 and 2 (Reference 5).

The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects, including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel, as well as the interfaces between the NSSS and BOP systems. The methodology includes the use of well-defined analysis input assumptions/parameter values, use of currently approved analytical techniques, and use of currently applicable licensing criteria and standards.

Overview of this Attachment

A comprehensive engineering review program consistent with the WCAP-10263 methodology has been performed for CNP Unit 1 to evaluate the increase in the licensed core power from 3250 MWt to 3304 MWt. Section I of this attachment describes the Caldon LEFM CheckPlus system that will be implemented to provide more accurate feedwater flow measurement. Section II provides the results of the accident and transient analyses for which the existing analyses of record clearly bound plant operation at the uprated power level. Section III summarizes those accidents and transient analyses that required re-analysis to produce analytical results that bound the uprated power level. Table 2, "MUR Power Uprate Impact on CNP Unit 1 Accident/Transient Analyses," summarizes the accident and transient analyses for CNP Unit 1, and documents whether or not each analysis of record bounds plant operation at the uprated power level proposed by the MUR Uprate Program. Table 2 also indicates whether the summary of the evaluation/analysis is addressed in Section II or Section III of this attachment.

Sections IV and V of this attachment address the impact of the power uprate on the structural integrity of major plant components and on electrical equipment. Section VI addresses the effect of the power uprate on major plant systems and Section VII addreses the identification and evaluation of operator actions, modifications, environmental, and programs impacts resulting from the 1.66 percent power uprate. Table 1, "System and Program Review Summary," summarizes the results of the evaluations that were performed on the CNP Unit 1 NSSS and BOP components and plant programs. The results of the analyses and evaluations addressed in Sections II through VII demonstrate that all acceptance criteria continue to be met.

Section VIII discusses the required changes to the CNP Unit 1 TS, protection system settings, and emergency system settings.

TABLE 1 – SYSTEM AND PROGRAM REVIEW SUMMARY				
SYSTEM/ COMPONENT/ PROGRAM	**PARAMETERS WITH MUR UPRATE POTENTIAL IMPACT**	**COMPONENTS IMPACTED**	**BOUNDED BY EXISTING DESIGN/ ANALYSES?**	**REFERENCES** (Report Section Number)
STEAM/POWER SYSTEMS				
Condensate	Flowrate (Increase) System Pressure (Decrease) System Temperature (Increase)	Hotwell Pumps	Yes	• VI.2.2, VII.3
		Condensate Booster Pumps	Yes	• VI.2.2, VII.3
		Piping	Yes	• VI.2.2, VII.3
		Valves and Miscellaneous Components	Yes	• VI.2.2, VII.3
		Low Pressure Feedwater Heaters	Yes	• VI.2.4, VII.3
Feedwater	Flowrate (Increase) System Pressure (Decrease) System Temperature (Increase)	Feedwater Pumps	Yes	• VI.2.2, VII.3
		Feedwater Regulating Valves	Yes	• VI.2.2, VII.3
		Feedwater Isolation Valves	Yes	• VI.2.2, VII.3
		High Pressure Feedwater Heaters	Yes	• VI.2.2, VII.3
		Heater Drain Pumps	Yes	• VI.2.2, VII.3
Auxiliary Feedwater System (AFWS) and Condensate Storage Tank	Required flow to steam generators when normal feedwater not available	Turbine Driven Auxiliary Feedwater Pump	Yes	• VI.2.3, VII.3
		Motor Driven Auxiliary Feedwater Pumps	Yes	• VI.2.3, VII.3
		Condensate Storage Tank	Yes	• VI.2.3, VII.3
Main Steam	Steam Flow (Increase) System Pressure (Decrease)	Steam Dump	No	• VI.2.1, VII.3, VII.4
		Main Feed Pump Turbines	Yes	• VI.2.1, VII.3, VII.4
		Steam Generator Blowdown	Yes	• VI.2.1, VII.3, VII.4

TABLE 1 – SYSTEM AND PROGRAM REVIEW SUMMARY				
SYSTEM/ COMPONENT/ PROGRAM	**PARAMETERS WITH MUR UPRATE POTENTIAL IMPACT**	**COMPONENTS IMPACTED**	**BOUNDED BY EXISTING DESIGN/ ANALYSES?**	**REFERENCES** (Report Section Number)
Main Steam	Steam Flow (Increase) System Pressure (Decrease)	Steam Generator Safety Valves	Yes	• VI.2.1, VII.3, VII.4
		Power Operated Relief Valves	Yes	• VI.2.1, VII.3, VII.4
		Main Steam Isolation Valves	Yes	• VI.2.1, VII.3, VII.4
		MSIV Bypass Valves	Yes	• VI.2.1, VII.3, VII.4
		Auxiliary Steam System	Yes	• VI.2.1, VII.3, VII.4
		Auxiliary Feed Pump Turbine	Yes	• VI.2.1, VII.3, VII.4
		Main Turbine	Yes	• VI.2.1, VII.3, VII.4
		Main Turbine Stop Valves	Yes	• VI.2.1, VII.3, VII.4
		Main Turbine Control Valves	Yes	• VI.2.1, VII.3, VII.4
		Moisture Separator Reheaters	Yes	• VI.2.1, VII.3, VII.4
Feedwater Heaters and Drains	Steam and Feedwater Flow (Increase)	Feedwater Heaters	Yes	• VI.2.4, VII.3
	System Pressure (Decrease)	Feedwater Heater Drains	Yes	• VI.2.4, VII.3
	System Temperature (Increase)	Feedwater Heater Level Control Valves	Yes	• VI.2.4, VII.3
COOLING/SUPPORT SYSTEMS				
Component Cooling Water (CCW)	Cooldown Flow to Residual Heat Removal (RHR) Heat Exchangers (Increase)	RHR System	Yes	• VI.3.1, VII.3
Essential Service Water (ESW)	No Changes	None	Yes	• VI.3.2, VII.3
Non-Essential Service Water (NESW)	Required flow to steam generator (SG) Blowdown Components (Increase)	SG Blowdown Components	Yes	• VI.3.3, VII.3
Turbine Auxiliary Cooling Water (TACW)	Required flow to Iso-phase Bus Duct Cooling (Increase)	Iso-phase Bus Duct	Yes**	• VI.3.4, VII.3, VII.4

TABLE 1 – SYSTEM AND PROGRAM REVIEW SUMMARY				
SYSTEM/ COMPONENT/ PROGRAM	**PARAMETERS WITH MUR UPRATE POTENTIAL IMPACT**	**COMPONENTS IMPACTED**	**BOUNDED BY EXISTING DESIGN/ ANALYSES?**	**REFERENCES** (Report Section Number)
Emergency Diesel Generator Cooling	No Changes	None	Yes	• VI.3.5, VII.3
Circulating Water (CW)	Condenser Operating Pressure (Increase)	Main Condenser	Yes	• VI.3.6, VII.3
Spent Fuel Pool Cooling (SFPC)	Spent Fuel Pit Decay Heatload (Increase)	SFPC Pumps and Heat Exchangers	Yes	• VI.3.7, VII.3
Auxiliary Building Vent System	Heat load in ECCS Rooms/ Cubicles (Negligible)	ESFVS Exhaust Air Fans	Yes	• VI.4, VII.3
ESF Ventilation System	Post-LOCA Hydrogen Generation (No Change)	Hydrogen Skimmer/ Recirculating Fans	Yes	• VI.4, VII.3
Containment Ventilation System	Heat load in Upper/Lower Containment (Bounded)	Upper/Lower Containment Recirculating Fan Coil Units	Yes	• VI.4, VII.3
AFWS Motor and Turbine Pump Room Coolers	Heat load in AFWS Pump Rooms (No Change)	Room A/C Units	Yes	• VI.4, VII.3
Iso-phase Bus Duct Cooling System	Heat load in Bus Duct Enclosures (Increase)	Bus Duct Fan Coil Units	Yes**	• VI.4, VII.3, VII.4
ELECTRICAL SYSTEMS				
Turbine/Generator	Generator Output (MVA Increase)	Generator	Yes	• V, VII.3
Iso-phase Bus	Main Generator Current (Increase)	Iso-phase Bus	Yes	• V, VII.3, VII.4
Main Transformer	Transformer Output (MVA Increase)	Transformers	Yes	• V, VII.3
Switchyard	Switchyard Current (Increase)	Circuit Breakers	Yes	• V, VII.3
Offsite Power Feeders	Tie Line Current (Increase)	Tie Line Current Rating	Yes	• V, VII.3
Grid Stability	N/A	Main Generator Impedance	Yes	• V, VII.3
Emergency Diesel Generators	No Changes	No Changes	Yes	• V, VII.3
Auxiliary Transformers	Transformer Output Increase	Transformer MVA	Yes	• V, VII.3
Station Service Transformers	Transformer Output Increase	Transformer MVA	Yes	• V, VII.3

TABLE 1 – SYSTEM AND PROGRAM REVIEW SUMMARY				
SYSTEM/ COMPONENT/ PROGRAM	**PARAMETERS WITH MUR UPRATE POTENTIAL IMPACT**	**COMPONENTS IMPACTED**	**BOUNDED BY EXISTING DESIGN/ ANALYSES?**	**REFERENCES** (Report Section Number)
Protective Relay Settings	Generator Current Increase	Grid Fault Protection	Yes	• V, VII.3, VII.4
Electrical Distribution System	Bus Current Increase	4160 Bus, Breakers, Cables	Yes	• V, VII.3, VII.4
NSSS SYSTEMS				
NSSS Fluid Systems	Pressure, Temperature, Flow	Reactor Coolant System (RCS) Components	Yes	• IV.1 - IV.6, VI.1
NSSS Auxiliary Systems	Pressure, Flow, Temperature, Cooldown Rate	Various	Yes	• VI.1
NSSS/BOP INTERFACE SYSTEMS				
NSSS Control Systems	Margin to trip	Valves, heaters	Yes*	• VI.5
Low Temperature Overpressure Protection (LTOP) System	None	Valves	Yes	• VI.5
Reactor Vessel	Fluence, temperature	Pressure vessel	Yes	• IV.1
Reactor Internals	Thermal hydraulic	Reactor internals	Yes	• IV.1.2
Piping and Supports	Stress, presssure, temperture	Piping and supports	Yes	• IV.2
Control Rod Drive Mechanisms	Pressure, temperature	Housings, drive mechanisms	Yes	• IV.3
Reactor Coolant Pumps and Motors	Pressure, temperature, amps	Pumps and motors	Yes	• IV.4
Steam Generators	Thermal-hydraulic, stress	Steam generators	Yes	• IV.5
Pressurizer	Stress, fatigue	Pressurizer	Yes	• IV.6
NSSS Auxiliary Equipment	Pressure, temperature, fatigue	Various	Yes	• IV.7
Fuel	None	Fuel	Yes	• IV.8
Containment	Mass and Energy Release	Containment and protection	Yes	• II.2
Spent Fuel Pool Cooling System	Temperature, cooling	Various	Yes	• VI.3.7
PROGRAMS				
Environmental Qualification	None	None	Yes	• VII.6.1
Motor Operated Valves	None	None	Yes	• VII.6.2
Air and Hydraulic Operated Valves	None	None	Yes	• VII.6.3
Flow Accelerated Corrosion	None	None	Yes	• VII.6.4

TABLE 1 – SYSTEM AND PROGRAM REVIEW SUMMARY				
SYSTEM/ COMPONENT/ PROGRAM	**PARAMETERS WITH MUR UPRATE POTENTIAL IMPACT**	**COMPONENTS IMPACTED**	**BOUNDED BY EXISTING DESIGN/ ANALYSES?**	**REFERENCES** (Report Section Number)
High Energy Line Break	None	None	Yes	• VII.6.5
Station Blackout	None	None	Yes	• II.3.14, V
Fire Protection and Appendix R/Safe Shutdown	None	None	Yes	• VII.6.6
In-service Inspection	None	None	Yes	• VII.6.7
In-service Testing	None	None	Yes	• VII.6.8
Individual and Occupational Radiation Exposure	None	None	Yes	• VII.6.10
Radiological Environmental Assessments	None	None	Yes	• VII.6.9
Emergency and Abnormal Operating Procedures	None	None	Yes	• VII.2
Probabilistic Risk Assessment	None	None	Yes	• VII.6.11
Mechanical Piping Design	None	None	Yes	• VII.7

- Section VI.5 is not bounded for the margin-to-trip transients that assume 40% steam dump capability.
** TACW flow to iso-phase bus duct cooling system may need to be raised, both systems have margins to accommodate higher flow.

Evaluation Approach for the MUR Uprate Program

The licensed core power and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses. The current NSSS analyses of record for CNP Unit 1 generally model the core and/or NSSS thermal power in one of four ways. The approach taken for the proposed 1.66 percent power uprate for each of the four modeling approaches is provided below.

1. Some CNP Unit 1 analyses assume a nominal power level. These analyses have either been evaluated or revised to bound the 1.66 percent increased power level. Results of these evaluations and re-analyses demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.66 percent uprate conditions. Analyses that bound the MUR power uprate are addressed in Section II, and those analyses that do not bound the MUR power uprate are addressed in Section III.

2. Some CNP Unit 1 analyses already assume a core power level in excess of the proposed 3304 MWt. These analyses were previously performed at a higher power level (typically 3588 MWt) as part of prior plant programs. For these analyses, a portion of the available margin may be applied to offset the 1.66 percent uprate. Consequently, these analyses have been evaluated to confirm that sufficient analysis margin exists to envelop the 1.66 percent uprate. These analyses bound this MUR power uprate, and are addressed in Section II.

3. Some CNP Unit 1 analyses apply a 2 percent increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been revised for the 1.66 percent uprate conditions because the sum of increased core power level (1.66 percent) and the reduced power measurement uncertainty (0.31 percent) fall within the previously analyzed conditions. These analyses bound the MUR power uprate, and are addressed in Section II.

   The power calorimetric uncertainty calculation described in Section I indicates that with the Caldon instrumentation installed, the power measurement uncertainty, based on a 95 percent probability at a 95 percent confidence interval (i.e., 95/95 power measurement uncertainty), is approximately 0.31 percent. Therefore, these analyses only need to account for this reduced power measurement uncertainty. Accordingly, the existing 2 percent uncertainty can be allocated such that up to 1.66 percent is applied to provide sufficient margin to address the uprate to 3304 MWt, and the reduced power measurement uncertainty value is retained in the analysis to account for the uncertainty in power. Additionally, this third type of analyses often include other conservative assumptions not affected by the 1.66 percent uprated power. In summary, the use of the calculated 95/95 power measurement uncertainty and retention of other conservative assumptions ensure that the margin of safety for these analyses would not be reduced.

4. Some CNP Unit 1 analyses are performed at zero percent power conditions, or do not model the core power level. Consequently, these analyses have not been re-performed, since they are unaffected by the core power level. These analyses bound this MUR power uprate, and are addressed in Section II.

Table 2, "MUR Power Uprate Impact on CNP Unit 1 Accident/Transient Analyses," summarizes the accident and transient analyses for CNP Unit 1, and documents whether the analysis of record bounds plant operation at the proposed uprated power level. Details of these evaluations are provided in subsequent sub-sections.

Table 2 -- MUR Power Uprate Impact on CNP Unit 1 Accident/Transient Analyses			
Accident/Transient	**CNP Unit 1 UFSAR Section**	**Impact of Uprate on Current UFSAR Analysis**	**Sub-Section of this Attachment**
Events Reanalyzed to Bound 1.66 Percent Power Uprate			
Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power	14.1.2	Not Bounded	III.3
Loss of External Electrical Load – Departure from Nucleate Boiling (DNB) analysis	14.1.8	Not Bounded	III.2
Excessive Heat Removal Due to Feedwater System Malfunctions – from hot-full power (HFP) conditions	14.1.10	Not Bounded	III.1
Loss of Coolant Accident (LOCA) and Related Analyses			
Loss-of-Coolant Accident (LOCA) Forces	3.5.1 4.3.1 14.3	Bounded	II.1.1
Large Break LOCA	14.3.1	Bounded	II.1.2
Small Break LOCA	14.3.2	Bounded	II.1.2
Post-LOCA Long-Term Core Cooling	14.3.1	Bounded	II.1.3
Hot Leg Switchover	14.3.1	Bounded	II.1.4
Steam Generator Tube Rupture (SGTR)	14.2.4	Bounded	II.1.5
Non-LOCA Events			
Events Bounded by Assuming 102% Power Assumption			
Locked Rotor – RCS overpressure, maximum cladding temperature, and maximum zirconium-water reaction analyses	14.1.6	Bounded	II.3.1
Loss of External Electrical Load – overpressure analysis	14.1.8	Bounded	II.3.2
Loss of Normal Feedwater Flow	14.1.9	Bounded	II.3.3
Loss of All Alternating Current (AC) Power to Plant Auxiliaries	14.1.12	Bounded	II.3.3
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – from HFP conditions	14.2.6	Bounded	II.3.4
Events Evaluated Using Existing DNB Margin			
Uncontrolled RCCA Withdrawal from a Subcritical Condition	14.1.1	Bounded	II.3.7
RCCA Misalignment	14.1.3	Bounded	II.3.5

Table 2 -- MUR Power Uprate Impact on CNP Unit 1 Accident/Transient Analyses			
Accident/Transient	**CNP Unit 1 UFSAR Section**	**Impact of Uprate on Current UFSAR Analysis**	**Sub-Section of this Attachment**
RCCA Drop	14.1.4	Bounded	II.3.5
Loss of Reactor Coolant Flow	14.1.6	Bounded	II.3.6
Locked Rotor Analysis – DNB case	14.1.6	Bounded	II.3.1
Non-Limiting/Bounding Events			
Chemical and Volume Control System (CVCS) Malfunction	14.1.5	Bounded	II.3.8
Excessive Heat Removal Due to Feedwater System Malfunctions – from hot-zero power (HZP) Conditions	14.1.10	Bounded	II.3.9
Excessive Load Increase Incident	14.1.11	Bounded	II.3.10
Rupture of a Steam Pipe (core response)	14.2.5	Bounded	II.3.11
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – from HZP conditions	14.2.6	Bounded	II.3.12
Feedwater and Steam Line Break Mass and Energy Releases			
Short-Term Feedwater Line Break Inside Containment	14.3.4.4.1	Bounded	II.2.1
Short-Term Inside Containment	14.3.4.4.1	Bounded	II.2.1
Long-Term Inside Containment	14.3.4.4.2	Bounded	II.2.1
Long-Term Outside Containment / Equipment Qualification	14.4.3.5.1 (Unit 2 UFSAR)	Bounded	VII.6.1
Post-LOCA Hydrogen Generation			
Post-LOCA Hydrogen Generation Rates	14.3.6	Bounded	II.2.2
LOCA Mass and Energy Releases			
Long-Term	14.3.4.3.1.2	Bounded	II.2.3
Short-Term	14.3.4.5.1	Bounded	II.2.3
Containment Integrity			
Peak Containment Pressure Transient	14.3.4.1.3.1.3	Bounded	II.2.3
Containment Subcompartment			
Reactor Cavity	14.3.4.2.8	Bounded	II.2.3
Pressurizer Enclosure Subcompartment	14.3.4.2.5	Bounded	II.2.3
Loop Subcompartment	14.3.4.2.7	Bounded	II.2.3
Steam Generator Enclosure Subcompartment	14.3.4.2.4	Bounded	II.2.3
Fan/Accumulator Room Subcompartment	14.3.4.2.6	Bounded	II.2.3

Table 2 -- MUR Power Uprate Impact on CNP Unit 1 Accident/Transient Analyses			
Accident/Transient	**CNP Unit 1 UFSAR Section**	**Impact of Uprate on Current UFSAR Analysis**	**Sub-Section of this Attachment**
Analyses Performed in Accordance with Specific Regulatory Requirements			
Anticipated Transients Without Scram (ATWS) (10 CFR 50.62)	3.3.3	Bounded	II.3.13
Station Blackout (SBO) (10 CFR 50.63)	8.7	Bounded	II.3.14

Design Operating Parameters and Initial Conditions

The revised NSSS design thermal and hydraulic parameters that were changed as a result of the MUR Uprate Program serve as the basis for the NSSS analyses and evaluations. These revised parameters are presented in Table 3, "CNP Unit 1 MUR Uprate - NSSS Design Parameters." The revised parameters include a full-power normal operating T_{avg} range of 553.7° to 575.4°F, from the current design values of 553.0° to 576.3°F. The full-power T_{avg} range was adjusted slightly for the MUR Uprate Program to maintain the current maximum vessel T_{hot} (609.1°F) and minimum vessel T_{cold} (519.2°F) conditions supported by the current design parameters.

Cases 1 and 2 of Table 3 vary the steam generator tube plugging (SGTP) from 0 percent to 30 percent, respectively, while maintaining a T_{avg} of 553.7°F, which was calculated based upon maintaining a minimum vessel T_{cold} of 519.2°F.

Cases 3 and 4 vary SGTP from 0 percent to 30 percent, respectively, while maintaining a T_{avg} of 575.4°F which was calculated based upon maintaining a maximum vessel T_{hot} of 609.1°F.

Higher steam pressures than those presented in Table 3 are possible, predominantly due to lower steam generator tube fouling. Where a higher steam pressure is more limiting for a particular analysis, a greater steam pressure of 856 psia at $14.54x10^6$ lb/hr has been used rather than the maximum steam pressure of 840 psia at $14.53x10^6$ lb/hr listed in Table 3.

For accident analyses that are performed to demonstrate that the DNB acceptance criteria are met, nominal values of initial conditions are assumed. In accordance with the Westinghouse Revised Thermal Design Procedure (RTDP) methodology delineated in WCAP-11397-P-A (Reference 6), uncertainty allowances on power, RCS flow, temperature, and pressure are considered in the convolution of uncertainties to statistically establish the DNB ratio (DNBR) limit.

For accidents that are not DNB-limited, or in which RTDP is not utilized, the initial conditions assumed in the analysis include the maximum steady-state errors applied in the direction that yields the more limiting analysis results.

The only uncertainty that changed as a result of the MUR Uprate Program is the power measurement uncertainty, which now is ±0.31 percent. None of the other uncertainties (i.e., average RCS temperature, pressurizer pressure, and RCS flow) need to be revised.

The effect of the reduced power measurement uncertainty has been accounted for in the analysis/evaluation of the various accidents and transients discussed in Sections II and III. For analyses that utilize the RTDP method for the calculation of the minimum DNBR, the uncertainties are accounted for in the minimum DNBR safety analysis limit, rather than being accounted for explicitly in the analyses resulting in no change to DNB-related TS.

The NSSS design parameters are the fundamental parameters used as input in all of the NSSS analyses. These design parameters are the primary and secondary side system conditions (temperatures, pressures, and flow) that are used as the basis for the NSSS analyses and evaluations. These parameters are revised to accommodate the proposed 1.66 percent increase in licensed core power from 3250 MWt to 3304 MWt. The NSSS parameters were conservatively generated for a power uprating of 2 percent (i.e., 3315 MWt) to bound the actual uprating. Furthermore, the evaluations have been performed to support a power uprate such that the sum of the uprate plus uncertainty is less than or equal to 2 percent. In support of the 1.66 percent uprate, these parameters have been incorporated, as required, into the applicable safety analyses and NSSS system and component evaluations.

I&M notes that various WCAPs that are part of the CNP Unit 1 licensing basis, as referenced in this licensing amendment request, may have included explicit references to their use of "102 percent of licensed core power levels," (e.g., WCAP-11902). I&M does not consider that these WCAPs require revision to reflect this requested power uprate. Rather, it will be understood that those statements refer to the previously-required 2 percent 10 CFR 50 Appendix K margin and the currently licensed power level.

The revised NSSS design thermal and hydraulic parameters that were changed as a result of the MUR Uprate Program are presented in Table 3, "CNP Unit 1 MUR Power Uprate - NSSS Design Parameters."

Table 3 -- CNP Unit 1 MUR Power Uprate - NSSS Design Parameters				
	2% Uprate			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	102	102	102	102
MWt	3327	3327	3327	3327
10^6 BTU/hr	11,352	11,352	11,352	11,352
Reactor Power, MWt	3315	3315	3315	3315
10^6 BTU/hr	11,311	11,311	11,311	11,311
Thermal Design Flow, Loop gpm	83,200	83,200	83,200	83,200
Reactor 10^6 lb/hr	130.1	130.1	126.5	126.5
Reactor Coolant Pressure, psia	2250[1]	2250[1]	2250[1]	2250[1]
Core Bypass, %	7.1	7.1	7.1	7.1
Reactor Coolant Temperature, °F				
Core Outlet	593.1	593.1	613.6	613.6
Vessel Outlet, T_{hot}	588.2	588.2	609.1	609.1
Core Average	557.6	557.6	579.5	579.5
Vessel Average, T_{avg}	553.7	553.7	575.4	575.4
Vessel/Core Inlet, T_{cold}	519.2	519.2	541.7	541.7
Steam Generator Outlet	518.9	518.9	541.5	541.5
Steam Generator				
Steam Temperature, °F	500.4	489.4	523.9	513.1
Steam Pressure, P_{steam}, psia	684	618	840	765
Steam Flow, 10^6 lb/hr total	14.46	14.44	14.53	14.50
Feedwater Temperature, °F	437.4	437.4	437.4	437.4
Moisture, % max.	0.10	0.10	0.10	0.10
Tube Plugging, %	0	30	0	30
Zero Load Temperature, °F	547	547	547	547
HYDRAULIC DESIGN PARAMETERS				
Mechanical Design Flow, gpm	99,700			
Minimum Measured Flow, gpm total	339,100			

[1] Plant may also operate at 2100 psia. Operating temperatures for 2100 psia have an approximate 0.2°F increase in T_{cold} and a 0.2°F decrease in T_{hot}.

Core Thermal Limits and Over-Temperature and Overpower Delta-T (ΔT) Setpoints

Two essential inputs to the non-LOCA safety analyses are the core thermal limits and the resulting over-temperature ΔT (OTΔT) and overpower ΔT (OPΔT) setpoints.

A revised set of core thermal limits was developed due to the MUR Uprate Program. It was determined that the OTΔT and OPΔT setpoint coefficients did not need to be revised to accommodate the increased core power, based on the revised set of core thermal limits. However, changes to the reference average temperatures in the CNP Unit 1 TS are required. The proposed changes will restrict OTΔT reference average temperature (T') to 574°F to ensure that the current TS f(ΔI) penalty limits remain bounding, and restrict OPΔT reference average temperature (T") to 562.1°F to preserve the current OTΔT and OPΔT setpoints. A description of the proposed TS changes is provided in Section VIII, "Changes to Technical Specifications, Protection System Settings, and Emergency System Settings."

The core thermal limits are inputs for the calculation of the OTΔT and OPΔT reactor protection setpoints. The approved method used by Westinghouse to calculate these setpoints is described in WCAP-8746, "Design Bases for the Thermal Overpower ΔT and Thermal Over-Temperature ΔT Trip Functions," (Reference 7). In addition, a partial derivative approximation of the DNB core thermal limit lines (DNBR model) is input to the LOFTRAN code to conservatively approximate the change in the DNBR during certain DNB-related transients (primarily those in which the reactor coolant flow is constant). This partial derivative approximation is possible because the DNB core thermal limit lines are relatively linear with respect to changes in reactor coolant temperature, pressure, and thermal power. A more detailed discussion of the Westinghouse DNBR model is provided in Section 6.1.6 of WCAP-7907-A, "LOFTRAN Code Description," (Reference 8). Applying the method described in WCAP-8746, the current OTΔT and OPΔT setpoints were found to protect the revised set of core thermal limits. Thus, no changes to the OTΔT or OPΔT setpoint coefficients are required.

NSSS Analysis History for CNP Unit 1

A significant number of analyses have been performed for CNP Unit 1 to support operating flexibility and potential plant uprating in the past. The following provides a brief history to aid in the understanding of the evaluations that support the 1.66 percent power uprating for CNP Unit 1.

In November 1988, analyses were performed to support operation at reduced primary system pressure and temperature. This effort, referred to as the Reduced Temperature and Pressure Program, was undertaken to reduce the propensity for the initiation and propagation of stress corrosion cracking in the Unit 1 steam generators. The licensing submittal for this effort was provided in an October 1988 license amendment request (Reference 9), which included WCAP-11902 (Reference 10) as an attachment. The Reduced Temperature and Pressure Program license amendment was approved in an NRC Safety Evaluation Report (SER) dated June 9, 1989 (Reference 11). The analysis for this work supported average SGTP levels of 10 percent, with peak levels of 15 percent. Subsequently, Supplement 1 to WCAP-11902 (Reference 12), referred to as the Rerating Program, was issued to support a rerating of CNP Units 1 and 2, with reduced temperature and pressure operation. The analyses in WCAP-11902, Supplement 1, supported Unit 1 operation for the range of conditions described in WCAP-11902, but at an increased NSSS power of 3425 MWt. Applicable portions of WCAP-11902, Supplement 1, were submitted in support of license amendment requests to support a change to the steam generator stop valve closure time (Reference 13). This change was approved by Unit 1 License Amendment No. 147 (Reference 14).

In 1995, efforts to increase the Unit 1 SGTP limit to 30 percent were documented in WCAP-14285 (Reference 15). While some of the analyses had to reduce the core power to 3250 MWt to offset the RCS flow reduction to 83,200 gallons per minute (gpm) due to the higher tube plugging levels, many of the analyses continued to cover the range of conditions and uprated power level documented in WCAP-11902, including Supplement 1. The 30 percent SGTP limit was approved by Unit 1 License Amendment No. 214 (Reference 16). In March 2001, the evaluations that supported implementation of the Unit 1 replacement steam generators (RSGs) were completed. These evaluations were adopted into the CNP Unit 1 licensing basis by a 10 CFR 50.59 evaluation. Finally, in March 2002, an evaluation was completed to support operation with or without plugging devices installed in the fuel assembly guide tubes. This evaluation was incorporated into the Unit 1 licensing basis by a 10 CFR 50.59 evaluation that was performed to support the Unit 1 Cycle 18 reload analysis. Significant portions of the evaluations and assessments presented in this license amendment request build from previous analyses and evaluations that were performed to establish the conditions to eventually uprate CNP Unit 1 to a higher power level than is being requested by this license amendment request.

References (Introduction Section)

1. Letter from W. D. Beckner, NRC, "NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

2. WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," dated January 1983

3. Letter from R. E. Martin, NRC, to J. A. Scalice, Tennessee Valley Authority, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendments Regarding Increase of Reactor Power to 3459 Megawatts Thermal (TAC No. MA9152)," dated January 19, 2001

4. Letter from D. H. Jaffe, NRC, to C. L. Terry, TXU Electric, "Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 – Issuance of Amendments Re: Increase in Allowable Thermal Power to 3458 MWt and Deletion of Texas Municipal Power Agency from the Operating Licenses (TAC Nos. MB1625 and MB1626)," dated October 12, 2001

5. Letter from L. J. Burkhart, NRC, to L. W. Myers, FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Re: 1.4 Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves (TAC Nos. MB0996, MB0997, and MB2557)," dated September 24, 2001

6. WCAP-11397-P-A, "Revised Thermal Design Procedure," Friedland, A. J. and Ray S., dated April 1989

7. WCAP-8746, "Design Bases for the Thermal Overpower ΔT and Thermal Over-Temperature ΔT Trip Functions," dated March 1977

8. WCAP-7907-A, "LOFTRAN Code Description," dated April 1984

9. Letter from M. P. Alexich, I&M, to T. E. Murley, NRC, "Reduced Temperature and Pressure Program Analyses and Technical Specification Changes," AEP:NRC:1067, dated October 14, 1988

10. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," dated October 1988

11. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

12. WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report," dated September 1989

13. Letter from M. P. Alexich, I&M, to T. E. Murley, NRC, "Expedited Technical Specification Change Request Steam Generator Stop Valves," AEP:NRC:1120, dated January 31, 1990

14. Letter from T. Colburn, NRC, to M. P. Alexich, I&M, "Amendment No. 147 to Facility Operating License No. DPR-58: (TAC No. 75892)," dated August 22, 1990

15. WCAP-14285, "Donald C. Cook Nuclear Power Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report", dated May 1995

16. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC Nos. M92587 AND M92588)," dated March 13, 1997

I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty

1. The feedwater flow measurement system being installed at CNP Unit 1 is an LEFM CheckPlus ultrasonic, multi-path, transit time flowmeter. The design of this advanced flow measurement system is addressed in detail by the manufacturer, Caldon, in Topical Reports ER-80P, Revision 0 (Reference I.1), and ER-157P, Revision 5 (Reference I.2).

   The LEFM CheckPlus system at CNP Unit 1 consists of one flow element installed in the common portion of the feedwater flow loops and an electronic unit installed in the Plant Process Computer (PPC) room. This flow element is installed approximately 10 pipe diameters downstream from the start of the common header and approximately 3.7 pipe diameters upstream from a 45° elbow. The installation location of this flow element conforms to the requirements in Topical Reports ER-80P and ER-157P.

   A. The referenced Topical Reports are:
      i. ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, dated March 1997
      ii. ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System," Revision 5, dated October 2001

   B. The NRC approved the subject Topical Reports referenced in Item (A) above on the following dates:
      i. ER-80P NRC SER dated March 8, 1999 (Reference I.3)
      ii. ER-157P NRC SER dated December 20, 2001 (Reference I.4)

   C. The LEFM CheckPlus system will be permanently installed at CNP Unit 1 in accordance with the requirements of Topical Reports ER-80P and ER-157P. This system will be used for continuous calorimetric power determination by serial link with the PPC and will incorporate self-verification features to ensure that hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

      The CNP Unit 1 LEFM CheckPlus system was calibrated in a site-specific model test at Alden Research Laboratories with traceability to National Standards. The LEFM CheckPlus system will be installed and commissioned according to Caldon procedures, which include verification of ultrasonic signal quality and hydraulic velocity profiles as compared to those tested during site-specific model testing.

D. In approving Caldon Topical Reports ER-80P and ER-157P, the NRC established four criteria to be addressed by each licensee. The four criteria and a discussion of how each will be satisfied for CNP Unit 1 follows:

Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

Response to Criterion 1

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM CheckPlus system. Plant maintenance and calibration procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the LEFM CheckPlus system operational and raising core power above 3250 MWt. The incorporation of, and continued adherence to, these requirements will assure that the LEFM CheckPlus system is properly maintained and calibrated.

Contingency plans for operation of the plant with the LEFM CheckPlus out of service are described in Section G/H below.

Criterion 2

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

Response to Criterion 2

This criterion is not applicable to CNP Unit 1. CNP Unit 1 currently uses venturies to obtain the daily calorimetric heat balance measurements. I&M is installing a new LEFM CheckPlus system at CNP Unit 1 in anticipation of approval of this proposed amendment. Installation of this system will be completed prior to implementation of the requested license amendment.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

The total power calorimetric accuracy using the LEFM CheckPlus system is determined by evaluating the reactor thermal power sensitivity to deviations in each of the process inputs used to calculate reactor thermal power. The methodology is consistent with that used for the reactor power uncertainty described in the Improved Thermal Design Procedure (ITDP), WCAP-8567 (Reference I.5), which was applied at CNP Unit 1, using feedwater venturies. The NRC has reviewed the ITDP methodology and approved its use at CNP (Reference I.6). Subsequent to this, the NRC also approved the use of the RTDP methodology (References I.7 and I.8), which uses the same methodology for evaluating calorimetric uncertainties as the ITDP. In both the ITDP and LEFM CheckPlus reactor power uncertainty calculations, the instrumentation uncertainties and calculated values are determined. The reactor thermal power sensitivity is then calculated for each parameter. The individual contributions to the power uncertainty are then combined using a statistical summation to determine the total power measurement uncertainty. The methodology used for this combination is not new, and this statistical approach has been utilized in other applications based on the ITDP. Also, the use of a statistical approach in analysis complies with the recommendations of ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation" (Reference I.9).

Criterion 4

For plants where the ultrasonic meter (including LEFM) was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

Criterion 4 does not apply to CNP Unit 1. The calibration factor for the Unit 1 spool piece was established by tests of this spool at Alden Research Laboratory in April 2002. These included tests of a full-scale model of the CNP Unit 1 hydraulic geometry and tests in a straight pipe.

Final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process verifies bounding calibration test data (See Appendix F of ER-80P, Reference I.1) and provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation. Final commissioning is expected to be completed in October 2002.

E. The following table summarizes the core thermal power measurement uncertainty at CNP Unit 1:

Table I-1 -- Unit 1 Process Parameter Inputs to Reactor Thermal Power

Parameter	Bounding Value	Uncertainty	Sensitivity %RTP
FW pressure, psig	725.10	±15	±0.001752%
FW temperature, °F	434.25	±0.6	±0.084520%
FW mass flow rate, 10^6 lb/hr	14.16	±0.411	±0.292524%
Average steam pressure, psig	670.00	±7.7	±0.021241%
Total blowdown flow, gpm	160.00	±10	±0.025239%
Charging flow, gpm	119.80	+5.7/-8.3	±0.000000%
Charging pressure, psig	2412.70	±49.82	±0.000000%
Charging temperature, °F	451.90	±6.5	±0.000000%
Letdown pressure, psig	339.60	+14.7876 / -14.0556	±0.000000%
Letdown temperature, °F	121.00	±1.3661	±0.000000%
Letdown flow, gpm	114.00	+5.2 / -5.6	±0.000000%
Pressurizer pressure, psig	2083.60	+19.752 / -18.441	±0.000000%
Reactor T_{cold}, °F	525.90	±2.81	±0.000000%
Volume Control Tank (VCT) temperature, °F	120.80	±1.5646	±0.000000%
RMS Total Uncertainty (Root Mean Squared)			+/- 0.31% RTP

F. The following information addresses specific aspects of calibration and maintenance procedures addressing the LEFM CheckPlus system.

i. Calibration and maintenance are performed by CNP Instrumentation and Controls (I&C) - Maintenance Department personnel using site procedures. The site procedures will be developed using the Caldon technical manuals. All work will be performed in accordance with site work control procedures.

Routine preventive maintenance activities will include physical inspections, power supply checks, back-up battery replacements, and internal oscillator frequency verification.

Ultrasonic signal verification and alignment are performed automatically with the LEFM CheckPlus system. Signal verification is possible by review of signal quality measurements performed and displayed by the LEFM CheckPlus system.

I&C maintenance personnel will be trained and qualified per the I&M INPO-accredited training program before calibration is performed and prior to raising power above 3250 MWt.

ii. The LEFM CheckPlus is designed and manufactured in accordance with Caldon's 10 CFR 50, Appendix B, Quality Assurance Program and its Verification and Validation Program. Caldon's Verification and Validation Program fulfills the requirements of ANSI/IEEE-ANS Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and ASME NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with guidance for software verification and validation in EPRI TR-103291S, "Handbook for Verification and Validation of Digital Systems," dated December 1994. Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM CheckPlus system are provided in Caldon Topical Report ER-80P, Section 6.4 and Table 6.1.

iii. Corrective actions involving maintenance will be performed by I&C maintenance personnel, qualified in accordance with I&M's I&C Training Program, and formally trained on the LEFM CheckPlus system.

iv. Reliability of the LEFM CheckPlus system will be monitored by CNP's System Engineering personnel in accordance with the requirements of I&M's Corrective Actions Program. Equipment problems for all 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. The Caldon LEFM CheckPlus system software falls under the I&M 10 CFR 50, Appendix B, Quality Assurance Program, which includes specific software quality assurance requirements. Corrective Action procedures are maintained that include instructions for notification of deficiencies and error reporting.

v. The CNP Unit 1 LEFM CheckPlus system is included in Caldon's Verification and Validation Program, and procedures are maintained for user notification of important deficiencies. The Caldon LEFM CheckPlus system purchase order included requirements that Caldon inform I&M of any deficiencies in accordance with the Caldon maintenance agreement and/or 10 CFR Part 21 reporting requirements.

G/H. The proposed allowed outage time for operation at the uprated power level with an LEFM CheckPlus system out of service is 48 hours, provided steady state conditions (i.e., no power changes in excess of 10 percent) persist throughout the 48-hour period. The bases for this proposed allowed outage time period are:

- There is an on-line calibration of a set of alternate plant instruments to be used if the LEFM CheckPlus system is out of service for a longer period. These alternate instruments will be calibrated to the last valid value provided by the LEFM CheckPlus system, and their accuracy will gradually degrade over time as a result of nozzle fouling and transmitter drift. The gradual accuracy degradation is likely to be imperceptible for a 48-hour period provided steady-state conditions persist.

- Most repairs to the LEFM CheckPlus system can be made within an eight-hour shift. Forty-eight hours gives plant personnel time to plan the work, make repairs, and to verify normal operation of the LEFM CheckPlus system within its original uncertainty bounds at the same power level and indications as before the failure.

- Operations personnel will operate the plant based on the calibrated alternate plant instruments when an LEFM CheckPlus system is not available. It is considered prudent to provide them time to become accustomed to operation with the alternate plant instruments prior to requiring a power de-rate. The power de-rate evolution could, in many cases, be avoided altogether since a repair would be accomplished prior to the expiration of the 48-hour period.

- If the plant experiences a power change of greater than 10 percent during the 48-hour period, then the permitted maximum power level would be reduced upon return to full power in accordance with the power levels described below, since a plant transient may result in calibration changes of the alternate instruments.

- As described in Reference I.2, the LEFM CheckPlus system consists of two planes (8 paths) of transducers. If the LEFM has experienced an outage of only one plane (4 paths) of the instrument, then the permitted maximum full power will not be affected. This operation is justified in accordance with topical report ER-157P (Reference I.2) and its associated NRC approvals referenced below (Reference I.4).

- If the 48-hour outage period is exceeded, then the plant will operate at a power level consistent with the accuracy of the alternate plant instruments.

The LEFM CheckPlus system at CNP Unit 1 consists of a single feedwater measurement spool piece installed in the feedwater header, and the associated electronics unit. Failure of the LEFM CheckPlus system will result in calculation of thermal power based on operation of the main feedwater flow measurement(s) venturies in the main feedwater lines and one or more resistance temperature detectors (RTDs) in the feedwater system. These alternate instruments would have been calibrated to the last valid value of the thermal power calculation based on the LEFM CheckPlus flow and temperature measurements. Operation during this period would be at a power level consistent with operation entirely on these calibrated alternate instruments. With the LEFM CheckPlus system out of service, the thermal power uncertainty increases such that the justifiable core power level is reduced from 3304 MWt to 3250 MWt.

References (Section I)

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

I.2. ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System," Caldon, Inc., dated October 2001

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

I.4. Letter from S. A. Richards, NRC, to M. A. Krupa, Entergy, "Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station –

Review of Caldon, Inc. Engineering Report ER-157P (TAC Nos. MB2397, MB2399 and MB2468)," dated December 20, 2001

I.5. WCAP-8567-P-A, "Improved Thermal Design Procedure", approved February 1989

I.6. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

I.7. Letter from E. E. Fitzpatrick, I&M, to Nuclear Regulatory Commission, "Proposed Technical Specification Changes Supported by Analysis to Increase Unit 1 Steam Generator Tube Plugging Limit and Certain Proposed Changes for Unit 2 Supported by Related Analyses," AEP:NRC:1207, dated May 26, 1995

I.8. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC Nos. M92587 and M92588)," dated March 13, 1997

I.9. ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation," approved February 29, 2000

II. Accidents and Transients for which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level

Table II-1 summarizes the CNP Unit 1 accident and transient analyses that were determined to bound plant operation at the 1.66 percent power level proposed by the MUR Uprate Program. Details of these evaluations follow in subsequent sub-sections.

Table II-1 Bounding Accident and Transient Design Basis Analyses

Accident/Transient	UFSAR Section	Assumed Core Power Level	NRC Approval (Date and/or Reference Number)
Loss of Coolant Accident (LOCA) - Related Events			
LOCA Forces	3.5.1.5.1 3.5.1.5.2 3.5.1.5.3 3.5.1.6.5.3.1 14.3.3.1 14.3.3.2 14.3.3.3 14.3.3.4 4.3.1	N/A N/A N/A N/A 3588 MWt 3588 MWt 3588 MWt 3588 MWt N/A	December 23, 1999 (Ref. II.4) NRC approval for MULTIFLEX methodology provided in WCAP-8708-P/A and WCAP-8709-A (Ref. II.9)

Table II-1 Bounding Accident and Transient Design Basis Analyses			
Accident/Transient	**UFSAR Section**	**Assumed Core Power Level**	**NRC Approval (Date and/or Reference Number)**
Large Break LOCA	14.3.1	3315 MWt	Previous NRC approval of LBLOCA provided in (Ref. II.1). As noted in Unit 1 50.46 Report (Ref. II.2), LBLOCA analysis of record was re-analyzed for the Cycle 17 reload safety evaluation. New analysis was incorporated into licensing basis by 50.59 evaluation.
Small Break LOCA	14.3.2	3315 MWt	Previous NRC approval of SBLOCA provided in (Ref. II.1) As noted in Unit 1 50.46 Report (Ref. II.2), SBLOCA analysis of record was re-analyzed for the Cycle 17 reload safety evaluation. New analysis was incorporated into licensing basis by 50.59 evaluation
Post-LOCA Long-Term Core Cooling	14.3.1	3481 MWt	December 13, 1999 (Ref. II.3) December 23, 1999 (Ref. II.4)
Hot Leg Switchover	14.3.1	3481 MWt	December 13, 1999 (Ref. II.3) December 23, 1999 (Ref. II.4)
SGTR Thermal-Hydraulic Analysis for Use in Determining Dose Consequences	14.2.4	3588 MWt	June 9, 1989 (Ref. II.12)
Operator Action and margin to overfill assessment - SGTR	14.2.4.4	3250 MWt	October 24, 2001 (Ref. II.10)
Non-LOCA Events			
Events Bounded by 102% Power Assumption			
Locked Rotor – RCS overpressure, maximum cladding temperature, and maximum zirconium-water reaction analysis	14.1.6	3315 MWt	March 13, 1997 (Ref. II.1)

Table II-1 Bounding Accident and Transient Design Basis Analyses

Accident/Transient	UFSAR Section	Assumed Core Power Level	NRC Approval (Date and/or Reference Number)
Loss of External Electrical Load – overpressure analysis	14.1.8	3315 MWt	March 13, 1997 (Ref. II.1) Re-analysis performed for Unit 1 steam generator replacement was incorporated into licensing basis by 50.59 evaluation.
Loss of Normal Feedwater Flow	14.1.9	3457 MWt	March 13, 1997 (Ref. II.1) Re-analysis performed for Unit 1 steam generator replacement was incorporated into licensing basis by 50.59 evaluation.
Loss of All AC Power to the Plant Auxiliaries	14.1.12	3457 MWt	March 13, 1997 (Ref. II.1) Re-analysis performed for Unit 1 steam generator replacement was incorporated into licensing basis by 50.59 evaluation.
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – from HFP conditions	14.2.6	3315 MWt	March 13, 1997 (Ref. II.1)
Events Evaluated Using Existing DNB Margin			
Uncontrolled RCCA Withdrawal from a Subcritical Condition	14.1.1	0 MWt	March 13, 1997 (Ref. II.1)
RCCA Misalignment	14.1.3	3250 MWt	March 13, 1997 (Ref. II.1)
RCCA Drop	14.1.4	3250 MWt	March 13, 1997 (Ref. II.1)
Loss of Reactor Coolant Flow	14.1.6	3250 MWt	March 13, 1997 (Ref. II.1)
Locked Rotor Analysis - DNB case	14.1.6	3250 MWt	March 13, 1997 (Ref. II.1)
Non-Limiting Events and Transients			
CVCS Malfunction	14.1.5	N/A	June 9, 1989 (Ref. II.12) March 13, 1997 (Ref. II.1)
Excessive Heat Removal Due to Feedwater System Malfunctions – from HZP conditions	14.1.10	0 MWt	March 13, 1997 (Ref. II.1) Re-analysis performed for Unit 1 steam generator replacement was incorporated into licensing basis by 50.59 evaluation.
Excessive Load Increase Incident	14.1.11	3413 MWt	June 9, 1989 (Ref. II.12) March 13, 1997 (Ref. II.1)
Rupture of a Steam Pipe (core response)	14.2.5	0 MWt	March 13, 1997 (Ref. II.1)

Table II-1 Bounding Accident and Transient Design Basis Analyses			
Accident/Transient	**UFSAR Section**	**Assumed Core Power Level**	**NRC Approval (Date and/or Reference Number)**
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – from HZP conditions	14.2.6	0 MWt	March 13, 1997 (Ref. II.1)
Feedwater and Steam Line Break Mass and Energy Releases			
Short-Term Feedwater Line Break Inside Containment	14.3.4.4.1	0 MWt and 3588 MWt	Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation.
Short-Term Inside Containment	14.3.4.4.1	0 MWt	Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation
Long-Term Inside Containment	14.3.4.4.2	3660 MWt	June 9, 1989 (Ref. II.12) December 13, 1999 (Ref. II.3) Evaluation performed for Unit 1 steam generator replacement was incorporated into licensing basis by 50.59 evaluation.
Long-Term Outside Containment / Equipment Qualification	14.4.3.5.1 (Unit 2 UFSAR)	3660 MWt	December 13, 1999 (Ref. II.3)
Post-LOCA Hydrogen Generation			
Post-LOCA Hydrogen Generation Rates	14.3.6	3411 MWt	March 29, 2001 (Ref. II.15)
LOCA Mass and Energy Releases			
Long-Term	14.3.4.3.1.2	3481 MWt	December 13, 1999 (Ref. II.3)
Short-Term	14.3.4.5.1	3660 MWt	Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation
Containment Integrity			
Peak Containment Pressure Transient Analysis	14.3.4.1.3.1.3	3481 MWt	December 13, 1999 (Ref. II.3)
Containment Subcompartments Analyses			
Reactor Cavity	14.3.4.2.8	3660 MWt	September 10, 1973 (Ref. II.13) December 12, 1974 (Ref. II.14) Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation.

Table II-1 Bounding Accident and Transient Design Basis Analyses			
Accident/Transient	**UFSAR Section**	**Assumed Core Power Level**	**NRC Approval (Date and/or Reference Number)**
Pressurizer Enclosure Subcompartment	14.3.4.2.5	3660 MWt	September 10, 1973 (Ref. II.13) December 23, 1977 (Ref. II.11) Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation.
Loop Subcompartment	14.3.4.2.7	3660 MWt	June 9, 1989 (Ref. II.12) December 12, 1974 (Ref. II.14) Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation.
Steam Generator Enclosure Subcompartment – Short-Term Steam Line Break	14.3.4.2.4	N/A	September 10, 1973 (Ref. II.13) December 23, 1977 (Ref. II.11) Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation.
Fan/Accumulator Room Subcompartment	14.3.4.2.6	N/A	June 9, 1989 (Ref. II.12) December 12, 1974 (Ref. II.14) Containment Subcompartment Re-analysis effort; incorporated into licensing basis by 50.59 evaluation.
Analyses Performed in Accordance with Specific Regulatory Requirements			
Anticipated Transients Without Scram (10 CFR 50.62)	3.3.3	3411 MWt	April 14, 1989 (Ref. II.5) August 16, 1989 (Ref. II.6)
Station Blackout (10 CFR 50.63)	8.7	3250 MWt	October 31, 1991 (Ref. II.7) April 23, 1992 (Ref. II.8)

References (Table II-2)

II.1. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC Nos. M92587 and M92588)," dated March 13, 1997 (SER for Unit 1 Steam Generator Tube Plugging Program, as evaluated in WCAP-14285)

II.2. Letter from M. W. Rencheck, I&M, to Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Unit 1 Report of Loss-of-Coolant Accident Evaluation Model Changes," C1200-03, dated December 20, 2000

II.3. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6766 and M6767)," dated December 13, 1999 (SER for Unit 1 and 2 Containment Sump Modification Evaluations, as evaluated in WCAP-15302)

II.4. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)," dated December 23, 1999 (SER for Unit 1 and 2 RCCA Insertion Credit following a LBLOCA, as evaluated in WCAP-15245)

II.5. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Donald C. Cook Nuclear Plant Nos. 1 and 2, Compliance with ATWS Rule 10 CFR 50.62 (TAC No. 59082 and 59083)," dated April 14, 1989

II.6. Letter from J. Giitter, NRC, to M. P. Alexich, I&M, "Safety Evaluation for Generic Letter 83-28, Item 4.5.3, Reactor Trip Reliability – On-Line Functional Testing of the Reactor Trip System (TAC No. 53971 and 53972)," dated August 16, 1989

II.7. Letter from T. G. Colburn, NRC, to E. E. Fitzpatrick, I&M, "Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. 68532/68533)," dated October 31, 1991

II.8. Letter from J. F. Stang, NRC, to E. E. Fitzpatrick, I&M, "Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. M68532 and 68533)," dated April 23, 1992

II.9. WCAP-8708-P/A (Proprietary) and WCAP-8709-A (Non-Proprietary), MULTIFLEX, A Fortran-IV Computer Program for Analyzing Thermal Hydraulic Structure System Dynamics," dated September 1977

II.10 Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC NOS. MB0739 and MB0740)," dated October 24, 2001

II.11 Letter from Nuclear Regulatory Commission to Indiana and Michigan Electric Company, "Supplement 7 to Safety Evaluation Report," dated December 23, 1977

II.12 Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

II.13 Safety Evaluation Report, "Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316," dated September 10, 1973

II.14 Supplement to Safety Evaluation Report, "Supplement No. 3 to Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316," dated December 12, 1974

II.15 Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment (TAC No. MB0908)," dated March 29, 2001

II.1 Loss of Coolant Accident (LOCA) and LOCA-Related Events (including SGTR)

II.1.1 LOCA Forces

The reactor vessel and internals have been qualified using LOCA hydraulic forces based on a minimum allowable cold leg operating temperature of 511.7°F, and a pressurizer pressure of 2250 pounds per square inch atmospheric (psia), plus uncertainty of 67 psi, for a total pressure of 2317 psia. The vessel and internals are qualified on the basis of branch line breaks, notably the accumulator line, RHR system line, and pressurizer surge line, as allowed under the leak-before-break criterion. The LOCA forces on the reactor coolant loop piping remain those cited in the existing CNP Unit 1 UFSAR (Reference II.1.1). These LOCA forces were evaluated against forces generated for (accumulator) branch line breaks at a cold leg temperature of 511.7 °F, and an RCS pressure of 2250 psia, using the MULTIFLEX code and the required one milli-second (msec) break opening time. This evaluation previously determined that the original double-ended guillotine break forces remain higher and are, therefore, bounding.

The MUR Uprate Program will use a reduced range of RCS temperatures, such that the minimum allowable RCS cold leg temperature for 102 percent power conditions at minimum thermal design flow will be 519.2°F, and the normal full power RCS pressure remains 2250 psia (see Table 3). Operation at the lower pressure of 2100 psia would reduce calculated LOCA hydraulic forces. Therefore, the existing analyses of LOCA forces remain bounding at the MUR uprate conditions.

The LOCA forces methodology applied in the most recent vessel and internals qualification analyses remains identical to that used in the analysis that credits control rod insertion for reactivity control in the long-term post-LOCA. This analysis was approved by the NRC in the SER for Unit 1 License Amendment No. 236 (Reference II.1.2). The MULTIFLEX methodology applied in determining that the existing loop LOCA forces remain bounding has been reviewed and approved by the NRC (Reference II.1.3).

II.1.2 Large Break LOCA (LBLOCA) and Small Break LOCA (SBLOCA)

The current licensing basis LBLOCA and SBLOCA analyses for CNP Unit 1 use a 10 CFR 50, Appendix K-type methodology. Due to the original requirements of Appendix K, both analyses employ a nominal core power of 3250 MWt plus an additional 2 percent calorimetric power measurement uncertainty (yielding an assumed core power of 3315 MWt). Consistent with the recent change to Appendix K, I&M has proposed to reduce the power measurement uncertainty to 0.31 percent for CNP Unit 1, and increase the nominal core power 1.66 percent to 3304 MWt. The analyses are conservative with respect to this uprate; thus, the uprate has no impact on the LBLOCA and SBLOCA analyses.

II.1.3 Post-LOCA Long-Term Core Cooling

The Westinghouse approach for satisfying the requirements of 10 CFR 50.46(b)(5), "Long-Term Cooling," concludes that the reactor will remain shut down by borated ECCS water residing in the RCS/recirculation sump following a LOCA. Since credit for the control rods is not taken in the short-term for a LBLOCA, the borated ECCS water provided by the refueling water storage tank and accumulators must have a boron concentration that, when mixed with the other sources of water, will result in the reactor core remaining subcritical, assuming all control rods out. However, control rod insertion is credited together with the available sources of boron to offset any potential effect of sump dilution during the cold leg injection recirculation cooling mode post-LBLOCA. The calculation is based upon the reactor steady-state conditions at the initiation of a LOCA and considers cases with borated and non-borated fluid in the post-LOCA recirculation sump. The other sources of water considered in the calculation of the recirculation sump boron concentration for CNP Unit 1 are the RCS, ECCS/RHR system piping, accumulators, ice bed mass, and boron injection tank (BIT) flow path. The water volumes and associated boric acid concentrations are not directly affected by the uprate. The core reload licensing process will confirm that there are no required changes to these volumes and boron concentrations. The current long-term core cooling analysis for CNP Unit 1 employs a nominal core power level of 3481 MWt. Also, consistent with the requirements outlined in 10 CFR 50, Appendix K, the decay heat model assumed in the LOCA Long-Term Core Cooling analysis is 1.2 times the values for infinite operating time in American Nuclear Society (ANS) Standard 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," dated 1971 (Reference II.1.4). Therefore, there is no impact on the LOCA Long-Term Core Cooling analysis, and the 1.66 percent power uprate is bounded.

II.1.4 Hot Leg Switchover

The licensing basis methodology employs a 2 percent calorimetric power uncertainty in accordance with the original requirements of 10 CFR 50, Appendix K. The current hot leg switchover analysis employs a nominal core power level of 3481 MWt. Therefore, a power increase to 3304 MWt has no impact on the hot leg switchover analysis and the 1.66 percent power uprate is bounded.

II.1.5 Steam Generator Tube Rupture (SGTR) - Thermal-Hydraulic Analysis

The analysis for the SGTR event, as documented in Section 14.2.4 of the Unit 1 UFSAR, is performed to demonstrate that the off-site radiological consequences remain below the guideline values. As input to the radiological consequences analysis, an SGTR thermal and hydraulic (T/H) analysis is performed. The T/H analysis calculates the primary-to-secondary break flow and steam released to the environment. The current SGTR T/H analysis for off-site radiological consequences was approved by the NRC in Unit 1 License Amendment No. 126 (Reference II.1.5). The SGTR analysis for on-site radiological consequences was submitted for review by Reference II.1.6, and is currently under NRC review. This not-yet-approved analysis demonstrates compliance with the regulatory dose rates. The SGTR T/H analysis considers core powers up to 3588 MWt. Therefore, the increase in core power to 3304 MWt is bounded by the analysis.

In addition to the SGTR analysis provided in the UFSAR, a supplemental SGTR analysis has been performed for Unit 1. The supplemental SGTR analysis provides a calculation of a more realistic response to an SGTR event by modeling operator actions and operator action times. The supplemental SGTR analysis is used to evaluate the margin to steam generator overfill and provides documentation to support the conclusion that the licensing basis SGTR T/H input into the radiological consequences is conservative. The NRC issued the SER approving the supplemental SGTR analysis for CNP Unit 1 via License Amendment No. 256, dated October 24, 2001 (Reference II.1.7).

The supplemental SGTR analysis is used to support the Emergency Operating Procedures (EOPs) and assumes nominal plant conditions. As such, the nominal NSSS power for Unit 1 (3262 MWt) was used in the supplemental analysis. Based on prior power uprate efforts, the change in power has a negligible effect on the SGTR margin-to-overfill analysis. A CNP Unit 1-specific sensitivity analysis was performed using a nominal NSSS power level increased by 2 percent (3327 MWt). The result of the sensitivity analysis demonstrated that this increase in power is bounded by the margin-to-steam generator overfill analysis results in the NRC-approved supplemental SGTR analysis. Additionally, the sensitivity analysis demonstrated that the licensing basis SGTR T/H results remained bounding compared to the supplemental SGTR calculation considering the uprated power.

Therefore, the current analyses for the SGTR event bounds the MUR Uprate Program.

References (Section II.1)

II.1.1. Updated Final Safety Analysis Report for CNP, Units 1 and 2, Version 17.2

II.1.2. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)," dated December 23, 1999

II.1.3. WCAP-8708-P/A (Proprietary) and WCAP-8709-A (Non-Proprietary), "MULTIFLEX, A Fortran-IV Computer Program for Analyzing Thermal Hydraulic Structure System Dynamics," dated September 1977

II.1.4. ANS Standard, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," dated 1971

II.1.5. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

II.1.6. Letter from R. P. Powers, I&M, to Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2 License Amendment Request for Control Room Habitability and Generic Letter 99-02 Requirements," C0600-13, dated June 12, 2000

II.1.7. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC NOS. MB0739 and MB0740)," dated October 24, 2001

II.2 Containment Analyses

II.2.1 Feedwater and Steam Line Break Mass and Energy Releases

The licensing basis safety analyses related to the feedwater and steam line break mass and energy releases were evaluated to determine the effect of a 1.66 percent power uprating. The evaluation determined that the NSSS design parameters, as shown in Table 3, "CNP Unit 1 MUR Uprate - NSSS Design Parameters," remain unchanged or are bounded by the current safety analyses (References II.2.1 and II.2.4 through II.2.8) and supplemental evaluations incorporated into the CNP Unit 1 licensing basis via 10 CFR 50.59.

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 included the steam generator enclosure and fan/accumulator room subcompartment analyses and long-term containment analyses, which were demonstrated to be unaffected by the MUR Uprate Program.

II.2.2 Post-LOCA Containment Hydrogen Generation

The CNP 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 1.66 percent power uprate.

II.2.3 LOCA Mass and Energy Releases

II.2.3.1 Long-Term LOCA Mass and Energy Release Analysis

The methodology for the most limiting LOCA mass and energy release calculation is contained in WCAP-10325-P-A (Reference II.2.2) up to the point of steam generator depressurization/equilibration. After steam generator depressurization/equilibration, the mass and energy release available to containment is generated directly from core boil-off/decay heat. The evaluations in WCAP-15302 (Reference II.2.3), together with supplemental evaluations as approved by Unit 1 License Amendment No. 234 (Reference II.2.1), together with supplemental evaluations incorporated into the CNP Unit 1 licensing basis via 10 CFR 50.59, constitute the current licensing basis containment analyses of record.

The current analyses of record was performed at an assumed Unit 1 core power level of 3481 MWt. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.2.3.2 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 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°F, minus 5 °F for uncertainty (574.1°F) and a cold leg (Vessel/Core Inlet Temperature) of 511.7°F, minus 5 °F for uncertainty (506.7°F), both conservatively bounded low for short-term considerations, were used. The MUR Uprate Program values of 588.2°F for the hot leg temperature and 519.2°F for the cold leg temperature are both bounded by the analyses of record. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

References (Section II.2)

II.2.1. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6766 and M6767)," dated December 13, 1999 (SER for Unit 1 and 2 Containment Sump Modification Evaluations, as evaluated in WCAP-15302)

II.2.2. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," dated May 1983

II.2.3. WCAP-15302, "Donald C. Cook Nuclear Plant Units 1 and 2, Modifications to the Containment Systems, Westinghouse Safety Evaluation (SECL 99-076, Revision 3)," dated September 1999

II.2.4 Letter from Nuclear Regulatory Commission to Indiana and Michigan Electric Company, "Supplement 7 to Safety Evaluation Report," dated December 23, 1977

II.2.5 Safety Evaluation Report, "Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316," dated September 10, 1973

II.2.6 Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

II.2.7 Supplement to Safety Evaluation Report, "Supplement No. 3 to Safety Evaluation by the Directorate of Licensing U. S. Atomoic Energy Commission in the Matter of Indiana & Michigan Electric company and Indiana & Michigan Power Company Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316," dated December 12, 1974

II.2.8 Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Increased Steam Generator

Plugging Limit (TAC Nos. M92587 and M92588), dated March 13, 1997 (SER for Unit 1 Steam Generator Tube Plugging Program, as evaluated in WCAP-14285)

II.3 Non-LOCA Analyses

This section addresses the potential effects of the MUR Uprate Program on the non-LOCA analyses presented in Chapter 14 of the UFSAR and analyses performed by I&M in response to regulatory requirements promulgated after the CNP Unit 1 OL was issued [i.e., ATWS (10 CFR 50.62) and SBO (10 CFR 50.63)].

The non-LOCA design-basis events are documented in Sections 14.1.1 through 14.1.12, 14.2.5 and 14.2.6 of the CNP Unit 1 UFSAR. Of these events, three non-LOCA events required re-analysis to demonstrate that the acceptance criteria will still be met at the 1.66 percent uprated power conditions. The re-analyses of these three events are provided in Section III of this attachment. Evaluations of the remaining non-LOCA events demonstrated that the existing analyses bound plant operation at the proposed 1.66 percent uprated power conditions. The following discussions summarize the evaluations of this latter set of non-LOCA events.

The following non-LOCA analyses are currently analyzed with an explicit 2 percent power measurement uncertainty.

- Locked Rotor – overpressure, maximum cladding temperature, and maximum zirconium-water reaction analysis (UFSAR Section 14.1.6)
- Loss of External Electrical Load – overpressure analysis (UFSAR Section 14.1.8)
- Loss of Normal Feedwater Flow (UFSAR Section 14.1.9)
- Loss of All AC Power to the Plant Auxiliaries (UFSAR Section 14.1.12)
- Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – full-power cases (UFSAR Section 14.2.6)

The improved thermal power measurement accuracy eliminates the need for the full 2 percent power uncertainty assumed in the analyses. The changes in the plant initial operating conditions resulting from the 1.66 percent power uprating were evaluated, and it was determined that these analyses remain valid. As such, the results and conclusions associated with these analyses remain valid at the 1.66 percent uprated power conditions.

Analyses that do not explicitly consider a 2 percent power uncertainty, such as those that use the RTDP methodology, were evaluated or re-analyzed to determine the effect of the 1.66 percent power increase. An evaluation was sufficient to determine that the effect the 1.66 percent power increase in nominal core power has on the following events is bounded by the current analyses:

- Uncontrolled RCCA Withdrawal from a Subcritical Condition (UFSAR Section 14.1.1)
- RCCA Misalignment (UFSAR Section 14.1.3)
- RCCA Drop (UFSAR Section 14.1.4)
- Loss of Reactor Coolant Flow (UFSAR Section 14.1.6)
- Locked Rotor – DNB case (UFSAR Section 14.1.6)

The following events continue to be bounded by related events or are otherwise unaffected by the proposed uprating:

- Chemical and Volume Control System Malfunction (UFSAR Section 14.1.5)
- Excessive Heat Removal Due to Feedwater System Malfunctions - zero power-cases (UFSAR Section 14.1.10)
- Excessive Load Increase Incident (UFSAR Section 14.1.11)
- Rupture of a Steam Pipe - core response analysis (UFSAR Section 14.2.5)
- Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – zero-power cases (UFSAR Section 14.2.6)
- Anticipated Transient Without Scram (ATWS)
- Station Blackout (SBO)

Evaluations of Non-LOCA Events

As shown in Table II-1, the majority of the non-LOCA events applicable to CNP Unit 1 have been evaluated as being bounding in support of the 1.66 percent power uprate. The evaluations are discussed by individual event in this section. The following subsections provide the details of the evaluations completed for the individual events.

II.3.1 Single Reactor Coolant Pump Locked Rotor Accident (UFSAR Section 14.1.6.4)

Since the MUR Uprate Program increase in core power may affect DNB, an evaluation was completed to confirm that the number of rods that exceed the DNBR limit is less than assumed in the dose analysis. The evaluation concluded that the existing statepoints for this event remain valid with the exception that the nominal core heat flux increases due to the 1.66 percent power uprate.

Revised statepoints that include the increased nominal heat flux were evaluated with respect to the rods-in-DNB limit. It was found that there are no rods-in-DNB.

The cases completed to confirm that the RCS pressure, clad temperature, and zirconium-water criteria are met were not re-analyzed. These cases currently model a 2 percent power uncertainty, which bounds the reduced uncertainty of 0.31 percent, combined with the 1.66 percent uprated power level. As such, the RCS pressure criterion continues to be met for the locked rotor event. Therefore, the current licensing basis remains bounding for the 1.66 percent MUR power uprate.

II.3.2 Loss of External Electrical Load – Overpressure Analysis (UFSAR Section 14.1.8)

The transient responses for a complete loss of load from full-power conditions are presented in the UFSAR as four cases; two cases with pressurizer pressure control evaluating minimum and maximum feedback conditions, and two without pressurizer pressure control evaluating minimum and maximum feedback conditions. Only the cases in which pressurizer pressure control is assumed available (cases in which the DNB design basis are examined) were re-analyzed for the MUR Uprate Program, and are discussed further in Section III.2.

The cases in the licensing basis analyses in which pressurizer pressure control is unavailable (the cases where RCS and main steam (MS) system overpressure criteria are examined), are not impacted by an increase in the nominal full power because the power level assumed in the current analysis for these cases (with 2 percent uncertainty) bounds that based upon the uprated power of 3304 MWt. Therefore, the results of the case analyzed without pressurizer pressure control available are still applicable and the complete loss of load presents no hazard to the integrity of the RCS or the MS system pressure boundary.

II.3.3 Loss of Normal Feedwater Flow (UFSAR Section 14.1.9) and Loss of All AC Power to the Plant Auxiliaries (UFSAR Section 14.1.12)

Both the loss of all AC power to plant auxiliaries and loss of normal feedwater analyses model a 2 percent power uncertainty in the NSSS thermal power. Since the core power level of 3457 MWt assumed in the current analyses is greater than the proposed uprated power, the results of these analyses are still applicable. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

The natural circulation cooldown capability is not affected because the T_{hot} and $T_{cold,}$, and therefore the no-load ΔT between the RCS and the steam generator values for the MUR Uprate Program are bounded by the current analysis. The loss of offsite power event, which credits the natural circulation process, was analyzed with 2 percent power measurement uncertainty, which bounds the 1.66 percent power uprate. Therefore, the uprate does not impact the conclusion of the analysis of the natural circulation capability stating that the RCS has demonstrated sufficient heat removal capability following Reactor Coolant Pump (RCP) coastdown to prevent fuel or clad damage and is bounded by the current analysis.

II.3.4 Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (UFSAR Section 14.2.6)

The HFP analysis is performed at 102 percent of licensed core power. As such, the increase in core power, combined with the reduction in the power uncertainty, is bounded by the current assumption in the analysis. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.5 RCCA Misalignment (UFSAR Section 14.1.3) and RCCA Drop (UFSAR Section 14.1.4)

The dropped RCCA transients (including the dropped RCCA bank) were previously analyzed using the methodology described in WCAP-11394-P-A (Reference II.3.1) and were reviewed to demonstrate that the DNB design basis is met.

The methodology described in WCAP-11394-P-A involves the use of generic statepoints for the dropped rod event. Sensitivity studies on the effect of a power increase on the generic statepoints were previously performed for a Westinghouse four-loop plant. The studies quantified the effect of an approximate 5 percent increase in power on the four-loop generic statepoints, and found that the statepoints were still applicable for use at the uprated conditions. Since the uprating is much smaller (1.66 percent) than the uprate (approximately 5 percent) used in the sensitivity studies, the generic statepoints also continue to be applicable. Although the statepoints are unaffected, the increase in nominal heat flux must be addressed with respect to the calculated DNBR. An evaluation of the DNB design basis using the generic statepoints and increased nominal heat flux confirmed that the DNB design basis continues to be met. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.6 Partial and Complete Loss of Forced Reactor Coolant Flow (UFSAR Section 14.1.6)

Since the 1.66 percent increase in core power could adversely affect the minimum DNBR, an evaluation was completed for this event. The evaluation concluded that the existing statepoints for the limiting complete loss of flow event remain valid, with the exception of the nominal core heat flux. The nominal core heat flux increases due to the 1.66 percent power uprating. The statepoints must therefore be applied to a higher nominal heat flux.

Revised statepoints that include the increased nominal heat flux were evaluated with respect to the DNBR. The evaluation showed that the DNB design basis is satisfied. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.7 Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition (RWFS) (UFSAR Section 14.1.1)

By definition, since the RWFS event occurs from a subcritical core condition with the RCS at no-load temperature conditions, this event is not affected by an increase in the reactor full power level and thus was not re-analyzed for the 1.66 percent power uprate.

The initial power increase that results from the rod withdrawal is terminated by reactivity feedback, not rod insertion. The power increases to its peak and is rapidly decreasing by the time the rods begin to drop. Since the rod drop time is essentially unaffected, and the initial nuclear power transient is defined by reactivity feedback, the total duration of energy addition would be almost identical. Therefore, the subsequent fuel rod heat flux increase resulting from the energy addition would also be insignificantly different.

The existing statepoints for the RWFS event remain valid, with the exception of the nominal core heat flux. The nominal core heat flux increases due to the 1.66 percent power uprating. The power statepoints, which are fractions of the initial value, must therefore be applied to a higher nominal core heat flux.

Revised statepoints that include the increased nominal core heat flux and flow asymmetry penalty were evaluated with respect to DNBR. The evaluation showed that the DNB design basis is satisfied. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.8 CVCS Malfunction (UFSAR Section 14.1.5)

An evaluation of the Mode 1 and 2 analysis was performed and showed that the 1.66 percent power increase has an insignificant impact on the automatic reactor trip time assumed in the analysis. Since the reactor trip time assumed in the analysis is still valid, the results of the Mode 1 and 2 analysis also remain valid. With respect to the Modes 2 through 6 analyses, the increase in power does not affect the results of these analyses, since the reactor is not at full power. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.9 Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 14.1.10)

An increase in feedwater flow can be caused by a failure in the feedwater control system that leads to the simultaneous full opening of the feedwater control valves. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-side temperature, and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient (MTC).

Transients initiated by increases in feedwater flow are attenuated by the thermal capacity of the primary and secondary sides. If the increase in reactor power is large enough, the primary Reactor Protection System (RPS) trip functions (e.g., high neutron flux, OTΔT, OPΔT) will prevent any power increase that can lead to a DNBR less than the safety analysis limit value. The RPS trip functions may not actuate, if the increase in power is not large enough.

The feedwater system malfunction that causes a reduction in feedwater temperature continues to be bounded by the excessive increase in secondary steam flow event and was not re-analyzed for the MUR Uprate Program.

The maximum feedwater flow considered to one or more steam generators corresponds to a control system malfunction that causes the feedwater control valves to fail in the full-open position. Cases with and without automatic rod control initiated at HFP conditions were re-analyzed in support of the MUR Uprate Program and are discussed in Section III.1. The licensing-basis analysis also addresses cases that are initiated at HZP conditions, but are not affected by an increase in the nominal full-power rating. Therefore, the conclusions of the feedwater malfunction analysis at HZP conditions continue to remain valid for the MUR Uprate Program.

II.3.10 Excessive Load Increase Incident (UFSAR Section 14.1.11)

This transient was evaluated by comparing plant conditions, conservatively bounding deviations in core power, average coolant temperature, and RCS pressure, to conditions corresponding to those required to exceed the core thermal limits. The evaluation concluded that there is sufficient margin to the core thermal operating limits in each case considered. Since the core thermal limits are not challenged, the minimum DNBR remains above the limiting value for all cases. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.11 Rupture of a Steam Pipe - Core Response Analysis (UFSAR Section 14.2.5)

The postulated rupture of a steam pipe is analyzed at HZP conditions to demonstrate that any return to power resulting from the uncontrolled steam release does not result in a violation of the DNB design basis. Because the rupture of a steam pipe is analyzed at shutdown conditions, the increase in nominal core power does not impact this analysis. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.12 Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (UFSAR Section 14.2.6)

This HZP analysis is unaffected, since it is performed at shutdown conditions. A change in the licensed full power value does not change the results. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.13 Anticipated Transients Without Scram

For Westinghouse-designed PWRs, the licensing requirements pertaining to ATWS are those specified in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants." The requirement set forth in 10 CFR 50.62(c) is that all Westinghouse-designed PWRs must install ATWS Mitigation System Actuation Circuitry (AMSAC). In compliance with 10 CFR 50.62(c), AMSAC has been installed and implemented at CNP Unit 1.

As documented in SECY-83-293 (Reference II.3.2), the analytical bases for the Final ATWS Rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. These generic ATWS analyses were performed based on the guidelines provided in NUREG-0460 (Reference II.3.3), and were transmitted to the NRC via Westinghouse letter NS-TMA-2182 (Reference II.3.4). The generic ATWS analyses applicable to CNP Unit 1 are provided for a Westinghouse 4-loop PWR with Model 51 steam generators, modeling NSSS power of 3423 MWt. For this plant configuration, the peak RCS pressure for the limiting loss of load ATWS event is 2974 psia, thereby resulting in 226 psi margin to the peak RCS limit of 3200 psia.

The combined effect of a 3.2 percent lower reactor power for the MUR Uprate Program conditions, additional available pressurizer power operated relief valve (PORV) and a reduced AFWS system capacity for CNP Unit 1 versus the generic ATWS analysis results in an overall peak RCS pressure benefit of 167 psi relative to the peak RCS pressure of 2974 psia reported in Reference II.3.4. This results in a net peak RCS pressure of 2807 psia (i.e., 2974 psia – 167 psia), or a margin to the ATWS peak RCS pressure limit of 3200 psia of 393 psi (i.e., 3200 psia – 2807 psia).

As prescribed by NUREG-0460, the 1979 generic ATWS analyses for Westinghouse PWRs assumes a full power MTC of -8 percent millirho per degree Fahrenheit (pcm/°F). A sensitivity analysis including the use of an MTC of –7 pcm/°F was also provided as prescribed by NUREG-0460. At that time, the MTC values of –7 pcm/°F and –8 pcm/°F represented MTCs that were bounding for Westinghouse PWRs over 99 percent and 95 percent of the cycle, respectively. The base case of 95 percent represents a 95 percent confidence limit on favorable MTC for the fuel cycle. Unit 1 currently operates with a slightly more positive full power MTC at beginning of life than that assumed in the 1979 generic analysis. The beginning of life MTC is –5.2 pcm/°F.

The MTC corresponding to the peak RCS pressure limit of 3200 psia calculated for this plant configuration is –5.5 pcm/°F. As noted earlier, the generic ATWS analysis (NS-TMA-2182) calculated a limiting peak RCS pressure of 2974 psia when modeling a full-power MTC of -8 pcm/°F. This equates to an increase in peak RCS pressure of 226 psi (3200 psia – 2974 psia) when the MTC is increased 2.5 pcm/°F (from -8 pcm/°F to –5.5pcm/°F). Based on this sensitivity, CNP Unit 1 having a 2.8 pcm/°F higher MTC than assumed in the 1979 generic ATWS analysis (from –8 pcm/°F to –5.2 pcm/°F) will proportionally increase the peak RCS pressure by 256 psia.

Considering the ATWS peak RCS pressure margin of 393 psi at the MUR Uprate Program conditions, CNP Unit 1 continues to have sufficient margin to the peak RCS pressure limit of 3200 psia.

Based on the above, it is concluded that operation of CNP Unit 1 at an uprated core power of 3304 MWt remains within the bounds of the generic Westinghouse ATWS analysis documented in NS-TMA-2182 and, therefore, will remain in compliance with the requirements of 10 CFR 50.62(c). Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

II.3.14 Station Blackout

A review was performed to determine if the current licensing basis for an SBO event remains bounding for the MUR Uprate Program.

The condensate inventory required for decay heat removal is bounded by the original 102 percent analysis. There are no changes to 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 instrument air are associated with this 1.66 percent power uprate. The main steam temperature remains the same and no other new heat loads are added; therefore, there is no impact on the loss of ventilation in the dominant areas of concern. The 1.66 percent power uprate will not impact the current containment isolation evaluations. Finally, the 1.66 percent power uprate will not impact the three areas of concern in the RCS inventory evaluations: RCP seal leak rates, the normal TS leak rates from the RCS, or letdown isolation capabilities. The ability of CNP Unit 1 to respond to an SBO event will not be impacted by the Unit 1 MUR Uprate Program. Therefore, the components required to cope with the SBO will not be impacted by the 1.66 percent power uprate, and the current licensing basis remains bounding for the MUR Uprate Program.

II.3.15 Flooding

Protection from flooding is afforded by features of CNP Unit 1 that are not affected by the changes associated with the 1.66 percent power uprate. These features include: the physical relationship between the plant grade and lake elevation, condenser circulating water pump and piping location, the essential service water pump and piping location, and site building construction.

In addition, the 1.66 percent power uprate does not affect the features associated with leakage detection and isolation, or the frequency of natural events such as seiche. No piping configuration or pump modifications for the CNP Unit 1 water systems are necessitated by the 1.66 percent power uprate. Therefore, the leakage conditions with the maximum flood potential (i.e., pipe break with pump runout) for the high volume water systems (essential and non-essential service water, circulating water, component cooling water, fire protection water, etc.) are not impacted by the proposed power uprate. Therefore, the changes associated with the MUR Uprate Program do not impact flooding.

References (Section II.3)

II.3.1. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990

II.3.2. SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," W. J. Dircks, dated July 19, 1983

II.3.3. NUREG-0460, "Anticipated Transients Without Scram for Light-Water Reactors," dated December 1978

II.3.4. Letter from T. M. Anderson, Westinghouse, to S. H. Hanauer, NRC, "ATWS Submittal," submittal number NS-TMA-2182, dated December 30, 1979

II.4 Design Transients

II.4.1 Nuclear Steam Supply System Design Transients

The basis for the NSSS design transient definitions is the analytical work performed for the CNP Unit 1 and 2 3600 MWt Rerating, and was documented in WCAP-12135, Appendix III (Reference II.4.6). This work, which is referred to as the Rerating Program, was done in support of WCAP-11902, including Supplement 1 (References II.4.1 and II.4.2). The Rerating Program analytical work was submitted to the NRC via Reference II.4.3, and approved in an SER dated June 9, 1989 (Reference II.4.4).

A comparison of the plant operating conditions used in the Rerating Program against the operating conditions for the MUR Uprate Program is shown in Table II-2.

Table II-2 -- Comparison of MUR Uprating Conditions with Values used in Design Basis Design Transients

	Unit 1 MUR Uprating (from Table 3)		Rerating Program (References II.4.2 and II.4.6)	
Parameter	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
Reactor Thermal Power, MWt	3315 ***	3315 ***	3588	3588
NSSS Thermal Power, MWt	3327	3327	3600	3600
RCS Flow, gpm/loop	83,200	83,200	88,500	88,500
RCS pressure, psia	2250/2100	2250/2100	2250/2000	2250/2000
T_{hot}, °F	609.1	588.2	616.9	582.3
T_{avg}, °F	575.4	553.7	583.1****	547.0
T SG outlet, °F	541.5	518.9	549.3	511.7
Steam temperature, °F *	513.1	489.4	521.0	481.8
Steam pressure, psia *	765	618	819.7	575.8
Feedwater temperature, °F **	437.4	437.4	435	435

* Unit 1 MUR Uprate Program values for limiting 30 percent steam generator tube plugging condition; Rerating Program values for 15 percent steam generator tube plugging condition.

** 435°F is listed in table at beginning of the design transient description in Reference II.4.2, but a review of the analyses and the transient figures indicates 440°F was used.

*** The reactor thermal power assumed in the NSSS design basis transient analyses (3315 MWt) conservatively bounds the value requested by this license amendment request (3304 MWt).

**** Higher T_{avg} value used than the high-end T_{avg} window for the Rerate Program, which was 581.3°F.

The following can be noted from Table II-2:

- The maximum T_{hot}, minimum steam generator outlet temperature, and minimum steam temperature used in the Rerating Program analyses bound the values for the MUR power uprating.

- The 3600 MWt NSSS power level used in the Rerating Program analyses would result in more severe transient parameter variations than would be the case for the MUR Uprate Program power level.

- The MUR Uprate Program for Unit 1 has a higher feedwater temperature (by 2.4°F) than the design value used in the Rerating Program. However, the design transients in the Rerating Program were analyzed with a feedwater temperature (440°F) that was higher than the value for the MUR Uprate Program (437.4°F). This higher value is conservative in regards to the feedwater temperature variation for any design transient.

Based upon this comparison, the existing design transients, as included in the Rerating Program, remain bounding and applicable for the Unit 1 MUR Uprate Program. However, the design transients for Unit 1 were revised as part of the effort to increase the SGTP limit to 30 percent (Reference II.4.7). The revised design transients reflect a 3 percent pressurizer safety valve tolerance. This revised the RCS pressure and pressurizer pressure transient responses for the loss of load and loss of power design transients. No other design transient changes were made. The analyses that supported the increased SGTP limit of 30 percent were approved by License Amendment No. 214 (Reference II.4.8).

Unit 1 presently operates with Babcock and Wilcox International (BWI) RSGs. Reference II.4.5 performed an analysis of the consequences of the implementation of the BWI steam generators on the existing safety analyses. While a direct evaluation on the continued applicability of the Reference II.4.6 design transients was not done, the certified design specification for the RSGs remained the same with respect to design transients. It was concluded that the RSGs will perform similarly to the Westinghouse Model 51 steam generators used in the Reference II.4.6 analyses. Therefore, the applicable design transients from References II.4.6 and II.4.5 are valid for the RSGs. The existing Reference II.4.6 design transient set, along with the revised Unit 1 RCS pressure and pressurizer pressure responses for the Reference II.4.5 loss of load and loss of power design transients, are the applicable design transients for the Unit 1 MUR Uprate Program. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

The design limit for the primary-to-secondary pressure differential is 1600 psid. To avoid violating this limit, when the full-power steam pressure is 679 psia or less, the pressurizer pressure setpoint must be reduced to 2100 psia. For steam pressures above 679 psia, the pressurizer pressure setpoint can be either 2100 psia or 2250 psia. CNP Unit 1 currently operates with a pressurizer set point of 2100 psia.

II.4.2 Auxiliary Equipment Design Transients

The review of the NSSS auxiliary equipment design transients was based on a comparison between the NSSS design parameters for the MUR Uprate Program described in Table 3 and the NSSS design parameters that make up the current auxiliary equipment design transients.

A review of the current auxiliary equipment transients determined that the only transients that could be potentially impacted by the MUR Uprate Program are those temperature transients that are impacted by the full-load NSSS design temperatures. These transients are currently based on an assumed full-load NSSS worst-case T_{cold} of 560°F and T_{hot} of 630°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS design temperatures.

A comparison of the MUR Uprate Program NSSS T_{cold} and T_{hot} design temperature ranges (519.2°F to 541.7°F and 588.2° to 609.1°F, respectively, from Table 3) with the T_{cold} and T_{hot} values used to develop the current design transients indicates that the MUR Uprate Program design temperature ranges are less than the values assumed to develop the design transients. Therefore, the MUR Uprate Program temperature transients are bounded by the current design temperature transients.

As the temperature transients dictated by the MUR Uprate Program conditions are less limiting than those that established the current auxiliary equipment design transients, I&M concludes that all of the applicable auxiliary equipment design transients for CNP Unit 1 still apply for the MUR Uprate Program. Therefore, the current licensing basis remains bounding for the 1.66 percent power uprate.

References (Section II.4)

II.4.1. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," dated October 1988

II.4.2. WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report," dated September 1989

II.4.3. Letter from M. P. Alexich, I&M, to T. E. Murley, NRC, "Reduced Temperature and Pressure Program Analyses and Technical Specification Changes," AEP:NRC:1067, dated October 14, 1988

II.4.4. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

II.4.5. WCAP-15608, Rev. 1, "D. C. Cook Unit 1 Replacement Steam Generator Safety Analysis Program Engineering Report," dated March 2001

II.4.6. WCAP-12135, "Donald C. Cook Units 1 and 2 Rerating Engineering Report," dated September 1989, Appendix III, "Cook Nuclear Plant Units 1 and 2 Rerating NSSS Design Transients"

II.4.7. WCAP-14285, "Donald C. Cook Nuclear Power Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," dated May 1995

II.4.8. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC Nos. M92587 AND M92588)," dated March 13, 1997

III. Accidents and Transients for which the Existing Analyses of Record do not Bound Plant Operation at the Proposed Uprated Power Level

Three non-LOCA events were identified for which the existing analyses of record do not bound plant operation following the 1.66 percent power uprating. The current analysis of record for the three events correspond to those implemented as part of the March 2001 BWI Series 51 replacement steam generator effort (RSG Program) at a core power level of 3250 MWt. These three events have been re-analyzed at 3315 MWt in support of this MUR Uprate Program. Each of these re-analyses specifically models the increased power level. Table III-1 provides a reference to the UFSAR analysis as well as the power level assumed in the re-analysis. Details of these evaluations are presented in subsequent sub-sections.

Table III-1 Re-Analyzed Accident Analysis Design Basis Events

	Accident/Transient	UFSAR Section	Assumed Core Power Level
III.1	Excessive Heat Removal Due to Feedwater System Malfunctions – full power cases	14.1.10	3315 MWt
III.2	Loss of External Electrical Load – DNB case	14.1.8	3315 MWt
III.3	Uncontrolled Rod Cluster Control Assembly Withdrawal at Power	14.1.2	3315 MWt 1989 MWt 332 MWt

III.1 Excessive Heat Removal Due to Feedwater System Malfunctions (full power cases) (UFSAR Section 14.1.10)

This event results from an increase in primary-to-secondary heat transfer caused by an increase in feedwater flow, that can result in the primary-side temperature and pressure decreasing significantly. The negative moderator and fuel temperature reactivity coefficients, and the actions initiated by the reactor rod control system can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of an RPS reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits.

An increase in feedwater flow can be caused by a failure in the feedwater control system that leads to the simultaneous full opening of the feedwater control valves. At power, this excess flow causes a greater load demand on the primary side due to increased subcooling in the steam generator.

Transients initiated by increases in feedwater flow are attenuated by the thermal capacity of the primary and secondary sides. If the increase in reactor power is large enough, the primary RPS trip functions (e.g., high neutron flux, OTΔT, OPΔT) will prevent any power increase that can lead to a DNBR less than the safety analysis limit value. The RPS trip functions may not actuate, if the increase in power is not large enough.

The analysis presented herein is for a feedwater system malfunction that causes an increase in feedwater flow event as discussed in UFSAR Section 14.1.10. The feedwater system malfunction that causes a reduction in feedwater temperature continues to be bounded by the excessive increase in secondary steam flow event and was not re-analyzed for the MUR Uprate Program.

The maximum feedwater flow to one or more steam generators due to a control system malfunction that causes the feedwater control valves to fail in the full-open position is assumed. Cases with and without automatic rod control initiated at hot full-power conditions were considered in support of the MUR Uprate Program.

The results of the analysis show that the minimum DNBR calculated is above the safety analysis limit value for an excessive feedwater addition at power. Therefore, the DNB design basis is met. Though bounded by the complete loss of load analysis documented in Section III.2, the RCS and MS system overpressure criteria are also met.

Table III-2 -- Excessive Heat Removal Due to Feedwater System Malfunctions (full power cases) (UFSAR Section 14.1.10)	
Key Inputs	• Initiating event: accidental opening of one or more feedwater control valves with the reactor at full power. This results in a feedwater flow increase to 200 percent of nominal flow to one or all steam generators. • Four cases were examined: Two cases assume full opening of one feedwater control valve with the rod control system in automatic and manual control and two cases assume full opening of all feedwater control valves with the rod control system in automatic and manual control. • Initial steam generator water level was minimized at the value that corresponds to the nominal level [43.8 percent narrow range span (NRS)] minus 10 percent. • The high-high steam generator water level turbine trip setpoint was conservatively maximized at 82 percent NRS. • Most-negative moderator and Doppler temperature coefficients • Least-negative Doppler power defect

<table>
<tr><th colspan="2">Table III-2 -- Excessive Heat Removal Due to Feedwater System Malfunctions
(full power cases)
(UFSAR Section 14.1.10)</th></tr>
<tr><td>Methodology</td><td>The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the BWI-Series 51 steam generators. As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-A "LOFTRAN Code Description," T. W. T. Burnett, April 1984 (Reference III.1.1) was used. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 (Reference III.1.2) was applied. The DNB methodology described in WCAP-11397-P-A, "Revised Thermal Design Procedure," Friedland, A. J. et al., April 1989, was applied.</td></tr>
<tr><td>Safety Analysis Limits</td><td>The minimum DNBR safety analysis limit is 1.42 for the MUR Uprate Program, corresponding to the WRB-1 DNBR correlation. The DNBR safety analysis limit and DNBR correlation have been maintained from the RSG program.</td></tr>
<tr><td>Calculated Results</td><td>The minimum DNBR values calculated using LOFTRAN for the four cases are listed as follows:
Single Loop Feedwater Malfunction
Automatic rod control 1.807 (1.66 percent uprate)
1.922 (RSG)
Manual rod control 1.806 (1.66 percent uprate)
1.919 (RSG)
Multi-Loop Feedwater Malfunction
Automatic rod control 1.723 (1.66 percent uprate)
1.867 (RSG)
Manual rod control 1.726 (1.66 percent uprate)
1.842 (RSG)</td></tr>
</table>

Since all applicable acceptance criteria have been satisfied, the failure of any of the feedwater control valves will not challenge the RCS and MS system pressure boundaries, nor will the integrity of the fuel cladding be compromised due to DNB.

References (Section III.1)

III.1.1. WCAP-7907-A, "LOFTRAN Code Description," dated April 1984

III.1.2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985

III.2 Loss of External Electrical Load - DNB Case (UFSAR Section 14.1.8)

This event was analyzed as a complete loss of load from full power conditions without a direct reactor trip.

The analysis conservatively assumes that the reactor trip is actuated by the RPS, and not by the turbine trip signal. This assumption is made because the UFSAR analysis is performed to show that the RPS signals are capable of providing a reactor trip in sufficient time following the event initiation to satisfy the acceptance criteria for the event.

For this event, the reactor may be tripped by any of the following RPS trip signals:

- OTΔT
- OPΔT
- Pressurizer high pressure
- Low-Low steam generator water level
- Direct reactor trip on turbine trip

In the event that the steam dump valves fail to open following a large loss of load, the sudden reduction in steam flow results in an increase in pressure and temperature in the steam generator secondary side. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise. This causes coolant expansion, a pressurizer insurge, and a rise in RCS pressure. Throughout the event, power is available for the continued operation of plant components, such as the RCPs.

Unless the transient RCS response to the complete loss of load event is terminated by manual or automatic action, the resultant reactor coolant temperature rise could eventually result in DNB and/or the resultant pressure increases could challenge the integrity of the RCS or MS system pressure boundaries. To avoid the potential damage that might otherwise result from this event, the RPS is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value and before the RCS and/or MS system pressures exceed the values at which the integrity of the pressure boundaries would be jeopardized.

The major challenges associated with the complete loss of load are over-pressurization of the RCS and MS system, and possible fuel cladding damage resulting from the increase in RCS temperature.

The transient responses for a complete loss of load from full-power conditions are presented in the UFSAR as four cases; two cases with automatic pressurizer pressure control evaluating minimum and maximum reactivity feedback conditions, and two without automatic pressurizer pressure control evaluating minimum and maximum reactivity feedback conditions.

Only the cases in which automatic pressurizer pressure control is assumed available (cases in which the DNB design basis are examined) were re-analyzed for the MUR Uprate Program. The cases in the licensing basis analyses in which automatic pressurizer pressure control is unavailable were evaluated, as discussed in Section II.3.2.

The results of these re-analysis of the cases analyzed with automatic pressurizer pressure control available demonstrate that the fuel design limits are maintained by the RPS, since the DNBR is maintained above the safety analysis limit value. Therefore, all ANS Condition II acceptance criteria are satisfied.

Table III-3 -- Loss of External Electrical Load (UFSAR Section 14.1.8)	
Key Inputs	• Initiating event: Complete loss of load at 100 percent full power operation. • Two cases were examined with automatic pressurizer pressure control evaluating minimum and maximum feedback conditions • The low-low steam generator water level trip setpoint was conservatively minimized at 0 percent NRS. • Initial operating conditions: Nominal conditions for reactor power, pressure, and RCS temperatures are assumed for statistical DNB analyses. The initial steam generator water level was set to the nominal level (43.8 percent NRS minus 10 percent mass (for modeling conservatism)). • Moderator and Doppler Coefficients of Reactivity: The loss of load is analyzed with both maximum and minimum reactivity feedback. The maximum feedback cases assume a large negative moderator coefficient and the most negative Doppler power coefficient. The minimum feedback cases assume a positive moderator temperature coefficient and the least negative Doppler coefficients. • Reactor Control: It is conservative to assume that the reactor is in manual rod control. If the reactor was in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient. • Pressurizer Spray and Power-Operated Relief Valves: Two cases for both the minimum and maximum moderator feedback cases are analyzed: • Full credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure. Safety valves are also available. • Steam Release: No credit is taken for the operation of the steam dump system or steam generator PORVs. The steam generator pressure rises to the safety valve setpoint where steam release through the safety valves limits the secondary steam pressure.

Table III-3 -- Loss of External Electrical Load (UFSAR Section 14.1.8)	
	• Feedwater Flow: Main feedwater flow to the steam generators is assumed to be lost at the time of the complete loss of load. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization. • Reactor trip is actuated by the first RPS trip setpoint reached. Trip signals are expected due to high pressurizer pressure, OTΔT, OPΔT, and low-low steam generator water level.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the BWI Series 51 replacement steam generators. As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-A "LOFTRAN Code Description," Burnett, T. W. T. et al., April 1984 (Reference III.2.1) was used. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 (Reference III.2.2) was applied. The DNB methodology described in WCAP-11397-P-A, "Revised Thermal Design Procedure," Friedland, A. J. et al., April 1989, was applied.
Safety Analysis Limits	The minimum DNBR safety analysis limit is 1.42 for the MUR Uprate Program, corresponding to the WRB-1 DNBR correlation. The DNBR safety analysis limit and DNBR correlation have been maintained from the RSG program.
Calculated Results	The minimum DNBR value calculated using LOFTRAN is 1.591 (1.66 percent uprate) versus 1.737 for the RSG Program.

Since all applicable acceptance criteria have been satisfied, a complete loss of load will not challenge the RC and MS system pressure boundaries, nor will the integrity of the fuel cladding be compromised due to DNB for the MUR power uprate conditions.

References (Section III.2)

III.2.1 WCAP-7907-A, "LOFTRAN Code Description," dated April 1984

III.2.2 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985

III.3 Uncontrolled RCCA Bank Withdrawal at Power (UFSAR Section 14.1.2)

An uncontrolled RCCA bank withdrawal at power which causes an increase in core heat flux may result from faulty operator action or a malfunction in the rod control system. Immediately following the initiation of the accident, the steam generator heat removal rate lags the core power generation rates, until the steam generator pressure reaches the setpoint of the steam generator relief or safety valves. This imbalance between heat removal and heat generation rates causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in DNB and/or fuel centerline melt. Therefore, to avoid damage to the core, the RPS is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value or the fuel rod linear heat generation rate (kW/ft) limit is exceeded.

The automatic features of the RPS that prevent core damage in an RCCA bank withdrawal incident at-power include the following:

1. Nuclear power range instrumentation actuates a reactor trip on high neutron flux if two-out-of-four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an OTΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an OPΔT setpoint. This setpoint is automatically varied with coolant average temperature to ensure that the allowable fuel power rating is not exceeded.

4. A high pressure reactor trip, actuated from any two-out-of-four pressurizer pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip, actuated from any two-out-of-three level channels, is set at a fixed point.

The high neutron flux, OTΔT, and high pressure reactor trip functions provide adequate protection over the entire range of potential reactivity insertion rates. The minimum value of DNBR is maintained above the safety analysis limit value, the peak MS system pressure is maintained below 110 percent of the design pressure and the peak core average heat flux is maintained below the 118 percent limit. The uncontrolled RCCA bank withdrawal at power analysis described in UFSAR Section 14.1.2 remains bounding for the MUR Uprate Program with respect to peak RCS pressure. A summary of the re-analysis is provided in the table below:

Table III-4 -- Uncontrolled RCCA Bank Withdrawal at Power (UFSAR Section 14.1.2)	
Key Inputs	• Initiating event: RCCA bank withdrawal • A spectrum of reactivity insertion rates ranging from 0.6 pcm/sec to 100 pcm/sec were examined at 10 percent, 60 percent and 100 percent of nominal power in order to demonstrate that the applicable acceptance criteria, primarily the minimum DNBR safety analysis limit, are satisfied over a wide range of conditions. • Both maximum and minimum reactivity feedback conditions were examined. • A conservatively high OTΔT reactor protection setpoint was assumed [K1 (constant term in OTΔT setpoint equation) = 1.35]. The OTΔT reactor protection setpoint has been maintained from the BWI-Series 51 RSG program. • A conservatively high neutron flux reactor protection setpoint of 118 percent of uprated RTP was assumed.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the BWI-Series 51 RSGs. As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-A, "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984 (Reference III.3.1) was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 (Reference III.3.2) was applied. The DNB methodology described in WCAP-11397-P-A, "Revised Thermal Design Procedure," Friedland, A. J. et al., April 1989 (Reference III.3.3) was applied.

Table III-4 -- Uncontrolled RCCA Bank Withdrawal at Power (UFSAR Section 14.1.2)	
Safety Analysis Limits	The minimum DNBR safety analysis limit is 1.42 for the MUR Uprate Program, corresponding to the WRB-1 DNBR correlation. The DNBR safety analysis limit and DNBR correlation have been maintained from the BWI-Series 51 RSG program. The peak primary and secondary pressure limits are 110 percent of design pressure, or 2748.5 psia and 1208.5 psia, respectively. There is a 118 percent limit for peak core average heat flux to preclude fuel centerline melt.
Calculated Results	• For the MUR power uprate, the minimum DNBR calculated using LOFTRAN is 1.513 and corresponds to a case initiated from 100 percent power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 1.0 pcm/sec. • For the BWI-Series 51 RSG program, the minimum DNBR calculated using LOFTRAN is 1.660 and corresponds to a case initiated from 100 percent power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 0.6 pcm/sec. • The peak secondary pressure calculated for the MUR power uprate is 1159 psia. • The peak core average heat flux calculated for the MUR power uprate is 116.7 percent. • The peak primary pressure that was calculated previously and remains limiting for the MUR power uprate is 2747.9 psia.

Therefore, the results of the analysis show that an uncontrolled RCCA bank withdrawal at power does not adversely affect the core, the RCS or the main steam system and all applicable acceptance criteria are satisfied.

References (Section III.3)

III.3.1. WCAP-7907-A, "LOFTRAN Code Description," dated April 1984

III.3.2 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985

III.3.3 WCAP-11397-P-A, "Revised Thermal Design Procedure", dated April 1989

IV. Mechanical/Structural/Material Component Integrity and Design

IV.1 Reactor Vessel Structural Evaluation

The CNP Unit 1 reactor vessel was evaluated for impact due to the MUR Uprate Program. There is no change to any of the design inputs that were previously considered in the reactor vessel evaluations for the Rerating Program (WCAP-11902, including Supplement 1 [Reference IV.1.1]), which was approved in Unit 1 License Amendment No. 126 (Reference IV.1.2). Therefore, the MUR Uprate Program has no effect on the results in the CNP Unit 1 reactor vessel analytical report.

The Unit 1 reactor vessel continues to satisfy the applicable requirements of Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code, 1965 Edition through Winter 1966 Addenda, in accordance with the reactor vessel design requirements.

IV.1.1 Reactor Vessel Integrity-Neutron Irradiation

Reactor vessel integrity is affected by changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The reactor vessel integrity evaluation for the MUR Uprate Program included the following evaluations:

- Review of the reactor vessel surveillance capsule removal schedule to determine if changes are required as a result of changes in the vessel fluence due to the MUR Uprate Program.

- Review of the existing pressure-temperature (P-T) limit curves to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

- Review of the existing RT_{PTS} values to determine if the effects of the uprated fluence projections result in an increase in RT_{PTS} for the beltline materials in the Unit 1 reactor vessel at the end-of-license per TS Table 4.4-5 (EOL), 32 Effective Full Power Years (EFPY).

- Review the upper shelf energy (USE) values at EOL for all reactor vessel beltline materials in the Unit 1 reactor vessel to assess the impact of the uprated fluence projections.

The calculated fluences used in the MUR Uprate Program evaluation comply with Regulatory Guide (RG) 1.190 (Reference IV.1.3). As these calculations are performed on a plant-by-plant basis, there is no generic topical report for an approved method – the methodology used is that of RG 1.190.

The fluence projections associated with the 1.66 percent uprated condition, while considering actual power distributions incorporated to date, will exceed the current fluence projections used in the evaluations of record (References IV.1.5 and IV.1.6) (withdrawal schedules, emergency response guideline (ERG) category, pressurized thermal shock (PTS), and USE), which were approved by Unit 1 License Amendment No. 167 (Reference IV.1.4). The effect of the higher fluence values is minimal for PTS and has not changed the ERG limits. With respect to the P-T curves, the current TS curves are based on the latest capsule report, WCAP-12483 (Reference IV.1.5), which was submitted to the NRC in a license amendment request to change the heatup and cooldown curves for the first 32 EFPY (Reference IV.1.6). These P-T curves used fluences that were developed prior to the MUR Uprate Program. Therefore, the applicability date of the P-T curves in the Unit 1 TS must be updated to reflect the uprating. The revised adjusted reference temperature (ART) after the MUR Uprate Program will be more restrictive than that used in developing the current ART values at 32 EFPY. The new applicability date for the Unit 1 P-T curves (28.4 EFPY) is reflected in the proposed changes to the RCS Pressure-Temperature Heatup and Cooldown Limit Curves (TS Figures 3.4-2 and 3.4-3) of TS 3/4 4.9, "Pressure/Temperature Limits – Reactor Coolant System." The Reactor Vessel Material Irradiation Surveillance Schedule (TS Table 4.4-5) of TS 3/4 4.9 "Pressure/Temperature Limits – Reactor Coolant System" is revised so the Capsule S removal interval is "Standby." The Capsule U accumulated neutron fluence analysis (Reference IV.1.5) revealed that the criteria associated with the fourth and fifth withdrawal intervals has been satisfied. Thus, the Capsule S removal interval is changed to be consistent with the other remaining irradiation surveillance capsules (Capsules V, W, and Z). Corresponding changes to revise the Bases for TS 3/4 4.9 are included in this license amendment request.

It is concluded that the MUR Uprate Program for CNP Unit 1 will not have significant effect on the reactor vessel integrity.

IV.1.2 Reactor Internals

The reactor internals support the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. The changes in the RCS temperatures produce changes in the boundary conditions experienced by the reactor internals components. Also, increases in core power may increase nuclear heating rates in the lower core plate, upper core plate, and baffle-barrel former region. This section describes the analyses performed to demonstrate that the reactor internals can perform their intended design functions at the 1.66 percent power uprate conditions.

IV.1.2.1 Thermal-Hydraulic Systems Evaluations

A key area in the evaluation of core performance is the determination of the hydraulic behavior of the coolant flow and its effect within the reactor internals system. The core bypass flow is defined as the total amount of reactor coolant flow which bypasses the core region, and is not considered effective in the core heat transfer process. Consequently, the effect of increasing core bypass flow is a reduction in core power capability. The RCCA scram time is affected by the flow and temperature conditions. The hydraulic lift forces are critical in the assessment of the structural integrity of the reactor internals and hold-down spring functionality. Baffle plate gap momentum flux and fuel stability is affected by pressure differences between the core and baffle former region.

The results of these evaluations are discussed below.

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region. The principal core bypass flows are the barrel-baffle region, vessel head spray nozzles, vessel outlet nozzle gap, baffle plate core cavity gap, and the fuel assembly thimble tubes.

The design core bypass flow limit is 7.1 percent of the total reactor vessel flow. The effect of the MUR Uprate Program has an insignificant effect on the core bypass flow. Therefore, the current total design core bypass flow value of 7.1 percent remains bounding.

RCCA Drop Time

An evaluation was performed to demonstrate that the RCCA drop time is still within the current value of 2.4 seconds (required by the TS) for the revised design conditions. The revised design conditions for the RCCA drop time consist of the core power and the core inlet temperature (T_{cold}). The core power increased due to the 1.66 percent power uprate from 3250 MWt to 3304 MWt. The lowest core inlet temperature has remained unchanged at 519.2°F for the uprate condition. The effect of the increase in core power increased RCCA drop time less than 0.01 seconds. This change is considered negligible and the RCCA drop time will still be less than the TS limit of 2.4 seconds.

Hydraulic Lift Forces and Pressure Losses

The reactor internals hold-down spring is essentially a large Belleville-type spring of rectangular cross section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and the upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation was performed 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 results indicate that the downward force remains essentially unchanged, indicating that the reactor internals would remain seated and stable for the 1.66 percent power uprate conditions.

Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically-induced instability or vibration of fuel rods caused by a high velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the core. The baffle jetting phenomenon could lead to fuel cladding damage.

A number of experimental tests have been performed to study the interaction between baffle joint jetting and the response of the fuel rod. These tests indicated that there are two vibration levels that can result in fuel rod damage. Lower levels of vibration amplitude can inflict damage in the form of vibration wear at the rod/grid interface. Large amplitude vibration (whirling), caused by fluid elastic instability, can result in fuel rod damage due to cladding fatigue failure, rod-to-rod contact or even rod-to-baffle plate wall contact.

In order to guard against fuel rod failures from flow-induced vibration, the cross-flow emanating from baffle joint gaps must be limited to a specific momentum flux, V^2h; that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V^2. This momentum flux varies from point to point along the baffle plate due to changes in pressure differential across the plate and the local gap width variations. In addition, the modal response of the vibrating fuel rod must be considered. That is, a large value of local momentum flux impinging near a grid is much less effective in causing vibration than the same V^2h impinging near the mid-span of a fuel rod.

Baffle joint momentum flux is dependent upon the pressure differential across the baffle plate, the baffle-to-baffle gap width, and the modal response of the fuel assembly. Any increase in baffle joint momentum flux would require an increase in at least one of these. The pressure differential across the baffle plate remains unchanged due to the 1.66 percent power uprate, likewise the baffle gap width and fuel assembly modal response. Therefore, the baffle joint momentum flux will not change as a result of the 1.66 percent power uprate.

IV.1.2.2 Mechanical Evaluations

The 1.66 percent power uprate conditions do not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural effects from the seismic operating basis earthquake and safe-shutdown earthquake loads and the LOCA hydraulic and dynamic loads.

With regards to flow-induced vibration, the lowest vessel/core inlet coolant temperature remains unchanged. The corresponding vessel outlet coolant temperature increases 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 MUR uprate essentially remain the same. Therefore, it is reasonable to conclude that there is no significant impact on the performance of the reactor internals with regard to flow-induced vibration.

IV.1.2.3 Structural Evaluations

Evaluations were performed to demonstrate that the structural integrity of the reactor components is not adversely affected by the 1.66 percent power uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of various components.

The core support structure components affected by the MUR Uprate Program are discussed below. The primary inputs to the evaluations are the revised RCS temperatures (as indicated in Table 3) and the gamma heating rates. The gamma heating rates take into account the 1.66 percent increase in core power.

The reactor internal components subjected to heat generation effects (either directly or indirectly) are the upper core plate, the lower core plate, and the baffle-barrel region. For all of the reactor internal components, except the lower core plate and the upper core plate, the stresses and cumulative fatigue usage factors were unaffected by the 1.66 percent uprate conditions, because the previous analyses remain bounding.

<u>Lower Core Plate Structural Analysis</u>

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle-barrel region.

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 1.66 percent power uprate conditions.

<u>Baffle-Barrel Region Evaluations</u>

The baffle-barrel regions consist of a core barrel into which baffle plates are installed. They are supported by bolting interconnecting former plates to the baffle and core barrel.

The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads, as well as secondary loads consisting of preload, and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles also channel and direct coolant flow such that a coolable core geometry can be maintained.

The thermally-induced displacements of the baffle-former bolts for the 1.66 percent power uprate relative to the original design conditions were calculated for a bounding range of conditions. The results demonstrated that the 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 MUR Uprate Program.

Upper Core Plate Structural Analysis

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes. It serves as the transitioning member for the control rods for entry into and retraction from the fuel assemblies. It also controls coolant flow in its exit from the fuel assemblies and serves as a boundary between the core and the exit plenum. The upper core plate is restrained from vertical movement by the upper support columns, which are attached to the upper support plate assembly. The lateral movement is restrained by four equally spaced core plate alignment pins.

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 1.66 percent power uprate is small. The results show that the structural integrity of the upper core plate is maintained for the 1.66 percent power uprate conditions. 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 1.66 percent power uprate conditions.

References (Section IV.1)

IV.1.1 WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988 and WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Power Plant Units 1 and 2 Licensing Report," dated September 1989

IV.1.2 Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

IV.1.3 Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001

IV.1.4 Letter from J. F. Stang, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit 1 – Amendment No. 167 to Facility Operating License No. DPR-58 (TAC No. M71480 and M75260)," dated October 26, 1992

IV.1.5 WCAP-12483, "Analysis of Capsule U from the American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program," E. Terek, S. L. Anderson, L. Albertin and N. K. Ray, dated January 1990

IV.1.6 Letter from M. P. Alexich, I&M, to T. E. Murley, NRC, "Technical Specification Change Request Revised Heatup and Cooldown, and LTOP Setpoint for the First 32 Effective Full Power Years," AEP:NRC:0892O, dated October 29, 1990

IV.2 Piping and Supports

IV.2.1 NSSS Piping

Parameters associated with the MUR Uprate Program 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 MUR Uprate Program NSSS performance parameters (discussed in the "Introduction" Section, including Table 3) are bounded by the NSSS performance parameters from WCAP-11902, including Supplement 1 (Reference IV.2.1), and the existing design basis piping analyses are still applicable for the MUR Uprate Program. The equipment nozzle and support loads, and the piping stresses are not affected by the MUR Uprate Program.

The existing RCL LOCA analysis and RCL analysis with compartment pressures due to the main steam and feedwater breaks are not affected by the MUR Uprate Program because of the following:

- LOCA hydraulic forcing functions are unchanged (as discussed in Section II.1.1).

- Steam generator secondary side steam pressure and feedwater pressure for the MUR Uprate Program are bounded by the pressures used in the existing piping analyses. Therefore, the jet forces due to a main steam or feedwater nozzle break from the existing analysis are still applicable.

- Compartment pressures and mass and energy releases are not affected by the MUR Uprate Program (as discussed in Section II.2).

Since the existing piping analysis is still applicable, there are no changes in the steam generator or reactor coolant loop displacements, or the primary equipment nozzle and support loads due to the MUR Uprate Program.

The operating temperature window of the RCL due to the MUR Uprate Program is bounded by the existing operating temperature window. With the continued applicability of the existing design transients, the impact of the MUR Uprate Program on the NRC Bulletin 88-08 evaluation of the auxiliary spray piping and NRC Bulletin 88-11 evaluation of the pressurizer surge line piping is judged as insignificant.

Hence, with the continued applicability of the design transients and the insignificant changes due to the thermal analysis, the impact of the MUR Uprate Program on the Auxiliary Class 1 branch nozzle displacements from the deadweight, thermal, seismic, and LOCA analyses is negligible.

IV.2.2 RCL Support System

The steam generator, RCP, reactor vessel, and pressurizer supports have been qualified for piping and component loads resulting from the BWI RSG program. The RCS supports were shown to meet the allowable stresses for all loading combinations for the CNP Unit 1 BWI RSG program loads. Since the MUR Uprate Program does not significantly change the loads exerted upon the support structures, the supports will continue to be qualified for the 1.66 percent power uprate condition.

IV.2.3 Leak-Before-Break (LBB) Analysis

By References IV.2.2 and IV.2.4, the NRC approved CNP's use of the LBB methodology. The LBB analyses justified the elimination of large primary loop pipe rupture and pressurizer surge line pipe rupture from the structural design basis for the CNP Unit 1. To demonstrate the continued acceptability of the elimination of RCS primary loop pipe rupture and pressurizer surge line pipe rupture from the structural design basis for the MUR Uprate Program, the following objectives must be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate.

- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Demonstrate margin on applied load.
- Demonstrate that fatigue crack growth is negligible.

These objectives were met by the analyses discussed in References IV.2.2 and IV.2.3.

There is no change in loads on the primary loop piping due to the uprating parameters. The effect of material properties due to the changes in temperature will have a negligible impact on the existing LBB analysis margins. Additionally, there is no significant impact on loads in the pressurizer surge line LBB analysis due to the MUR Uprate Program.

Therefore, the existing LBB analyses conclusions that were approved by References IV.2.2 and IV.2.4 remain applicable for the Unit 1 MUR Uprate Program.

References (Section IV.2)

IV.2.1. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988 and WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Power Plant Units 1 and 2 Licensing Report," dated September 1989

IV.2.2. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Review of Leak-Before-Break for the Pressurizer Surge Line Piping as Provided by 10 CFR Part 50, Appendix A, GDC 4 (TAC Nos. MA7834 and MA7835)," dated November 8, 2000

IV.2.3. Letter from M. W. Rencheck, I&M, to Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2 – Request to Apply Leak Before Break (LBB) Methodology to the Pressurizer Surge Line," C0800-04, dated August 22, 2000

IV.2.4. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474), dated December 23, 1999

IV.3 Control Rod Drive Mechanisms (CRDM)

The CRDMs are subjected to 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). These are the only NSSS design parameters considered in the CRDM evaluation.

Higher temperatures are more limiting for the CRDM structural design qualification because it results in a decrease in the margin to the allowable design stress limits. The maximum T_{hot} from the MUR Uprate Program NSSS design parameters (Table 3) for any case is 609.1°F. Furthermore, the possible RCS operating pressure values continue to remain at either 2100 psia or 2250 psia for the MUR Uprate Program.

The CNP Unit 1 CRDMs were evaluated as part of the Rerating Program (References IV.3.1 and IV.3.2) for temperatures higher than the maximum temperature of 609.1°F associated with the MUR Uprate Program. Therefore, the evaluations performed remain bounding and applicable to the MUR Uprate Program.

References (Section IV.3)

IV.3.1. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," dated October 1988

IV.3.2. WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report," dated September 1989

IV.4 Reactor Coolant Pumps and Motors

The RCPs and RCP motors were evaluated to determine the impact of the revised RCS conditions to demonstrate that the RCP structural integrity is not adversely affected.

Reactor Coolant Pump

The RCPs are located between the steam generator outlet and reactor vessel inlet in the RCL. The maximum vessel inlet (RCP outlet) temperature is 541.7°F for the MUR Uprate Program conditions, as shown in Table 3. This temperature is lower than the vessel inlet temperature of 547°F used in the previous 3425 MWt Rerate evaluation (Reference IV.4.1), and therefore, represents a less limiting condition.

The revised pressure changes and temperature changes are less than those previously evaluated and are bounded for the MUR Uprate Program.

Reactor Coolant Pump Motor

The limiting design parameter for 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 uprating. The loads on the motor thrust bearings were also determined for the uprated conditions and determined to be acceptable. Based upon the evaluations of the RCP motors as described above, the motors are acceptable for the MUR Uprate Program conditions.

References (Section IV.4)

IV.4.1 WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," dated October 1988, and WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Power Plant Units 1 and 2 Licensing Report," dated September 1989.

IV.5 Steam Generators

Evaluations of the thermal-hydraulic performance, structural integrity, and steam generator tube wear have been performed to address operation at the MUR Uprate Program conditions.

IV.5.1 Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation of the BWI Series 51 steam generator focused on the changes to secondary-side operating characteristics at the 1.66 percent power uprate conditions. The following evaluations were performed to confirm the acceptability of the steam generator secondary-side parameters.

The original steam generators for CNP Unit 1 were Westinghouse Model 51 units. The original units were replaced with RSGs supplied by BWI. The operating characteristics of the RSGs were established through the use of the BWI thermal-hydraulic computer code. The results of the T/H calculations served as the design basis for the MUR Uprate Program evaluations.

The thermal-hydraulic performance evaluation of the RSG for uprated power was based on thermal design flow of the RCS. The RSG design basis included two calculations that were based on thermal design flow. One calculation was at the base design NSSS power level of 3264 MWt (100 percent load), and the other was at 3424 MWt (105 percent load). These results are the basis for the MUR Uprate Program evaluations.

Bundle Mixture Flow Rate

The product of the steam flow rate and the circulation ratio, which equals the bundle mixture flow rate, remains essentially the same after the 1.66 percent power uprate. Therefore, it was concluded that the proposed uprate has essentially no effect on the mixture flow in the tube bundle.

Steam Pressure

The steam pressure is affected by the reduction of the T_{avg}, and to a greater extent, by the SGTP level. As T_{avg} is reduced from 567.7° to 553.7°F and the SGTP is increased from 10 percent to 30 percent, the resulting steam pressure is reduced from 772 psia to 618 psia. This is below the 679 psia minimum steam pressure limit established by the generator design specification primary-to-secondary differential pressure limit of 1600 psi. The predicted low steam pressure indicates additional analyses may be required prior to operating at the increased power level, with 30 percent SGTP. However, the BWI RSGs are currently in the first cycle of operation with less than 0.03 percent tubes plugged (total of four tubes plugged in four steam generators with 3496 tubes each); therefore, the current configuration bounds the MUR Uprate Program.

Moisture Carryover

The separator combinations have undergone full-scale laboratory testing. The test pressures ranged from 950 to 750 psia. The test steam flow rate per separator unit varied from 10,000 pounds-mass per hour (lb_m/hr) to about 58,000 lb_m/hr. The test results revealed that the moisture carryover is not sensitive to water flow and pressure changes for all test conditions. The estimated moisture carryover for the 1.66 percent uprated power conditions is on the order of 0.020 percent. This indicates that the uprate will have essentially no effect on the moisture carryover of the RSG.

Two-Phase Flow Stability

The flow stability for the RSG design is based on a guideline, referred to as the "0.2 Instability Rule," regarding the two-phase flow stability in nuclear steam generators. This rule states that, for a circulation ratio equal to five or greater, if the single-phase pressure losses within the circulating loop are greater than 20 percent of the two-phase losses, the circulation is stable.

For the MUR Uprate Program, since the downcomer flow rate is essentially the same as in the base case, and the downcomer water density is also essentially unchanged, the single-phase pressure losses would be similar to that of the base case at approximately 2.02 psi. On the other hand, the two-phase pressure losses will be affected by the change in steam pressure. The two-phase pressure losses for four MUR Uprate Program cases were estimated. The pressure losses were estimated to be about 5.41 psi, 5.61 psi, 4.95 psi, and 5.17 psi for the four cases. The corresponding ratios of single-phase to two-phase pressure losses are all greater than 20 percent. On this basis, it was concluded that the operation of the RSG at the 1.66 percent power uprate conditions remains stable.

Secondary Side Pressure Drop, Liquid Mass, and Steam Mass

Based on the results of the 105 percent power uprate RSG evaluations, and from uprating analyses performed for other Westinghouse steam generators, it was concluded that these parameters remain acceptable for the 1.66 percent power uprate.

Thermal-Hydraulic Conclusions

The design basis for the RSGs considered the thermal-hydraulic performance of the generators at both the 100 percent and 105 percent of current power levels. This evaluation considered the thermal-hydraulic performance at an intermediate power level of 101.66 percent, using the design basis calculations as reference cases. It was concluded that all thermal-hydraulic performance parameters remain acceptable for operation at the 1.66 percent power uprate conditions. It was noted that the potential exists for steam pressure to fall below the minimum value during operation at uprated power, when SGTP approaches 30 percent. If steam generator tube plugging approaches these higher levels, additional analyses will be required to address the potential for reduced steam pressure. However, the BWI RSGs are currently in the first cycle of operation with less than 0.03 percent tubes plugged (total of four tubes plugged in four steam generators with 3496 tubes each); therefore, the current configuration bounds the MUR Uprate Program.

IV.5.2 Structural Integrity Evaluation

The structural evaluation focused on the critical steam generator components as determined by the stress ratios and fatigue usage. The following discussions address the evaluations of the primary-side and secondary-side components. Mechanical repair hardware was not evaluated for the CNP Unit 1 steam generators because they are new replacements with no installed repair hardware and minimal tube plugging (less than 0.03 percent SGTP).

Comparisons of previous CNP power uprate efforts were performed to determine if the results from these evaluations envelop the current MUR Uprate Program at CNP Unit 1.

The evaluations discussed in the paragraphs below were performed to confirm the acceptability of the critical primary and secondary side components when subjected to the uprated operation conditions defined by the NSSS design parameters in Table 3, and the applicable design transients discussed in Sections II and III.

Input Parameters and Assumptions

The Westinghouse Model 51 design steam generators originally installed in CNP Unit 1 were designed and analyzed to the specifications provided in the original plant design specifications for a 3264 MWt NSSS power rating. Subsequently, analyses were performed to assess the impact of uprated power conditions on the structural integrity of the affected components. The primary side components that were evaluated included the tubes, tube/tubesheet weld, tubesheet/shell junction, and divider plate. The secondary side components included the feedwater nozzle, secondary manway bolts and shell penetration, and the steam nozzle.

In 2000, portions of the original Westinghouse Model 51 steam generators were replaced with vertical shell and U-tube heat exchangers with integral moisture separating equipment. This combination was designated the BWI Series 51 replacement steam generator. The replacement steam generators were designed and analyzed for both the 3264 MWt NSSS thermal power conditions and a 3600 MWt uprated NSSS thermal power condition.

The design transient set for the MUR Uprate Program conditions is discussed in Section II. 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.

Structural Integrity Conclusions

Results of the analyses performed on the BWI Series 51 steam generators show that all steam generator components continue to meet ASME Boiler and Pressure Vessel Code, Section III, 1989 Edition, limits for the 1.66 percent 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.

IV.5.3 Tube Vibration and Wear

The impact of the MUR Uprate Program on the steam generator 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 steam generator resulting from the uprate. The effects of these changes on the fluidelastic instability ratio and amplitudes of tube vibration due to both vortex shedding and turbulence were addressed. In addition, the potential effect of the 1.66 percent power uprate on future tube wear was considered.

The analysis of the RSGs indicates that significant levels of tube vibration will not occur from either the fluidelastic, 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.

IV.5.4 Regulatory Guide 1.121 Analysis

The heat transfer area of steam generators in a PWR NSSS comprises over 50 percent of the total primary system pressure boundary. The steam generator tubing, therefore, represents a primary barrier against the release of radioactivity to the environment. For this reason, conservative design criteria have been established for the maintenance of tube structural integrity under the postulated design-basis accident condition loadings in accordance with Section III of the ASME Code.

Over a period of time, under the influence of the operating loads and environment in the steam generator, some tubes may become degraded in local areas. Partially degraded tubes are satisfactory for continued service provided that defined stress and leakage limits are satisfied, the prescribed structural limit is adjusted to take into account possible uncertainties in eddy current inspection, and an operational allowance for continued tube degradation until the next scheduled inspection is defined.

RG 1.121 (Reference IV.5.1) describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The level of acceptable degradation is referred to as the "repair limit".

An analysis was performed 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 allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit, a growth allowance for continued operation until the next scheduled inspection, and an allowance for eddy current measurement uncertainty. The operating parameters applicable to the MUR Uprate Program are defined in the Introduction section of this attachment (Table 3). Parameters are defined for "High T_{avg}" and "Low T_{avg}" conditions for both 0 percent and 30 percent SGTP levels. In addition, the plant may also operate at an RCS pressure of 2100 psia. The RG 1.121 analysis establishes minimum wall requirements for transient conditions corresponding to the 30 percent SGTP case, which envelopes the primary-to-secondary pressure gradients for the 0 percent SGTP condition. An evaluation was performed which demonstrated that the results of the RG 1.121 analysis are acceptable for the 1.66 percent power uprate.

References (Section IV.5)

IV.5.1. Regulatory Guide 1.121, "Bases for Plugged Degraded PWR Steam Generator Tubes (for Comment)," dated August 1976

IV.6 Pressurizer

A review of the revised temperature parameters presented in Table 3 showed that any changes in T_{hot} and T_{cold} are small, and are bounded by the existing pressurizer stress analysis performed for the CNP Unit 1 SGTP program conducted in 1995 (Reference IV.6.1). 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 MUR Uprate Program. Therefore, it is concluded that the revised parameters would not have any impact on the pressurizer stress and fatigue analysis and that the current evaluations remain valid.

It is concluded that the pressurizer components meet the stress/fatigue analysis requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition through 1996 Winter Addenda, for plant operation at the MUR uprate conditions.

References (Section IV.6)

IV.6.1 WCAP-14285, "Donald C. Cook Nuclear Power Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," dated May 1995

IV.7 NSSS Auxiliary Equipment

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. An evaluation was performed to determine the potential effect that the revised design conditions will have on the 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 MUR Uprate Program; therefore, these tanks are not affected by the MUR Uprate Program. Additionally, the MUR Uprate Program has no effect on the pressurizer relief tank or the volume control tank.

The revised design conditions have been evaluated with respect to the impact on the auxiliary heat exchangers, valves, pumps, and tanks. The results of this review 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.

IV.8 Fuel Evaluation

This section summarizes the evaluations performed to determine the effect of the MUR Uprate Program on the nuclear fuel. In general, the fuel evaluation for CNP Unit 1 is performed for each specific fuel cycle and varies according to the needs and specifications for each cycle, consistent with WCAP-9272-P-A (Reference IV.8.1). However, some fuel-related analyses are not cycle-specific. The nuclear fuel review for the MUR Uprate Program evaluated the nuclear design, fuel rod design, core thermal-hydraulic design, and fuel structural integrity.

Reload-specific evaluations that confirm the loading patterns and associated fuel types utilized in future reload designs will be performed. In addition, prior to implementing this uprate, a reload safety evaluation will be performed to ensure that the core design bounds the uprated condition.

IV.8.1 Nuclear Design

The neutronics impacts of two different scenarios were analyzed or evaluated. The first scenario addresses the mid-cycle power uprate of CNP Unit 1 Cycle 18 and the second scenario considers subsequent cycles which, in addition to the 1.66 percent power uprate, are also affected by changes to system pressure and to vessel average moderator temperature.

I&M currently plans to perform the Unit 1 MUR uprate in mid-cycle of the current operating cycle (Cycle 18). The HFP inlet temperature of the core will be decreased by about 0.4°F to maintain HFP vessel average moderator temperature of the core at its original value of 556°F and to minimize the fuel T/H impacts of the uprate. System pressure will remain at the current value of 2100 psia. The mid-cycle power uprate will result in small changes to the axial and radial power distribution in the core and to critical boron concentration.

In the second scenario, system pressure was increased from 2100 to 2250 psia and vessel average moderator temperature was increased from 556 to 574°F in the neutronics model of the core for Cycle 19 and beyond. The net impact of the 1.66 percent power uprate, including associated fuel T/H parameter changes and the feed fuel enrichment increase, is that small changes to the axial and radial power distribution in the core and to critical boron concentration will result.

For both scenarios described above, changes to the power distribution are small compared to typical cycle-to-cycle variability and when compared to typical peaking factor limit margins. Also, for both scenarios described above, changes to boron concentrations, reactivity coefficients, shutdown margin, and to other safety analysis inputs will be small. Each future Unit 1 cycle will be routinely analyzed to confirm that all applicable limits are met. Any future differences in key parameters, beyond what was considered in this report, will be routinely addressed via the standard reload design process.

IV.8.2 Fuel Rod Design

The fuel rod design criteria evaluated for a standard reload design have been evaluated at the MUR Uprate Program conditions for both the Cycle 18 core and a representative future cycle core for CNP Unit 1. The results of these evaluations demonstrated that the fuel would be expected to meet all fuel rod design criteria at the MUR Uprate Program conditions.

Fuel rod design analyses are performed on a cycle-specific basis considering the plant conditions of the specific cycle as well as the fuel duty of each of the fuel regions in the core during the cycle. These analyses are performed using NRC-approved models in References IV.8.2 to IV.8.4, and methods in References IV.8.5 to IV.8.7 to demonstrate that all fuel rod design criteria will be met. The results of the fuel rod design analyses are reported in the cycle-specific Reload Safety Evaluation (RSE) report as part of the normal reload design process.

IV.8.3 Core Thermal-Hydraulic Design

The CNP Unit 1 core T/H analysis and evaluations were performed at a 1.7 percent uprated core power level of 3305 MWt, which bounds the proposed 1.66 percent uprate. The DNBR design limits and safety analysis limits were kept unchanged from the values used in the non-uprate analysis. DNB margin was used to account for the penalty due to the power uprate. The DNBR portion of the Core Limits and the Axial Offset Limits were kept unchanged to minimize the impact on the OTΔT and OPΔT protection setpoints.

The DNBR analysis of CNP Unit 1 at the MUR Uprate Program conditions showed that the DNBR design basis continued to be met. The core T/H evaluations were the only portion of the uprate evaluation that evaluated a 1.7 percent uprate, rather than a 2 percent uprate for bounding considerations. However, these analyses are reanalyzed each reload.

IV.8.4 Fuel Structural Evaluation

The 15x15 Optimized Fuel Assembly (OFA) assembly designs were evaluated to determine the impact of the MUR Uprate Program on the fuel assembly structural integrity. The original core plate motions remain applicable for the MUR Uprate Program. Therefore, there is no effect on the fuel assembly seismic/LOCA structural evaluation. The MUR Uprate Program has an insignificant impact on the operating and transient loads, such that there is no adverse effect on the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected, and the seismic and LOCA evaluations for the 15x15 OFA fuel assembly designs remain applicable.

References (Section IV.8)

IV.8.1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985

IV.8.2 Weiner, R. A. et al, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-proprietary), dated August 1988

IV.8.3 Davidson, S. L., and D. L. Nuhfer, "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A, dated April 1995

IV.8.4 Foster, J. P., et al, "Westinghouse Improved Performance and Analysis Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 (Proprietary) and WCAP-15064-NP-A, Revision 1 (Non-proprietary), dated July 2000

IV.8.5 D. H. Risher, Ed., "Safety Analysis for the Revised Rod Internal Pressure Design Basis," WCAP-8963-P-A, dated August 1978

IV.8.6 Davidson, S. L., et al, "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, dated December 1985

IV.8.7 Kersting, P. J., et al, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A, dated March 1995

V. Electrical Equipment Design

This evaluation describes the impact on various electrical systems due to the proposed 1.66 percent power uprate. To ensure that the proposed power uprate was bounded by the evaluation, several areas were examined using a 2 percent power uprate value while others used a 1.7 percent power uprate value.

As a result of this MUR Uprate Program, the station electrical output will increase. The design ratings of the main generator, main step-up transformer, and iso-phase bus bound the increases expected from the uprate. The existing switchyard circuit breaker ratings bound the expected current flow associated with the 1.66 percent power uprate. The generator voltage controls and grid source impedance will not be impacted by the 1.66 percent power uprate. The proposed MUR Uprate Program will not impact the grid stability analysis. Auxiliary transformer design ratings bound any expected bus loading increases that result from increased system flow rates in the secondary plant. The station electrical distribution system (4160 Volts) is not directly impacted by the 1.66 percent power uprate. EDG loading is not impacted by the proposed uprate.

Table V-1 Impact of Power Uprate on Electrical Equipment

Component	Impact of Power Uprate
Turbine/Generator	None. Bounded by design rating.
Iso-phase Bus	None. Bounded by design rating.
Main Transformer TR1	None. Bounded by design rating.
Switchyard	None. Bounded by design rating.
Offsite Power Feeders	Insignificant. [2]
Grid Stability	None.[1]
EDGs	Within design rating.
Auxiliary Transformers	None. Bounded by design rating.
Station Service Transformers	None. Bounded by design rating.
Protective Relay Settings	No impact on protective relay settings.
Electrical Distribution System	Insignificant.

Table Notes: 1 Grid Stability is based on Short Circuit Analysis which is not affected by increased unit output.
2 Switchyard breakers protect tie lines, and breaker ratings are not challenged.

In each case, the current design of these components and systems continue to bound the 1.66 percent power uprate conditions. Major components and impacts of the uprate are discussed in further detail below.

Turbine-Generator

The proposed 1.66 percent power uprate will yield an increased turbine generator output.

A conservative power uprate of 2 percent of the current typical generator output of 1077 MW electric yields a maximum net increase in generator output to 1099 MW electric. This is equivalent to 1221 MVA, still below the generator nameplate rating of 1280 MVA. A 2 percent power uprate would result in the generator output current being increased by approximately 530 amperes (amps), to a new output total of 27,113 amps at 26,000 volts (V), which is within the nameplate rating for the generator. Further evaluations of turbine support systems, located in Sections VI.2 and VI.3 of this report, demonstrate that their design bounds the 1.66 percent power uprate.

Main Transformer

The main transformer rating of 1300 MVA provides a margin of 20 MVA above the maximum rating of the main generator. A 2 percent power uprate will raise the main generator output to 1221 MVA. Therefore, a 2 percent power uprate will not challenge the main transformer rating of 1300 MVA, and this MUR Uprate Program will not exceed the capability of the main transformer.

Offsite Power Feeders

Based on the small increase expected due to a 2 percent power uprate (about 40-amp increase on the 345 kV grid), the impact on the tie-line ratings will be insignificant. The tie-lines are protected by the switchyard circuit breakers. The ratings of these breakers are not challenged by the expected 1.66 percent power increase. Therefore, the tie-lines will not be impacted by the expected uprate.

Grid Stability

The CNP Unit 1 connection to the 345 kV grid is through the various tie-lines that connect at the 345 kV switchyard, with each line isolated by circuit breakers. The present output of the main generator at 100 percent rated thermal power is approximately 1077 MW electric, which is equivalent to 1197 MVA at a power factor of 0.9. This is below its maximum rating of 1280 MVA. This output supplies the switchyard, and thus the 345 kV grid, with an equivalent current flow of 2,003 amperes at 345,000 V. A 2 percent power uprate would result in the main generator output being increased to 1221 MVA. This higher generator output is equivalent to 2,043 amps at 345,000 V. Thus, the 345 kV grid could see an increase in current flow of about 40 amps.

The impact of this increase in load output on grid stability is insignificant. The impact on grid stability, if any, would be noted in the associated short circuit analysis. A short circuit analysis uses the assumptions for impedance values of the components tied to the system being analyzed. The values assumed for the main generator source impedance will not be changed by the MUR Uprate Program. Therefore, the proposed 1.66 percent power uprate will not impact the grid stability analysis.

Emergency Diesel Generators

Existing accident analyses bound the increase in reactor core decay heat. The heat removal systems, including the RHR, CCW, and ESW system pumps, have been evaluated and found to have insignificant effects on system flow due to the proposed 1.66 percent 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 1.66 percent power uprate.

SBO and Environmental Qualification (EQ)

The effect of the MUR Uprate Program on the SBO event and the plant's ability to cope with a complete loss of AC electric power has been reviewed. The ability of the CNP Unit 1 to respond to an SBO event will not be impacted by the Unit 1 MUR Uprate Program. This review is further documented in Section II.3.14 of this license amendment request. Similarly, there is no impact on the EQ of electrical equipment, or the EQ Program. This review is further documented in section VII.6.1 of this license amendment request.

VI. System Design

VI.1 NSSS Interface Systems

This section discusses the evaluations performed on the NSSS fluid systems using the revised design parameters presented in Table 3, "CNP Unit 1 MUR Uprate - NSSS Design Parameters." For this evaluation, calculations were evaluated to determine whether the NSSS would be impacted by the MUR Uprate Program.

The following parameters have been determined to be bounded by analysis performed for the 3588 MWt rerating in WCAP-12135 (Reference VI.1) and the associated license amendment request (Reference VI.2), which was approved by the NRC in CNP Unit 1 License Amendment No. 126 (Reference VI.3):

- RTD bypass delay times,
- pressurizer surge line pressure drop,
- pressurizer relief line pressure drop,
- pressurizer spray flow capability, and
- RCS loop pressure drops

In addition, a calculation was performed at 3588 MWt to verify that the boration volumes in the TS and UFSAR are adequate at that power level. The calculation demonstrates that boration capabilities for cold shutdown bound the MUR Uprate Program conditions. These boration volume requirements are verified on a cycle-specific basis.

The natural circulation cooldown capability is not affected because the T_{hot} and T_{cold}, and therefore the no-load ΔT between the RCS and the steam generator, values for the MUR Uprate Program are bounded by the current analysis. The loss of offsite power event, which credits the natural circulation process, was analyzed with 2 percent power measurement uncertainty, which bounds the 1.66 percent power uprate.

VI.1.1 CVCS System/Boron Capability

The T_{hot} and T_{cold} values for the 1.66 percent power uprating are bounded by the Rerating Program values; therefore, the operating temperatures and boron capability of the CVCS associated with MUR Uprate Program are acceptable.

VI.1.2 Auxiliary Heat Exchanger Performance

The impact of the MUR Uprate Program on the performance of the regenerative, letdown, excess letdown, and seal water heat exchangers was evaluated. The NSSS performance parameters for the MUR Uprate Program are bounded by the Rerating Program performance parameters for these components; therefore, auxiliary heat exchanger performance will not be impacted by the MUR Uprate Program.

VI.1.3 RHR System

Various CNP TS Action Statements require that, if the applicable LCO is exceeded, the plant must be placed in cold shutdown ($T_{RCS} < 200°F$, Mode 5) within 36 hours. To demonstrate that the plant has the capability to meet these TS requirements, an analysis of the time required to cool the RCS from Mode 1 to Mode 5 was performed assuming a single train of the RHR system and associated cooling support system equipment. The single train cooldown analysis requirement is the standard Westinghouse assumption for RHR cooldown analyses.

CNP's current licensing basis requires that, under normal operating conditions, the RHR system will be capable of reducing RCS temperature to 140°F within 20 hours following a reactor shutdown (UFSAR Section 9.3.1). The capability to cool the RCS from Mode 1 to Mode 5 within 20 hours is demonstrated via an analysis that assumes two-train cooldown.

Single-Train Cooldown
The results of the single-train cooldown analysis demonstrate that the plant can be cooled from Mode 1 to Mode 5 (200°F) within 36 hours, which is within TS requirements.

Two-Train Cooldown
The current two-train cooldown analysis assumes a core power of 3411 MWt, which bounds the proposed 1.66 percent power uprate conditions.

VI.1.4 ECCS and Containment Spray System (CTS)

The MUR Uprate Program design operating parameters for RCS temperature, pressure, and flow are bounded by the previous uprating program values. The current long-term core cooling analysis for CNP employs a nominal core power high enough to cover both Units 1 and 2 (3481 MWt). Also, consistent with the requirements outlined in 10 CFR 50, Appendix K, the decay heat model assumed in the LOCA long-term core cooling analysis is 1.2 times the values for infinite operating time in the 1971 ANS Standard. Therefore, there is no impact due to decay heat considerations on the ECCS system, and there is no affect on the performance of the ECCS or the CTS.

VI.2 Power/Steam Systems

As part of the CNP Unit 1 MUR Uprate Program, the following BOP fluid systems were reviewed to assess compliance with the Westinghouse NSSS/BOP interface guidelines:

- Main Steam System
- Steam Dump System
- Condensate and Feedwater System
- Auxiliary Feedwater System
- Steam Generator Blowdown System

The review was performed based on the range of NSSS design parameters presented in Table 3, "CNP Unit 1 MUR Uprate - NSSS Design Parameters." The various interface systems were reviewed to provide interface information that could be used in the BOP analyses.

<u>Input Parameters and Assumptions</u>

The parameters in Table 3 were compared with the non-uprated parameters previously evaluated for the WCAP-15608, Revision 1, "D. C. Cook Unit 1 Replacement Steam Generator Safety Analysis Program Engineering Report" (Reference VI.4), which was incorporated into the CNP Unit 1 licensing basis via a 10 CFR 50.59 evaluation. The comparison indicated differences that could impact the performance of the BOP systems identified above. For example, the 2 percent increase in core power (from 3250 to 3315 MWt) coupled with a zero SGTP level would result in about a 2.4 percent increase in steam/feedwater mass flow rates. Additionally, the average SGTP level of 30 percent in combination with the upper limit on T_{avg} (575.4°F) would result in a reduction in full-load steam pressure from 780 psia to 765 psia. Currently, a minimal number of tubes are plugged in the Unit 1 steam generators (less than 0.03 percent). Evaluation of the interface systems, delineated below, indicates that, except for the steam dump valves, the design of these systems bounds operation at the uprated core power level, 3304 MWt.

Description of Analyses and Results

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines were performed to address the NSSS design parameters for the 1.66 percent power uprate analyses. NSSS design parameters were evaluated for a range of values, including parameters such as T_{avg} (553.7°F to 575.4°F) and SGTP (0 percent to 30 percent). Based on the range of NSSS design parameters that were evaluated, the resultant range of BOP parameters, such as steam generator outlet pressure (618 psia to 856 psia), were determined. The NSSS/BOP interface evaluations were performed to address these NSSS and BOP design parameters. The following is a brief summary of the NSSS/BOP interface evaluation conclusions for the MUR Uprate Program.

VI.2.1 Main Steam (MS) System and Steam Dump System

Major piping, valves, tanks, and turbines of the 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 1.66 percent uprated power. However, the design of the steam dump valves does not meet the current licensing basis requirements to provide 40 percent steam dump capacity. These valves are currently gagged to limit valve travel to 2.75-inches. Westinghouse has evaluated the capability of the steam dump valves to satisfy their design basis function at the 1.66 percent uprated power level and preliminarily concludes that the steam dump valves at their current travel stop positions have sufficient flow capacity in the current configuration for the 1.66 percent power uprate. However, if final steam dump valve flow capacity analysis is not successful, 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 1.66 percent power uprate.

The maximum operating design pressure and temperature of the MS system are not changed for the proposed increase in plant power conditions. The system full load operating pressure is expected to decrease; therefore, the MUR Uprate Program will have no impact on the existing structural evaluations of the MS system piping and supports.

The MS system has been reviewed for the proposed power uprate operating conditions and found to be acceptable. Based on the results of analyses that bound the impacts of the proposed 1.66 percent power uprate, changes to UFSAR Chapter 14 Sections 14.2.4, "Steam Generator Tube Rupture", and 14.2.5, "Rupture of a Steam Pipe", are not required for the MUR Uprate Program.

CNP Unit 1 TS LCO 3.7.1.1, "Turbine Cycle – Safety Valves," specifies the OPERABILITY requirements for the main steam line code safety valves. TS Table 3.7.1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation," specifies the maximum allowable Power Range Neutron Flux High setpoints with inoperable main steam safety valves (MSSVs). To support the MUR Uprate Program, a calculation was performed to identify changes to the maximum allowable power limits with inoperable MSSVs. Using the equation provided in the Bases for TS 3/4.7.1.1, with a revised NSSS power rating, Q, of 3325 MWt (3305 MWt core power plus 20 MWt RCP pump power), a revised set of setoints was calculated for cases assuming one, two, or three inoperable MSSVs on one operating steam generator loop. The results of this calculation are reflected in the proposed changes to TS Table 3.7-1 (see Section VIII of this attachment).

VI.2.2 Condensate and Feedwater Systems

A comparison between operating requirements for the 3304 MWt conditions generated by heat balances compared to historical operating data demonstrates that the following pumps have more than sufficient design and operational margin to accommodate the MUR uprated conditions:

- Hotwell pumps
- Condensate booster pumps
- Main feedwater pumps
- Heater drain pumps

In addition, the margin in the design and operation of the feedwater regulating and isolation valves will continue to bound the uprated conditions. Finally, the uprated flow rates for the condensate and feedwater heaters have been demonstrated to be bounded by the design flowrates for these components. Also, the lower system pressures ensure that the piping and support systems are not affected by the 1.66 percent power uprate.

The condensate and feedwater systems have been reviewed for the proposed 1.66 percent power uprate operating conditions and were found to be acceptable. Components within these systems are bounded by previous analyses.

VI.2.3 AFWS and Condensate Storage Tank (CST)

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the steam generator as a heat sink. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an Engineered Safety Feature (ESF). In the latter function, the AFWS is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary system pipe break. The minimum flow requirements of the AFWS are dictated by accident analyses, and since the uprating impacts safety analyses performed at the current 100 percent power rating, evaluations were performed to confirm that the AFWS performance is acceptable at the 1.66 percent power uprate conditions. These evaluations show acceptable results.

The AFWS pumps are normally aligned to take suction from the CST. To fulfill the ESF design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST inventory requirements is the loss-of-offsite power (LOOP) transient. The CNP Unit 1 licensing basis requires that, in the event of a LOOP, sufficient CST usable inventory must be available to bring the unit from full power to hot standby conditions, and maintain the plant at hot standby for 9 hours. In addition, Westinghouse recommends that, in the event of a LOOP, sufficient CST useable inventory be available to bring the unit from full power to hot standby conditions, hold the plant at hot standby conditions for two hours, and then cooldown the RCS to the RHR system cut-in temperature (350°F) in 4 hours. In light of these design bases requirements, the CNP Unit 1 CST TS (3/4.7.1.3) and the CNP Administrative Technical Requirements ensure a usable volume of 175,000 gallons.

The minimum required useable inventory of 175,000 gallons is based on reactor trip from 102 percent of maximum calculated power of about 3481 MWt (or 1.05 x 1.02 x 3250). Since the MUR Uprate Program is based on a reduced calorimetric error, no change in the plant TS is required for operation at the uprated power level.

The maximum operating design pressure and temperature of the AFWS system are not changing based on the 1.66 percent power uprate conditions. Therefore, the 1.66 percent power uprate will have no impact on the existing structural evaluations of the AFWS piping and supports.

There are no component level impacts due to the 1.66 percent power uprate. The normal unit startup, hot standby, and cooldown functions of the AFWS remain unchanged since the power uprate has no significant impact on the feedwater flow in these modes.

VI.2.4 Feedwater Heaters and Drains

I&M evaluated the nominal and maximum nozzle velocities for the feedwater heater and drain system. The results of these evaluations indicated that fluid velocities in the feedwater heaters and drains do not exceed maximum limits in industry standards. Therefore, I&M concludes that corrosion of these nozzles will not become an issue at the uprated conditions. In addition, current heat exchanger and system engineering system health monitoring programs will identify any issues with these heat exchangers.

Feedwater heater design temperatures and pressures were compared to the temperatures and pressures that were determined via a heat balance for this system. None of the temperatures and pressures indicated on the heat balance, for either the current or 1.66 percent power uprate conditions, exceed the design temperatures and pressures for these components.

The effect on the feedwater heater drain lines was evaluated. Each drain line contains an installed regulating valve to control the level of the associated feedwater heater, except Feedwater Heater No. 1. Feedwater Heater No. 1 drain lines do not contain regulating valves; level is controlled by a loop seal. The flow rates through each of these valves will increase by approximately 1 to 4 percent due to the MUR Uprate Program. Since none of the drain line valves are typically more than 66 percent open, no valve modifications are needed. The design flow velocity for steel pipe is not exceeded for any of the drain lines at the 1.66 percent power uprated condition.

Since the maximum operating pressures and temperatures of the feedwater heaters are not changing, the existing code piping analyses are not impacted by the 1.66 percent power uprate and will have no effect on qualification or adequacy of piping components.

The feedwater heater and drain systems have been reviewed for the 1.66 percent power uprate operating conditions and found to be acceptable. Components within the system are either unaffected or are bounded by previous analyses.

VI.2.5 Steam Generator Blowdown System

The inlet pressure to the Steam Generator Blowdown (BD) system varies with steam generator operating pressure. As steam generator full-load operating pressure decreases, the inlet pressure to the BD system control valves decreases and the valves must open to maintain the required blowdown flow rate into the system flash tank. The current NSSS design parameters (Reference VI.4) permit a maximum decrease in steam pressure from no-load to full-load of 399 psi (i.e., from 1020 psia to 621 psia). Based on the revised range of NSSS design parameters approved for the MUR Uprate Program, the no-load steam pressure (1020 psia) remains the same and the minimum full-load steam pressure (618 psia) is 3 psi lower than the original full-load pressure at 30 percent SGTP. This decrease in BD system inlet pressure at 30 percent SGTP will not impact the required maximum lift of the blowdown flow control valves. Therefore, the range of design parameters approved for 1.66 percent power uprate will not impact blowdown flow capability.

VI.3 Cooling and Support Systems

VI.3.1 CCW System

The CCW system has been reviewed for the 1.66 percent power uprate operating conditions and found to be acceptable. Most components cooled by the CCW system are not impacted by the MUR Uprate Program, while existing design analyses bound the proposed power uprate for the remaining cases. Additionally, margin exists in the current system design for the current cooling water flow requirements.

VI.3.2 ESW System

The ESW system has been reviewed for the 1.66 percent power uprate operating conditions and found to be acceptable. Most components cooled by the ESW system are not impacted by the MUR Uprate Program, while existing design analyses bound the proposed power uprate for the remaining components. Additionally, margin exists in the current system design for the current cooling water flow requirements.

VI.3.3 NESW System

The existing NESW system capacity has been reviewed and determined to be capable of accommodating the 1.66 percent power uprate. There is margin in the system to accommodate increased cooling flow from the steam generator blowdown system that would result from operation at increased power levels.

VI.3.4 TACW System

Evaluation of the TACW system shows that this system is currently sufficient to support the 1.66 percent power uprate for all generator-specific component cooling The system, as designed, is capable of accommodating additional iso-phase bus duct enclosure cooling requirements; however, increased TACW cooling water flow to the iso-phase bus duct cooling system may be required. There is no effect on the steam packing exhauster as a result of the MUR Uprate Program.

Therefore, the TACW system is capable of accommodating the MUR Uprate Program.

VI.3.5 EDG Aftercooler, Lube Oil, and Jacket Cooling Water System

The EDG aftercooler, lube oil and jacket cooling water system, as designed, is capable of accommodating the proposed power uprate. There is currently margin between the existing required cooling water flowrates and the original design flowrates. Furthermore, the EDGs will not be subjected to any additional loading requirements as a result of the MUR power uprate. Therefore, the EDG aftercooler, lube oil, and jacket cooling water system is capable of accommodating the proposed 1.66 percent power uprate, and no additional cooling is required.

VI.3.6 Circulating Water (CW) System

The only CW system impact would be that the main condensers will operate at a slightly higher back pressure (approximately 0.1 psi) and condenser hotwell temperature will increase (approximately 1°F), thereby resulting in an increase in the CW heat rejection rate. The current system design and operation bound these changes. A review of the thermal discharge limits (Section VII.5 of this Attachment) concluded that there was no impact from the MUR Uprate

Program to Lake Michigan since the current thermal discharge permit bounds the 1.66 percent power uprated conditions.

Therefore, there is no impact to the CW system as a result of the MUR Uprate Program.

VI.3.7 SFPC System

The only potential impact to the SFPC system resulting from the MUR Uprate Program is the amount of additional decay heat resulting from operation at higher power. Existing analyses assumed 102 percent of the current power level, which bounds the MUR Uprate Program conditions (References VI.10 and VI.11). Additionally, to ensure compliance with the CNP Unit 1 TS requirements for spent fuel pool loading, reviews are performed or confirmed to be bounding prior to each off-load. Therefore, there is no impact to the SFPC system as a result of the MUR Uprate Program.

VI.4 Heating, Ventilating and Air-Conditioning (HVAC) Systems

I&M has evaluated the radiological consequences of the Chapter 14 design basis events and concluded that the increased power output is fully bounded by the existing analyses. Therefore, there will be no effect on the ability of the Engineered Safety Features Ventilation System (ESFVS) to perform its functions.

The proposed generating capability increase, in conjunction with typical summer temperatures, will increase the temperatures inside the iso-phase bus ducts. To support reliable power transmission at the increased output, the cooling capacity of the generator bus duct cooling system will be monitored during plant operation. High temperatures will indicate the need to increase the flow of TACW to the fan coil units. This action will increase the temperature difference between the air entering the coil and the air leaving the coil. The fans have been verified to have more than sufficient capacity to meet system design basis requirements. The evaluation of TACW performance indicates that there is sufficient margin in system capacity to provide additional flow. Air flow, in conjunction with increased TACW flow, will assure adequate supply and return air temperature difference necessary to maintain the temperature inside all three ducts within established operating limits following implementation of the MUR Uprate Program.

The iso-phase bus duct cooling system, which is non-safety related but required for transmission of power, will need to be monitored for the additional heat load due to increased amperage passing through the bus as a result of the MUR Uprate Program. High temperatures in the bus ducts will need to be compensated for by adjustment of the flow of TACW to the fan coil units.

Any changes to TACW flow to the generator bus duct cooling system will also require changes to procedures associated with these systems. There are no UFSAR or TS changes for the generator bus duct cooling system associated with the Unit 1 MUR Uprate Program.

The auxiliary building ventilation system, ESFVS, containment ventilation system, MDAFW/TDAFW room coolers and the iso-phase bus duct coolers were reviewed to evaluate the impact of the 1.66 percent power uprate on these systems. Only the ESFVS, containment equalization (CEQ) fans, and the MDAFW/TDAFW room coolers serve a licensing basis function. The conclusions presented in the UFSAR related to these systems will not change as a result of the 1.66 percent power uprate. No changes to the TS are required.

VI.5 NSSS Control Systems

Evaluations of the instrumentation and control capabilities of the individual systems discussed in sections VI.1 through VI.4 were determined to be unaffected by the 1.66 percent power uprate. This section specifically addresses the effects of the 1.66 percent power uprate on the NSSS control systems.

ANS Condition I transients (as described in Reference VI.12) are evaluated to confirm that the plant can respond to these transients without generating a spurious reactor trip or ESF actuation.

The design basis for the analyses used to determine NSSS control setpoints is the 3600 MWt rerating effort documented in References VI.1 and VI.9. Analyses were also performed for the SGTP Program (Reference VI.7), which largely continue to use the Reference VI.9 analyses. These are, therefore, the design basis operability analyses, which will be used to evaluate the continued acceptability of the plant control system operation for the MUR Uprate Program.

The limiting transients analyzed in References VI.7 and VI.9 are the following:

- Ramp load increase of a maximum of 1 percent per minute between 20 percent and 100 percent power.
- Ramp load decrease of a maximum of 5 percent per minute between 100 percent and 20 percent power, without steam dump actuation.
- 10 percent load decrease at a maximum rate of 200 percent per minute.
- Approximately 40 percent load rejection (from 100 percent to 60 percent power) at a maximum rate of 200 percent per minute with steam dump actuation.

The following transients were analyzed in References VI.7 and VI.9 to determine the acceptable operation of the Plant/Turbine Trip Controller mode of steam dump operation:

- Turbine and reactor trip transients initiated from 100 percent power with steam dump actuation assuming 27 percent steam dump capacity.

The analyses performed in References VI.7 and VI.9 were reviewed for continued acceptability for the MUR Uprate Program conditions discussed in the Introduction (Table 3) and were concluded to bound the MUR Uprate Program.

As discussed in Section VI.2.1, MS System and Steam Dump System, additional analyses will be performed to determine the acceptability of the actual steam dump capacity for the 1.66 percent power uprating.

Condition I Transient Evaluations

The analyses performed for the Rerating Program (References VI.1 and VI.9) and the SGTP program (References VI.7 and VI.13) were based on a nominal power level of 3600 MWt. Table VI-1 contains a comparison of the plant operating condition used in References VI.1 and VI.13 versus the plant operating conditions identified in the Introduction Section of this attachment, including Table 3. The operating conditions used in the References VI.9 and VI.7 analyses bound those for the MUR Uprate Program. Therefore the References VI.1 and VI.13 analyses are also valid and bounding for the MUR Uprate Program. The analyses demonstrated that there is acceptable margin to the Reactor Trip System (RTS) setpoints and ESF actuation setpoints for all of the above limiting operability transients except for potentially the load rejection from 100 percent to 60 percent power. This transient could potentially result in a reactor trip from the limiting lower bound full-power T_{avg} values and beginning of core life conditions. The Reference VI.1 and VI.13 analyses noted that this could be the case with a 40 percent steam dump capability; the results would be slightly aggravated by the reduced steam dump capability that is discussed in Section VI.2.1, MS System and Steam Dump System. Operation at higher values of T_{avg} and/or higher core burnups would tend to result in the limiting 100 percent to 60 percent load rejection at 200 percent per minute being able to be accommodated (i.e., without a reactor trip occurring). Also, the Reference VI.1 and VI.13 analyses concluded that if the load rejection rate is reduced to approximately 20 percent per minute, any value of full power T_{avg} within the analysis range and any cycle burnup could be accommodated. Therefore, the RTS and ESFAS setpoints remain bounding for the MUR Uprate Program.

Table VI-1 -- Unit 1 Two Percent Uprate Conditions vs. Values used in Design Basis Transients

	Unit 1 MUR Uprate Program (from Table 3)		Rerating Program (Ref. VI.1 and VI.9)	
Parameter	**High T_{avg}**	**Low T_{avg}**	**High T_{avg}**	**Low T_{avg}**
Reactor Thermal Power, MWt	3315 **	3315 **	3588	3588
NSSS Power, MWt	3327	3327	3600	3600
Thermal Design Flow, Loop gpm	83,200	83,200	88,500	88,500
Reactor Coolant pressure, psia	2250/2100	2250/2100	2250/2100	2250/2100
T_{hot}, °F	609.1	588.2	615.2	582.3
T_{avg}, °F	575.4	553.7	581.3	547.0
SG Outlet Temperature, °F	541.5	518.9	547.1	511.7
P_{steam}, psia *	765	618	820	587
Feedwater Temperature, °F	437.4	437.4	449	449

* Unit 1 Uprating values are for limiting 30 percent SGTP; Rerating Program values are for 10 percent SGTP condition.

** The reactor thermal power assumed in the NSSS design basis transient analyses (3315 MWt) conservatively bounds the value requested by this license amendment request (3304 MWt).

NSSS Pressure Control Component Sizing

I&M evaluated the sizing of NSSS pressure control components to determine if the installed capacity of the various pressure control components is still acceptable for the 1.66 percent power uprate conditions. The results obtained from the 3600 MWt rerating effort were used as the primary basis for the evaluation.

The following pressure control components were each evaluated separately:

- Pressurizer heaters
- Pressurizer spray
- Pressurizer PORVs

The heatup time from cold shutdown to hot standby is not impacted by the MUR Uprate Program; the heatup maneuver would be essentially the same as that presently experienced. Therefore, the installed pressurizer heater capacity is acceptable for the MUR Uprate Program.

The Reference VI.9 analyses demonstrated that transients affecting the pressurizer sprays could be accommodated for a design NSSS power level of 3600 MWt without challenging the pressurizer PORVs. These analyses were performed based on a maximum pressurizer spray flow of 736 gpm, rather than the design analysis value of 800 gpm used for the MUR Uprate Program.

The 3600 MWt power level analysis has sufficient conservatism to bracket the 1.66 percent power uprate conditions with a steam dump capacity as low as 27.9 percent of nominal full-power steam flow (steam dump limitations discussed in Section VI.2.1). Based on the higher power level and lower spray flow capacity, the Reference VI.9 analyses are bounding for the MUR Uprate Program.

Low Temperature Overpressure Protection (LTOP) System

The MUR Uprate Program does not change any plant conditions that would impact the LTOP system. For low temperature/overpressure events, the plant is in a shutdown condition; therefore, the uprating does not impact the plant response for these events. Therefore, there is no direct impact on the LTOP system due to the MUR Uprate Program.

References (Section VI)

VI.1. WCAP-12135, "Donald C. Cook Units 1 and 2 Rerating Engineering Report," dated September 1989

VI.2. Letter from M. P. Alexich, I&M, to T. E. Murley, NRC, "Reduced Temperature and Pressure Program Analyses and Technical Specification Changes," AEP:NRC:1067, dated October 14, 1988

VI.3. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC No. 71062)," dated June 9, 1989

VI.4. WCAP-15608, Rev. 1, "D. C. Cook Unit 1 Replacement Steam Generator Safety Analysis Program Engineering Report," dated March 2001

VI.5. WCAP-14489 Revision 1, "Donald C. Cook Nuclear Plant Unit 2 3600 MWt Uprating Program Licensing Report", dated May 1996

VI.6. Letter from E. E. Fitzpatrick, I&M, to Nuclear Regulatory Commission, "Proposed License and Technical Specification Changes Supported by Analyses to Increase Unit 2 Rated Thermal Power and Certain Proposed Changes for Unit 1 Supported by Related Analyses," AEP:NRC:1223, dated July 11, 1996

VI.7. WCAP-14285, "Donald C. Cook Nuclear Power Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," dated May 1995

VI.8. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC Nos. M92587 AND M92588)," dated March 13, 1997

VI.9. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988 and WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report," dated September 1989

VI.10. Letter from W. M. Dean, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Amendment Nos. 169 and 152 to Facility Operating License Nos. DPR-58 and DPR-74 (TAC Nos. M80615 and M80616)," dated January 14, 1993

VI.11. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant Amendment No. 260 to Facility Operating License No. DPR-58 and Amendment No. 243 to Facility Operating License No. DPR-74: Indiana Michigan Power Company Donald C. Cook Nuclear Plant, Units 1 and 2; Docket Nos. 50-315 and 50-316 (TAC Nos. MB1975 and MB1976)," dated November 30, 2001

VI.12. ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 1973

VI.13 WCAP-14286, "Donald C. Cook Nuclear Power Plant Unit 1 Steam Generator Tube Plugging Program Engineering Report," dated December 1995

VII. Other

VII.1 Control Room and Simulator

There are no annunciators being added to the control room as a result of the MUR Uprate Program. Notification of the operators of the LEFM CheckPlus system condition will be through computer alarms and annotation of the computer display. Response to this computer alarm will be proceduralized. This will be finalized in the design change to implement the MUR Uprate Program in coordination with Operations to ensure that implementation meets operations and design requirements. Control room instrumentation and displays will be re-scaled as a result of implementation of the 1.66 percent power uprate. This will be addressed in the design change package that implements the installation of the LEFM CheckPlus system.

The CNP simulator, which reflects the design of the Unit 2 control room, will not be modified during the Unit 1 modification.

VII.2 Operator Actions

No changes are required to the CNP EOP program as a result of the MUR Uprate Program. Specific procedures within the EOP program may require review and revision based upon the MUR Uprate Program plant parameters for thermal power, temperature, and pressure values. These changes will be identified and implemented under the design change process to implement the MUR Uprate Program. Specifically, values in the EOPs, the EOP Footnotes document, and the Abnormal Operating Procedures (AOPs) may need to be revised based upon the 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 MUR Uprate Program.

The MUR Uprate Program will have no impact on the time available for operator actions as assumed in the accident analysis. Specific impacts on operating procedures are further discussed in section VII.4 of this license amendment request.

VII.3 Power Uprate Modifications

As demonstrated in Sections II through VI, the current and re-analyzed plant analyses, design, and operation ensure that the applicable acceptance criteria are met for the MUR Uprate Program. Therefore, no changes to the RCS or NSSS systems are required to support the MUR Uprate Program, other than a potential change to the steam dump valves and the installation of the LEFM itself, as discussed below.

With the exception of the steam dump valves, there is no impact to the MS system, and no other modifications are required to implement the MUR Uprate Program. Because of small changes in main steam pressure, the uprate results in a slightly reduced steam dump capability. Westinghouse has evaluated the capability of the steam dump valves to satisfy their design basis function at the 1.66 percent uprated power level. The preliminary conclusion from this evaluation is that the steam dump valves have sufficient flow capacity in the current configuration for the 1.66 percent power uprate. A final analysis of the steam dump valve flow capacity is being performed. If the final analysis determines that these valves do not have sufficient capacity, then the steam dump valves' travel stop position will be changed to ensure that the reduced steam dump capability is adequate. The adequacy of the steam dump valve flow capacity will be confirmed prior to increasing plant power above 3250 MWt, and if necessary, the steam dump travel stop position will be changed to ensure that the reduced steam dump capability is adequate.

The changes in flowrates, pressures and other operating parameters can be accommodated by all existing equipment in the condensate or feedwater systems. Therefore, no plant changes/modifications are required to the condensate or feedwater systems to implement the MUR Uprate Program other than the installation of the LEFM flow instrumentation itself.

As the impacts of the MUR Uprate Program are bounded by the current design and operation of the AFWS, no modifications are required to this system for implementation of the MUR Uprate Program.

None of the existing feedwater heaters, nozzles, and drain lines, including regulating valves, will have to be replaced or modified to accommodate the uprated flows, temperatures, and pressures. No plant changes/modifications are required to the feedwater heaters and drains for implementation of the MUR Uprate Program.

The 1.66 percent power uprate will result in minimal changes in the CCW system flow requirements. These changes are bounded by current system design; therefore, no plant changes/modifications are required to the CCW system to implement the MUR Uprate Program.

Since there is no impact on the ESW system, no plant changes/modifications are required to implement the MUR Uprate Program.

There are no impacts to the NESW system that are not bounded by current system design; therefore, no NESW System changes will be required as a result of the MUR Uprate Program.

The only plant change for the TACW system is the potential additional flow requirement to the iso-phase bus duct cooling system, which may require increased flow from the TACW system. No TACW system modifications are required to support the MUR Uprate Program.

Because the EDGs will not be subjected to additional loading as a result of the MUR Uprate Program, no changes are required to the EDG cooling water systems. Therefore, no EDG cooling water systems modifications are required to support the MUR Uprate Program.

The only CW system impact would be that the main condensers will operate at a slightly higher back pressure (approximately 0.1 psi backpressure) and condenser hotwell temperature will increase (approximately 1°F), thereby resulting in an increase in the CW heat rejection rate. The existing system design, including instrumentation, bounds the uprated operating conditions; therefore, no CW system modifications are required to support the MUR Uprate Program.

The MUR Uprate Program will not impact the SFPC system; therefore, no SFPC system modifications are required to support the MUR Update Program.

No HVAC system modifications are required to implement the MUR Uprate Program. The iso-phase bus duct cooling system will be monitored to ensure the increased amperage through the iso-phase bus duct does not increase the system temperature above allowable limits. If so, increased TACW system flow may be required to the iso-phase bus duct coolers. Changes in flowrates can be accommodated by existing equipment. However, no modifications are required to the HVAC systems, including the ESFVS, to implement the MUR Uprate Program.

The review of electrical systems in support of the proposed uprate indicates that no changes are required to support the MUR Uprate Program.

VII.4 Plant Operating Procedure Changes

Procedural impacts for the RCS and NSSS systems will be identified in the process for the implementation of the design change package that installs the LEFM CheckPlus system. Impacts are anticipated to normal operating, alarm response, AOP and EOP procedures. In particular surveillance procedures for reactor thermal power will be affected, as well as operator responses to an out-of-service condition on the LEFM CheckPlus system, as described in Section I. These changes will be implemented prior to raising plant core power above 3250 MWt.

There are no main steam operating procedural changes required to implement the 1.66 percent uprated power level. The changes in flowrates, pressures, and other parameters due to the 1.66 percent power uprate will not necessitate equipment or operational changes outside of the existing MS system equipment design and operation. For the potential steam dump valve limit stop change, there may be a change in the installed position but this will not impact the operation of the steam dump system and so there are no anticipated changes to the plant operating procedures. For the TS 3.7.1.1 LCO change, the value of the nuclear instruments high flux setpoint will change in the CNP Unit 1 TS, but this will not change the LCO entry or exit criteria and, as such, will not change the plant operating procedures.

If increased TACW flow to the generator bus duct cooling system is required, the changes in flowrates will necessitate revisions to procedures that direct the operation of the TACW and iso-phase bus duct systems.

No other procedural impacts were identified in the review of NSSS, BOP, and support systems and their associated analyses.

VII.5 Environmental Review

I&M has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with

10 CFR 51.21. I&M has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Enclosure 2, Section 5.1, No Significant Hazards Consideration, this proposed amendment does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The MUR Uprate Program thermal power increase will not alter or increase the inventory of radionuclides in the RCS. This change will not alter the fuel cladding in a way that affects its mechanical and structural integrity or affects its leakage characteristics. This power uprate will not alter or increase the primary pressure or temperature, so there is no additional challenge to the RCS or other fission product barriers. Additionally, increasing core thermal power by 1.66 percent will not affect or increase water production or inventory use in any way that will affect effluent volume or production. Finally, the 1.66 percent uprated plant heat discharge combined with the existing Unit 2 discharge will remain below the site National Pollutant Discharge Elimination System (NPDES) limit of 17,300 million Btu/hr (Reference VII.5). The 1.66 percent power uprate is bounded by the previously-evaluated NPDES thermal effluent limits. Therefore, this change will not result in a significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The MUR Uprate Program thermal power increase will not alter or increase the inventory of radionuclides in the RCS. The radionuclide source terms applicable to personnel dose determination were calculated assuming a core thermal power of 3588 MWt, which bounds the uprated core power of 3304 MWt. This change will not alter the fuel cladding in a way that affects its mechanical and structural integrity or affects its leakage characteristics; therefore, there is no additional challenge to the RCS or other fission product barriers. Finally, no new effluents or effluent release paths are created by the MUR Uprate Program. Therefore, this change will not result in an increase in individual or cumulative occupational radiation exposures.

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

VII.6 Programs

VII.6.1 EQ Program

The CNP EQ Program has been reviewed in support of the Unit 1 MUR Uprate Program. This review has determined that no EQ Program changes are required to be implemented as a result of the MUR Uprate Program. The development of the EQ parameters bound the 102 percent core thermal power; therefore, the programs, activities, elements and philosophy that are currently in-place are not affected by the 1.66 percent power uprate. In accordance with CNP's design change process, any specific component modifications or changes that may be required to support the MUR Uprate Program will be evaluated against the EQ Program requirements.

VII.6.2 Motor-Operated Valve (MOV) Program

Design basis review calculations for feedwater, main steam, and plant cooling systems (i.e., CCW, ESW, NESW) were reviewed to determine potential impacts on the MOV Program. The limiting (bounding) differential pressures were based on system capacities and setpoints (e.g., steam generator safety valve setpoints, pump shutoff head), which will not change due to the proposed 1.66 percent power uprate to be implemented by the MUR Uprate Program.

Therefore, the 1.66 percent power uprate does not challenge the capability of valves in the MOV Program to satisfy their design functions. No changes are required to the MOV Program as a result of the MUR Uprate Program.

VII.6.3 Air and Hydraulic Operated Valve (AHOV) Program

The Unit 1 AHOV Program is still under development. Subsequent design analyses will be developed for normal plant operating conditions at a value as high as 102 percent of the plant's current RTP. The 1.66 percent power uprate to be implemented by the MUR Uprate Program will not affect the design basis of valves in the AHOV Program, because the analysis of these valves will remain bounded by the design basis requirements of their respective system-related functions. No changes are required to the AHOV Program as a result of the MUR Uprate Program. No AHOV valves were identified as impacted by the MUR Uprate Program, which will be verified during the design change process for installation of the LEFM CheckPlus system.

VII.6.4 Flow-Accelerated Corrosion (FAC) Program

The CNP FAC Program was reviewed in support of the MUR Uprate Program. This review determined that no FAC Program changes are required as a result of the MUR Uprate Program. The activities, elements and philosophy that are currently in place are not affected by the MUR Uprate Program. Flowrates and temperatures for piping components within the scope of the FAC Program remain within the system design specifications. In accordance with CNP's design change process, the design change package for installing the LEFM CheckPlus system will be evaluated against the FAC Program requirements as required in the CNP plant modification process. No specific TS or operating procedure changes were identified by the FAC Program review for the MUR Uprate Program. No changes are required to the FAC Program as a result of the MUR Uprate Program.

VII.6.5 High-Energy Line Break (HELB) Program

The CNP HELB Program was reviewed in support of the MUR Uprate Program. This review determined that no HELB program changes are required to be implemented as a result of the MUR Uprate Program. The activities, elements, and philosophy that currently constitute the HELB Program are not affected by the MUR Uprate Program. In accordance with CNP'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 CNP plant modification process. No new piping is added, no postulated break locations changed, and no changes are made to the assumed blowdown from any currently-postulated breaks; therefore, there is no impact on the current CNP Unit 1 HELB analysis.

The proposed 1.66 percent power uprate is bounded by the existing HELB analysis-of-record. These analyses are consistent with the requirements of Generic Letter 87-11 (Reference VII.1) and the CNP current licensing basis, as indicated in Unit 1 License Amendment Nos. 244 and 249 (References VII.2 and VII.3). No specific TS or operating procedure changes were identified by the HELB Program review for the MUR Uprate Program. No changes are required to the HELB Program as a result of the MUR Uprate Program.

VII.6.6 Fire Protection/Appendix R Programs

The activities and elements currently in place to implement the Fire Protection Program are not affected by the MUR Uprate Program or continued plant operation at the uprated thermal power level.

The post-fire safe shutdown aspect of the Fire Protection Program is in place to meet the requirements of 10 CFR 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 (i.e., AFW, RHR, MS). 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 50, Appendix R, remains unchanged.

No changes are required to the Fire Protection or Appendix R/Safe Shutdown Programs as a result of the MUR Uprate Program. In accordance with CNP's design change process, the design change package for installing the LEFM CheckPlus system will be evaluated against the Fire Protection/Appendix R Program requirements as required in the CNP plant modification process.

VII.6.7 Inservice Inspection (ISI) Program

The CNP ISI Program was reviewed in support of the MUR Uprate Program. No impacts were identified for the ISI Program system or component scope, boundaries, exemption or selection criteria, examinations, or acceptance standards. No changes are required to the ISI Program as a result of the MUR Uprate Program.

VII.6.8 Inservice Testing (IST) Program

The CNP IST Program has been reviewed in support of the MUR Uprate Program. The MUR Uprate Program does not impact the requirements, criteria, and philosophy that currently constitute the IST Program. The operating condition changes required by the MUR Uprate Program do not affect component or system design conditions; therefore, no changes to the IST Program pump or valve scope, selection criteria, tests, or acceptance standards are required. No changes are required to the IST Program as a result of the MUR Uprate Program.

VII.6.9 Radiological Environmental Monitoring Program (REMP)

No changes will be required to the REMP for monitoring the types or amounts of any effluents that may be released offsite. The current UFSAR Chapter 14 radiological accident analysis fully bounds the MUR Uprate Program. Also, the power uprate will not increase the inventory of radionuclides in the RCS above analyzed limits, nor will it affect the fuel cladding in a way that alters its structural integrity or leakage characteristics. The radionuclide activity core inventory used in the radiological consequences analyses were calculated at a core thermal power of 3588 MWt. Therefore, no changes are required to CNP's REMP as a result of the MUR Uprate Program.

VII.6.10 Radiological Dose Monitoring and Radiological Dose Control Programs

No changes will be required to the current programs for monitoring individual and cumulative occupational radiation exposure along with radiological dose control program. Plant programs and procedures will continue to ensure that dose and effluent releases are maintained within the limits of applicable regulations. The MUR Uprate Program does not change radiological source terms; therefore, the current UFSAR Chapter 14 radiological accident analysis fully bounds the 1.66 percent power uprate. The radionuclide activity core inventory used in the radiological consequences analyses were calculated at a core thermal power of 3588 MWt, which bounds plant operation following the 1.66 percent power uprate. No changes are required to the individual or cumulative occupational radiation exposure programs as a result of the MUR Uprate Program.

VII.6.11 Probabilistic Risk Assessment (PRA) Program

The CNP PRA Program was evaluated in support of the MUR Uprate Program. A review of the PRA Success Criteria (Reference VII.4) indicates that the rated thermal power used in the analyses is 3493 MWt. Therefore, the existing analyses bound the proposed MUR power uprate. Additionally, the only physical change to the plant will be the installation of an improved feedwater flow instrument (i.e., installation of LEFM CheckPlus system in the feedwater system). This modification will not affect the plant's PRA model, because flow instrumentation is below the level of detail of the plant's PRA model. Therefore, there is no impact on the PRA model or the CNP PRA Program as a result of the MUR Uprate Program.

VII.7 Mechanical Piping Design

Maximum operating pressures and temperatures will not change as result of the 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. No changes are required to the mechanical piping design and code piping analyses as a result of the MUR Uprate Program.

References (Section VII)

VII.1. Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," dated June 19, 1987

VII.2. Letter from J. F. Stang, NRC,to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MA8183 and MA8184)," dated April 25, 2000

VII.3. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MA8893 and MA8894)," dated November 21, 2000

VII.4. Letter from E. E. Fitzpatrick, I&M, to T. E. Murley, NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Individual Plant Examination Submittal Response to Generic Letter 88-20," AEP:NRC:1082B, dated May 1, 1992

VII.5. Letter from W. E. McCracken, PE, Michigan Department of Environmental Quality, to Indiana Michigan Power Company, "NPDES Permit No. M10005287," dated September 28, 2000

VIII. Changes to Technical Specifications, Protection System Settings, and Emergency System Settings

The proposed license amendment would revise the CNP Unit 1 OL and TS to increase licensed power level to 3304 MWt, or 1.66 percent greater than the current level of 3250 MWt. The proposed changes, which are indicated on the marked-up pages in Attachment 1, are described below:

1. Paragraph 2.C.(1) in OL DPR-58 is revised 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 in TS 1.3 is revised 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," are revised to limit Indicated T_{avg} at RTP (T' for Overtemperature ΔT; T'' for Overpressure ΔP) to less than or equal to 574°F and 562.1°F, respectively.

4. TS Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation," is revised to reflect the maximum allowed power for operation with inoperable MSSVs. With one inoperable MSSV per loop, the power reduction is 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 is revised to 45.5 percent (two inoperable MSSVs) and 27.4 percent (three inoperable MSSVs). In addition, the TS Bases for LCO 3.7.1 are revised to reflect this change.

5. TS Figure 3.4-2, "Reactor Coolant System Pressure – Temperature Limits Versus 60°F/HR Rate Criticality Limit and Hydrostatic Test Limit," and TS Figure 3.4-3, "Reactor Coolant System Pressure – Temperature Limits Versus Cooldown Rates," are revised to reflect the new limit of applicability of 28.4 EFPY versus 32 EFPY for these figures. The TS Bases for LCO 3/4.4.9, "Reactor Coolant System – Pressure/Temperature Limits," are revised to reflect this change.

6. TS Table 4.4-5, "Reactor Vessel Material Irradiation Surveillance Schedule," is revised by changing the removal interval for Capsule S from "32 EFPY" to "Standby." The TS Bases for LCO 3/4.4.9, "Reactor Coolant System – Pressure/Temperature Limits," are revised to reflect this change.

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
I&M is installing a new LEFM CheckPlus system at CNP Unit 1 in anticipation of approval of this proposed amendment. Installation of this system will be completed prior to implementation of the requested license amendment. The design change for the installation will include instrumentation rescaling, maintenance and operational procedure impacts, training, monitoring iso-phase bus duct temperature, and the LEFM CheckPlus system out-of-service administrative technical requirements.	Prior to implementing this license amendment and prior to raising power above 3250 MWt
Prior to implementing this uprate, a reload safety evaluation will be performed to ensure that the core design bounds the uprated condition.	Prior to implementing this license amendment and prior to raising power above 3250 MWt
Perform an analysis of the steam dump valve flow capacity at the uprated power level and implement changes/adjustments as required to ensure the valves have sufficient capacity prior to implementing the 1.66 percent power uprate.	Prior to implementing this license amendment and prior to raising power above 3250 MWt