September 9, 1994

Docket Nos. 50-315
and 50-316

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43215

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: INCREASE IN MAIN STEAM SAFETY VALVE SETPOINT TOLERANCES (TAC NOS. M84979 AND M84980)

The Commission has issued the enclosed Amendment No. 182 to Facility Operating License No. DPR-58 and Amendment No. 167 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 11, 1992, as supplemented December 17, 1993, and March 29, 1994.

The amendments increase the main steam safety valve setpoint tolerance for both units to ± 3% and revise the emergency core cooling system section for Unit 2 to reflect a thermal power limitation resulting from the small break LOCA analysis when a safety injection cross-tie is closed.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 182 to DPR-58
2. Amendment No. 167 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PM:PD31	OGC	D:PD31
NAME	CJamerson	JHickman:gll		LBMarsh
DATE	08/1/94	7 06/01/94	08/2/94	09/8/94

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cc:

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DATED: September 9, 1994

AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK
AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK

Docket File
NRC & Local PDRs
PDIII-1 Reading
J. Roe
J. Zwolinski
C. Jamerson
L. B. Marsh
J. Hickman
OGC-WF
D. Hagan
G. Hill (2)
C. Grimes, O-11F23
S. Brewer, O-8E23
ACRS (10)
OPA
OC/LFDCB
W. Kropp, R-III
SEDB

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 11, 1992, and supplemented December 17, 1993, and March 29, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 182 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Marsha Gamberoni for

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: September 9, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 182

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 7-1	3/4 7-1
3/4 7-4	3/4 7-4
B 3/4 7-1	B 3/4 7-1

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE.

APPLICABILITY: Modes 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the reactor trip breakers are opened; otherwise, be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE in accordance with Specification 4.0.5 and with lift settings as shown in Table 4.7-1. The safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING ($\pm$3%)*	ORIFICE SIZE
a. SV-1A	1065 psig	16 $in.^2$
b. SV-1B	1065 psig	16 $in.^2$
c. SV-2A	1075 psig	16 $in.^2$
d. SV-2B	1075 psig	16 $in.^2$
e. SV-3	1085 psig	16 $in.^2$

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7 PLANT SYSTEMS

BASES

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 percent of the total secondary steam flow of 14,120,000 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:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line = 1, 2 or 3.

X = Total relieving capacity of all safety valves per steam line = 4,288,450 lbs/hour.

Y = Maximum relieving capacity of any one safety valve = 857,690 lbs/hour

(109) = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 11, 1992, and supplemented December 17, 1993, and March 29, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 167, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Marsha Gamberoni for

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 9, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 5-3	3/4 5-3
3/4 7-1	3/4 7-1
3/4 7-4	3/4 7-4
B 3/4 5-1	B 3/4 5-1
B 3/4 7-1	B 3/4 7-1

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump,

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

f. All safety injection cross-tie valves open.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With a safety injection cross-tie valve closed, restore the cross-tie valve to the open position or reduce the core power level to less than or equal to 3250 MW within one hour. Specification 3.0.4 does not apply.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE.

APPLICABILITY: Modes 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the reactor trip breakers are opened; otherwise, be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE in accordance with Specification 4.0.5 and with lift settings as shown in Table 4.7-1. The safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (±3%)*	ORIFICE SIZE
a. SV-1A	1065 psig	16 $in.^2$
b. SV-1B	1065 psig	16 $in.^2$
c. SV-2A	1075 psig	16 $in.^2$
d. SV-2B	1075 psig	16 $in.^2$
e. SV-3	1085 psig	16 $in.^2$

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

If a safety injection cross-tie valve is closed, safety injection would be limited to two lines assuming the loss of one safety injection subsystem through a single failure consideration. The resulting lowered flow requires a decrease in THERMAL POWER to limit the peak clad temperature within acceptable limits in the event of a postulated small break LOCA.

3/4.7 PLANT SYSTEMS

BASES

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 of all safety valves on all of the steam lines is 17,153,800 lbs/hr which is at least 105 percent of the maximum secondary steam flow rate at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per 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:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

X = total relieving capacity of all safety valves per steam line in lbs./hours = 4,288,450

Y = maximum relieving capacity of any one safety valve in lbs./hour = 857,690

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated November 11, 1992, and supplemented December 17, 1993, and March 29, 1994, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The proposed amendments would increase the main steam safety valve (MSSV) setpoint tolerance for both plants from ± 1% to ± 3%. The TS changes include Tables 4.7-1 (Unit 1) and 3.7-4 (Unit 2), associated with TS 3.7.1.1, to reflect the increased MSSV setpoint tolerance of ± 3%. Specifically, the licensee proposes that during normal surveillance, if the valves are found to be within ± 3%, they will be within the bases of the accident analyses; however, the valves will be reset to ± 1% to prevent the accumulation of drift. This accumulation of drift could eventually put the setpoints outside the limits of the accident analyses.

The licensee also requested that the Unit 2 TS 3.5.2 be amended to reflect a thermal power limitation resulting from the small break loss-of-coolant accident (LOCA) analysis when a safety injection cross-tie valve is closed.

The December 17, 1993, supplement was provided to correct errors that the staff identified in the Small Break Loss of Coolant Accident analysis which was submitted with the November 11, 1992, letter. The March 29, 1994, supplement modified the requested amendment to include the requirement to reset the MSSV setpoints to ± 1% whenever they are found outside that range. This change was proposed in response to a staff concern. Both the December 17, 1993, and March 29, 1994, letters provided clarifying information which did not change the staff's initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The MSSVs vent steam from a point upstream of the main steam isolation valves to prevent steam system overpressurization. A bank of five valves are located on each of the four steam lines, and the relief capacity is designed such that the total steam flow from the 20 valves will bound the capacity requirement for the limiting licensing-basis event, i.e, 17.2×10^6 lbm/hr at 1186.5 psia.

The lift setpoints of the individual valves on each steamline are staggered at different pressures to minimize chattering once the valves are actuated.

Staggering the valves also minimizes the total number of valves actuated during those transients where less than the maximum relief capacity is required. This reduces the chances of a valve failing to re-seat and reduces the maintenance requirements on the valves.

2.1 Setpoint Increase Due to Tolerance of +3%

2.1.1 Non-LOCA Events

The non-LOCA accident analyses were evaluated by the licensee and include events which approach departure from nucleate boiling (DNB), boron dilution, steamline break mass and energy release to containment, and loss of heat sink events. In the current Updated Final Safety Analysis Report (UFSAR), the MSSVs are modeled as a bank of five valves with the same lift setpoint equal to the highest set valve. The evaluations and analyses for this TS change request were modelled with the staggered behavior of the MSSVs. The licensee also modelled the flow rate of each valve to ramp linearly from no flow at its lift setpoint (nominal TS setpoint plus or minus the 3% tolerance value) to full open flow at its full open point (3% above the pressure at which the valves were assumed to pop open - i.e., accumulation effect).

Events Where DNB is Approached in the Core

The D. C. Cook UFSAR DNB events that were evaluated to demonstrate that the DNB design basis is satisfied are listed below:

14.1.10	Excessive Heat Removal Due to Feedwater System Malfunction
14.1.11	Excessive Load Increase
14.2.5	Rupture of a Steam Pipe (SLB [steamline break] Core Response)
14.1.6	Loss of Reactor Coolant Flow (Includes Locked Rotor Analysis)
14.1.1	Uncontrolled RCCA [rod cluster control assembly] Bank Withdrawal from a Subcritical Condition
14.1.2	Uncontrolled RCCA Withdrawal at Power
14.1.3	RCCA Misalignment

The licensee's evaluation of these non-LOCA events indicated that the transients are not adversely impacted by the proposed change because either the events (1) do not result in the actuation of the MSSVs prior to reactor trip, (2) are core-related only and do not result in changes in the steam pressure, or (3) are cooldown events which result in a decrease in steam pressure.

Boron Dilution Event

The licensee evaluated the boron dilution event (Section 14.2.5) and concluded that the secondary system parameters do not change significantly for this event; therefore, the changes to the MSSVs do not impact this event. Consequently, the results and conclusions presented in the UFSAR remain valid.

Steamline Break Mass and Energy Releases

The steamline break mass and energy release calculations are unaffected by the proposed changes to the MSSV lift setpoints since the majority of these calculations result in depressurization of the secondary side such that the MSSVs are not actuated. For the smaller break cases that might result in a heatup, one MSSV per steam generator is sufficient (based on the existing analyses) to provide any required heat removal following a reactor trip. The licensee indicated that the secondary pressures would not exceed those presently calculated. Therefore, the licensee concluded that the existing steamline break mass and energy release calculation remain valid.

Loss of Heat Sink Events

The licensee evaluated the remaining non-LOCA events. These events, which are listed below, involve a loss of heat sink.

Loss of AC Power to the Plant Auxiliaries	14.1.12
Loss of Normal Feedwater (LONF)	14.1.9
Feedwater System Pipe Break (FLB)	14.2.8
Loss of External Load/Turbine Trip	14.1.8

The calculations that determine the amount of auxiliary feedwater (AFW) flow available assume a maximum given steam generator backpressure in order to determine the amount of AFW that can be delivered. Therefore, the transients listed above could be impacted by the increase in the MSSV setpoint tolerance. The licensee evaluated the increase in the setpoint tolerance and determined that the current UFSAR analyses remain valid for the first three above-mentioned events for both Unit 1 and Unit 2.

Loss of External Load/Turbine Trip

The only non-LOCA event that was reanalyzed, in detail, was the loss of external load/turbine trip, Section 14.1.8. This event is the limiting non-LOCA event for potential overpressurization at D.C. Cook, and, therefore, forms the design basis for the primary and secondary safety valves. The turbine trip event was analyzed using a modified version of the LOFTRAN digital computer code. All of the assumptions were the same as those used in the UFSAR analysis, except that (1) a 2% uncertainty on initial core power was assumed for Unit 1, consistent with the Improved Thermal Design Procedure methodology (WCAP-8567) and (2) the MSSVs were modeled as a bank of five valves on each steam generator with staggered lift setpoints. Four cases were considered for each unit/core type--minimum feedback with and without pressure control and maximum feedback with and without pressure control.

The licensee concluded that for Unit 1 and 2 turbine trip analyses with a +3% tolerance on the MSSV lift setpoints, (1) the analyses resulted in the most limiting cases having no worse consequences than the current UFSAR; and (2) all of the applicable acceptance criteria are met, i.e., the DNBR [DNB ratio] for each case is greater than the limit value and the peak primary and secondary pressures remain below 110% of design values at all times.

Non-LOCA Events Conclusion

The licensee reanalyzed the most limiting non-LOCA event, i.e., loss of external load/turbine trip and demonstrated that all applicable acceptance criteria are met for the +3% MSSV lift setpoint tolerance. The remaining non-LOCA events were evaluated; and, it was shown by the licensee that the results and conclusions in the Donald C. Cook Unit 1 and 2 UFSAR remain valid.

2.1.2 LOCA Evaluations

Large Break LOCA (LBLOCA)

In the current LBLOCA analysis of record the secondary system pressure does not increase; and because of this, the MSSVs are not modelled in the analyses. Consequently, the increase in the allowable MSSV setpoint tolerance for D.C. Cook will not impact the current UFSAR LBLOCA analysis.

Small Break LOCA (SBLOCA) - Unit 1

The licensee reanalyzed the SBLOCA using the approved evaluation model based on the NOTRUMP code to determine the impact of an increased MSSV setpoint tolerance of +3%. The Unit 1 peak clad temperature (PCT) is 2068°F and the Unit 2 PCT is 2125°F. These PCT are less than the acceptance criterion limit of 2200°F, and therefore satisfy the criterion.

Unit 2 is currently licensed with a thermal rated power of 3411 MWt. The safety injection (SI) system consists of two high head safety injection (HHSI) pumps (along with two centrifugal charging pumps and two low pressure safety injection pumps). Each HHSI pump discharge line splits to deliver flow to two of the four cold legs. The two discharge lines are connected by a cross-tie, enabling one pump to deliver flow to all four cold legs. When the cross-tie valve is closed (typically due to maintenance) on Unit 2, it is necessary to reduce the power level to 3250 MWt. The reduction is required because when the cross-tie valve is closed, one pump is discharging through two lines. Should an SBLOCA occur in conjunction with a single failure, the pump would have only one line to discharge through. In this state the pump is unable to deliver enough water to mitigate an SBLOCA with the plant at full power. Therefore, when the cross-tie valve is closed, the reactor power must be reduced to 3250 MWt. The licensee's SBLOCA analysis demonstrates that 3250 MWt is an acceptable power operating limit for operation with the cross-tie valve closed.

Other LOCA-Related Evaluations

The licensee also evaluated other LOCA-related issues: (1) Post-LOCA Long-Term Cooling, (2) Hot Leg Switchover to Prevent Potential Boron Precipitation, and (3) LOCA Hydraulic Forcing Functions. It was determined that the increase in the MSSV setpoint tolerance to 3% will not impact these LOCA-related issues. The staff finds this analysis acceptable.

2.2 Setpoint Decrease Due to Tolerance of -3%

Locked Rotor

The secondary steam release assumed in the locked rotor offsite dose calculations for Unit 2 could be potentially increased by an increase in the MSSV setpoint tolerance from -1% to -3%. This would result in a larger calculated dose. The licensee evaluated this condition and found that any increased dose would remain well within a small fraction of the limits in 10 CFR Part 100. This is acceptable.

Steam Generator Tube Rupture

The licensee indicated that the lowest set MSSV on each steam generator will open at 1080 psia (without tolerances). When lowered by 3%, the MSSV opens at 1047.6 psia. This resulted in a slightly higher (0.5%) equilibrium primary to secondary break flow. The licensee concluded that the steam release to the atmosphere is therefore increased by 0.2%. The licensee indicated that the associated increased doses remain well within the acceptable limits and the offsite doses for the steam generator tube rupture event remain within a small fraction of the 10 CFR Part 100 guidelines. The staff finds this analysis acceptable.

3.0 SUMMARY

The staff finds the proposed changes to TS Tables 4.7-1 (Unit 1) and 3.7-4 (Unit 2), associated with TS 3.7.1.1 acceptable in that the licensee has demonstrated, by acceptable methods, that with the proposed increase in the MSSV setpoint tolerance from ± 1% to ± 3%, the affected UFSAR analyses of record still remain within the acceptance criteria. The staff also finds that Unit 2 operation with a reduced power level of 3250 MWt when the high head cross-tie valves are closed is acceptable based on the results of the licensee's SBLOCA analysis.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10

CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (58 FR 12262). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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