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Indiana Michigan Power
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March 7, 2006

AEP:NRC:6331
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 Units 1 and 2
Docket Nos. 50-315 and 50-316
Technical Specification Change of Interlock for a Reactor Trip on Turbine Trip

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Units 1 and 2, proposes to amend Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to modify Technical Specifications (TS) to change the reactor trip on turbine trip interlock from P-7 to P-8. The Nuclear Regulatory Commission (NRC) has previously approved similar TS amendments at Indian Point Unit 3 (September 8, 1999, ML003780834), North Anna (July 18, 1989, ML013460457), Salem (June 27, 1988, ML011690022), and Braidwood/Byron (December 8, 1987, ML020850675). The proposed change decreases potentially unnecessary transients on the reactor and increases plant availability when the cause of a turbine trip is readily correctable.

Enclosure 1 provides an affirmation statement pertaining to this letter. Enclosure 2 provides I&M's evaluation of the proposed change. Enclosure 3 provides an analysis in support of the proposed change. Enclosure 3 is proprietary to Westinghouse Electric Company. Enclosure 4 provides the non-proprietary version of Enclosure 3. Attachments 1A and 1B provide TS pages marked to show changes for Unit 1 and Unit 2, respectively. Attachments 2A and 2B provide TS pages with the proposed changes incorporated.

The proprietary information in Enclosure 3 is supported by an affidavit signed by Westinghouse, the owner of the proprietary information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations. Attachment 3 contains the Westinghouse authorization letter, CAW-06-2098, accompanying affidavit, Proprietary Information Notice, and Copyright Notice for Enclosure 3.

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The proposed changes to Unit 1 and Unit 2 TS and associated plant modification can be implemented during any plant outage. I&M requests approval of the proposed amendment prior to October 1, 2006, in support of the Unit 1 Cycle 21 outage. Implementation of the amendment will be completed prior to entering Mode 1 following the Unit 1 Cycle 21 outage (Fall 2006) and Unit 2 Cycle 17 outage (Fall 2007).

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

There are no commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Supervisor, at (269) 466-2649.

Sincerely,

Joseph N. Jensen
Site Vice President

KS/rdw

Enclosures:

1. Affirmation
2. Indiana Michigan Power Company's Evaluation
3. Donald C. Cook Units 1 and 2 Turbine Trip without a Reactor Trip Transient from the P-8 Setpoint Analysis (Proprietary)
4. Donald C. Cook Units 1 and 2 Turbine Trip without a Reactor Trip Transient from the P-8 Setpoint Analysis (Non-Proprietary)

Attachments:

1A. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked To Show Changes
1B. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked To Show Changes
2A. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages With the Proposed Changes Incorporated
2B. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages With the Proposed Changes Incorporated
3. Application for Withholding Proprietary Information from Public Disclosure

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o enclosures/attachments
J. T. King, MPSC
MDEQ – WHMD/RPMWS
NRC Resident Inspector
P. S. Tam, NRC Washington, DC

Enclosure 1 to AEP:NRC:6331

AFFIRMATION

I, Joseph N. Jensen, 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

Joseph N. Jensen
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 7th DAY OF March, 2006

Notary Public

My Commission Expires 6/10/2007

Enclosure 2 to AEP:NRC:6331

INDIANA MICHIGAN POWER COMPANY'S EVALUATION

Subject: Technical Specification Change of Interlock for a Reactor Trip on Turbine Trip

1.0 DESCRIPTION

This letter is a request by Indiana Michigan Power Company (I&M) to amend Facility Operating Licenses DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant (CNP) Units 1 and 2. The proposed change modifies Technical Specifications (TS) to change the reactor trip on turbine trip from being applicable above the P-7 interlock to being applicable above the P-8 interlock. The proposed change will decrease unnecessary challenges to the reactor protection system.

2.0 PROPOSED CHANGE

TS 3.3.1, Reactor Trip System Instrumentation, Table 3.3.1-1, Function 16.a, Turbine Trip – Low Fluid Oil Pressure, and Function 16.b, Turbine Trip – Turbine Stop Valve Closure, footnotes are changed from "(e)" to "(h)." A new footnote, (h), is added which states, "Above the P-8 (Power Range Neutron Flux) interlock."

In summary, the proposed change will modify TS to change the reactor trip on turbine trip from the P-7 interlock to the P-8 interlock. Changes to TS Bases 3.3.1 are required to reflect enabling the reactor trip on turbine trip at the P-8 interlock versus the P-7 interlock. These changes will be made in accordance with the CNP Technical Specification Bases Control Program.

3.0 BACKGROUND

3.1 System Descriptions

The CNP protection system includes interlocks which automatically enable protective functions when certain operating conditions are met. The proposed TS changes modify the interlock at which a reactor trip on turbine trip is enabled from the P-7 interlock to become enabled by the P-8 interlock. The reactor trip on turbine trip is an anticipatory trip that will actuate on loss of heat removal capabilities of the secondary system following a turbine trip at a power level exceeding the conditions of the interlock. The conditions of P-7 are determined by two-out-of-four power range nuclear instruments at approximately 10 percent power or one-out-of-two first stage high pressure turbine detectors greater than the CNP TS setpoint. The conditions of P-8 are met by two-out-of-four power range nuclear instruments at approximately 31 percent power.

The P-7 interlock receives input from the power range neutron flux instrumentation and the turbine first stage pressure. The P-8 interlock only receives input from the power range neutron flux instrumentation. This represents a logic change for the reactor trip on turbine trip function. This change is acceptable because the P-8 interlock will continue to receive reliable input from the power range neutron flux instrumentation, and the accident analyses do not credit the turbine first stage pressure input to the permissive as a trip initiator or as an accident mitigation function.

The result of the proposed change is an increase in the power level at which a reactor trip would not occur on a turbine trip from approximately 10 percent reactor power to less than or equal to

31 percent reactor power. Three control systems respond to a transient involving a turbine trip without a reactor trip. They are the rod control system, steam dump control system, and Reactor Coolant System (RCS) pressure control system.

The rod control system, described in Updated Final Safety Analysis Report (UFSAR) Section 7.3.1, enables the nuclear unit to accept a 10 percent load decrease at rates up to 200 percent per minute within the load range of 25 percent to 100 percent without steam dump valves or a reactor trip. The system also enables the nuclear unit to accept a rapid load decrease of up to 40 percent, at a maximum rate of 200 percent per minute, in combination with steam dump actuation without a reactor trip. The rod control system is capable of restoring coolant average temperature to within the programmed temperature deadband, following a scheduled or unexpected change in load.

The steam dump system is described in UFSAR Section 7.3.2. The purpose of the steam dump system is to reduce RCS pressure / temperature transients following substantial turbine load reductions by bypassing main steam directly to the condensers, thereby maintaining an artificial load on the steam generators. The rod control system can then reduce the reactor temperature to a new equilibrium value without causing RCS overtemperature and/or overpressure conditions.

The steam dump system is designed to relieve steam from the steam generators to the condenser thus reducing the sensible heat in the primary system in the event of load reduction. The steam dump design capacity is approximately 26 percent to 39 percent of full load steam flow, depending upon the full load steam pressure. All steam dump steam flows to the main condensers via the steam lines.

When a load rejection occurs, if the difference between the temperature setpoint of the RCS and the actual average temperature exceeds a predetermined amount, a signal will actuate the steam dump valves to maintain the RCS temperature within control range until a new equilibrium condition is reached. The steam dump flow reduces proportionally as the control rods act to reduce the average coolant temperature. The artificial load is therefore removed as the coolant average temperature is restored to its programmed equilibrium value.

The RCS pressure control system, described in UFSAR Section 7.3.2, maintains system pressure at a constant value by using either the pressurizer heaters (in the water region) or the spray (in the steam region). Two groups of electrical immersion heaters are located near the bottom of the pressurizer; one controls small pressure variations due to heat losses and the other is turned on when the pressurizer pressure controller signal is below a given value. A spray nozzle is located in the upper portion of the pressurizer cavity. Spray is initiated when the pressure controller signal is above a given setpoint, and spray rate increases proportionally with increasing pressure. Steam is condensed by the spray, which will return the pressurizer pressure to its program value.

3.2 Reason for Requesting Amendment

Changing the interlock for a reactor trip on a turbine trip to a permissive with a higher setpoint decreases potentially unnecessary transients on the reactor and increases plant availability when the cause of a turbine trip is readily correctable.

4.0 TECHNICAL ANALYSIS

An evaluation has been performed to determine the impact of increasing the power level at which a turbine trip without a reactor trip would occur on the pressurizer power-operated relief valves (PORVs). Following the Three Mile Island accident, the Nuclear Regulatory Commission (NRC) expressed concern about the implementation of blocking the reactor trip on turbine trip function on a permissive with an increased setpoint because of the potential to increase the probability of a stuck open pressurizer PORV. The NRC position is addressed in NUREG-0737, Item II.K.3.10. In NUREG-0737, the NRC has stated that any modifications to anticipatory trips should not be made until it has been shown by a licensee that the probability of a small break loss of coolant accident (LOCA) resulting from a stuck-open PORV is substantially unaffected by the modification. To satisfy the NRC requirements stated in Item II.K.3.10, a plant-specific analysis (Enclosure 3) was performed to show that the implementation of the block of reactor trip on turbine trip at the P-8 setpoint will not result in challenges to the pressurizer PORVs. The analysis was performed using the P-8 setpoint of 31 percent power for a turbine trip without a reactor trip. The results show that for this setpoint value, the pressurizer PORVs will not be challenged.

A review of UFSAR Chapter 14 safety analyses has been performed in order to confirm that the safety analyses results are not adversely affected by this proposed change of moving the reactor trip on turbine trip function from P-7 (approximately 10 percent power) to P-8 (less than or equal to 31 percent power). The following provides an assessment of the proposed change with respect to Unit 1 and Unit 2 CNP safety analyses and evaluations.

1. UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY (RCCA) BANK WITHDRAWAL FROM A SUBCRITICAL CONDITION

Event Definition: An RCCA bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks, resulting in a power excursion. Should a continuous RCCA withdrawal be initiated, the transient will be terminated by the following reactor trip functions:

1. Source range neutron flux level trip,
2. Intermediate range neutron flux level trip,
3. Power range neutron flux level trip (low setting), and
4. Power range neutron flux level trip (high setting).

Plant Operating Conditions: The plant is assumed to be at the no-load reactor coolant average temperature.

Effect of Proposed Change: In this scenario, the reactor is not critical and the turbine generator is not on-line. The direct reactor trip from turbine trip is not credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

2. UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER

Event Definition: This event is defined as the inadvertent addition of positive reactivity to the core caused by the uncontrolled withdrawal of an RCCA bank(s) while at power. The automatic features of the reactor protection system which prevent core damage in an RCCA bank withdrawal incident at power include the following:

1. Power range neutron flux level trip (high setting),
2. Overtemperature Delta T,
3. Overpower Delta T,
4. High pressure reactor trip, and
5. High pressurizer water level reactor trip.

Plant Operating Conditions: Analyses cases are evaluated for initial reactor power at 10 percent, 60 percent, and 100 percent of rated thermal power.

Effect of Proposed Change: The direct reactor trip from turbine trip is not credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

3. RCCA MISALIGNMENT

Event Definition: RCCA accident misalignment accidents include:

1. A dropped RCCA,
2. A dropped RCCA bank, and
3. Statically misaligned RCCA.

Plant Operating Condition: Analyses are performed at nominal full power conditions.

Effect of Proposed Change: There is no reactor trip credited in any of the three cases of the analysis. The reactor trip on turbine trip function is not credited for this event as either a primary or backup trip. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

4. CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Event Definition: This event is the inadvertent dilution of the RCS boron concentration.

Plant Operating Condition: Boron dilution during shutdown, refueling, startup, and power operations were examined.

Effect of Proposed Change: The reactor trip on turbine trip function is not credited for this event as either a primary or backup trip. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

5. LOSS OF REACTOR COOLANT FLOW

Event Definition: The loss of flow incident can result from a mechanical or electrical failure in a reactor coolant pump (RCP), or from a fault in the power supply of these pumps.

The following trip circuits provide the necessary protection against a loss of coolant flow incident:

1. Undervoltage or underfrequency on pump power supply buses,
2. Pump circuit breaker opening (Unit 1 only), and
3. Low reactor coolant flow.

Plant Operating Condition: The following loss of flow cases are analyzed:

1. Loss of four pumps from nominal full power conditions with four loops operating,
2. Loss of one pump from nominal full power conditions with four loops operating, and
3. Locked Rotor Accident from nominal full power conditions with four loops operating.

Effect of Proposed Change: The low primary coolant loop flow, RCP undervoltage, RCP underfrequency, and RCP breaker position reactor trip functions provide the necessary protection for this event. These trips are not affected by the direct reactor trip on turbine trip interlock setpoint.

Therefore, the proposed change has no effect on this accident scenario group and the conclusions of the UFSAR remain valid.

6. STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

Event Definition: The inadvertent startup of an idle loop while operating would result in the sudden introduction of colder water into the core from the idle loop which could cause an unplanned reactivity insertion and power increase.

Plant Operating Condition: The CNP TS preclude operation of the plant with one or more loops out of service.

Effect of Proposed Change: This event is not part of the plant's licensing basis. Therefore, the proposed change has no effect on this accident scenario.

7. LOSS OF EXTERNAL ELECTRICAL LOAD

Event Definition: The loss of external electrical load and/or turbine trip event is defined as a complete loss of steam load from full power without a direct reactor trip, or a turbine trip without a direct reactor trip. It may result from a trip of the turbine generator or in an unlikely opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large nuclear steam supply system load reduction by the action of the turbine control. Trip signals are expected due to:

1. High pressurizer pressure,
2. Overtemperature delta-T,
3. High pressurizer water level, and
4. Low-low steam generator water level.

Plant Operating Condition: The analysis assumes a complete loss of steam load from full power with no credit taken for the direct reactor trip on turbine trip.

Effect of Proposed Change: Protection for this event is provided by the Overtemperature delta-T, high pressurizer pressure, high pressurizer water level, or low-low steam generator water level signals. The loss of external electrical load/turbine trip event from a full power condition bounds a turbine trip with no subsequent reactor trip from 10 percent power (P-7) as well as from 31 percent power (P-8). Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

8. LOSS OF NORMAL FEEDWATER FLOW

Event Definition: The design basis loss of normal feedwater event is defined as a reduction in the capability of the secondary system to remove heat generated in the reactor core.

The reactor trip is initiated by:

1. Low - low steam generator water level trip, and
2. Low feedwater flow signal in any steam generator (Unit 2 only).

Plant Operating Condition: A complete loss of main feedwater flow is assumed to occur from 102 percent of rated thermal power.

Effect of Proposed Change: The credited trip signals are not affected by the change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

9. EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

Event Definition: This event may result from an increase in feedwater flow to one or more of the steam generators or a decrease in feedwater temperature. This event will result in an increase in the heat transfer rate from primary to secondary in the steam generators and a consequential reduction in primary system temperature and pressure.

Plant Operating Condition: This event is analyzed at power levels corresponding to zero and full load.

Effect of Proposed Change: Opening of either a low pressure heater bypass valve or a high pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. This reduction in feedwater temperature results in an increase in heat load on the primary system of less than 10 percent of full power. The increased thermal load would result in a transient very similar (but of reduced magnitude) to that presented in Item 10, below, for an excessive increase in secondary steam flow incident, which evaluates the consequences of a 10 percent step load increase. Reactor trip from turbine trip is not credited in Item 10. Therefore, the results of the feedwater temperature reduction analysis are not affected by the proposed change.

For the feedwater flow increase analysis, the reactor trip on turbine trip function is currently disabled below 10 percent power via the P-7 setpoint. The proposed change is to move the reactor trip on turbine trip from P-7 to P-8, which results in the trip function being disabled below 31 percent power (the P-8 setpoint). For the full and zero load cases this analysis is unaffected by the proposed change since the reactor trip on turbine trip function is still available at full load and remains unavailable at the zero load case before and following the proposed change.

For power levels below 31 percent, where the reactor trip on turbine trip function is to be disabled, the feedwater flow increase event would be no more severe than the 100 percent power case. Similar to the 100 percent power case, a feedwater isolation and turbine trip would be generated on a high-high steam generator water level signal. However, since the reactor would not be tripped on a turbine trip signal, the transient would progress into a heatup event and the reactor would eventually trip on a low-low steam generator water level signal. Although credited in the analysis for the full power cases, the reactor trip on turbine trip is not a critical function that is required in order to get acceptable results. The minimum departure from nucleate boiling ratio would be essentially unchanged if the reactor trip was not assumed to occur on turbine trip.

Based on the above, the proposed change does not invalidate the results of the analysis and the conclusions of the UFSAR remain valid.

10. EXCESSIVE LOAD INCREASE INCIDENT

Event Definition: An excessive load increase event is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Protection against an excessive load increase accident is provided by the following reactor protection signals (RPS) signals:

1. Overpower delta-T,
2. Overtemperature delta-T,
3. Power range high neutron flux, and
4. Low pressurizer pressure.

Plant Operating Condition: The analysis is performed at 100 percent power.

Effect of Proposed Change: Although the RPS is assumed to be operable, a reactor trip does not occur in this analysis. Instead, the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow. A reactor trip from turbine trip is not among the credited trip actuations. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

11. LOSS OF ALL ALTERNATING CURRENT (AC) POWER TO THE PLANT AUXILIARIES

Event Definition: A complete loss of non-emergency power (i.e. offsite power) may result in the loss of all power to the plant auxiliaries: i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of onsite non-emergency AC distribution system. The reactor trip is initiated by low - low steam generator water level trip.

Plant Operating Condition: The plant is initially operating at 102 percent of rated thermal power.

Effect of Proposed Change: The direct reactor trip from turbine trip is described in the analysis only because it is expected to occur. The reactor trip on turbine trip was not required for core protection for this event and was not credited in the analysis. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

12. STEAM GENERATOR TUBE RUPTURE (SGTR)

Event Definition: This event is assumed to be the complete severance of a single tube. The reactor trip signal is generated by low pressurizer pressure or overtemperature delta-T.

Plant Operating Condition: The accident is assumed to take place at full reactor power.

Effect of Proposed Change: The trip mechanisms for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

13. RUPTURE OF A STEAM PIPE

Event Definition: A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. Such a release may result from either the opening of a steam generator relief or safety valve, or from a steam system pipe break. Protection for this event is provided by the overpower reactor trips (neutron flux and Delta-T), and the reactor trip occurring in conjunction with receipt of the Safety Injection Signal.

Plant Operating Condition: The analysis assumes that the reactor is initially at hot shutdown conditions with the most reactive RCCA in a fully withdrawn position.

Effect of Proposed Change: The limiting zero power analysis does not specifically credit the reactor trip system. Only the Engineered Safety Features Actuation System is needed to limit the consequence of the analyzed events. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

14. RUPTURE OF A CONTROL ROD MECHANISM HOUSING - RCCA EJECTION

Event Definition: This event is an assumed failure of a control rod mechanism pressure housing such that the RCS pressure would eject the control rod and drive shaft. The reactor will trip on the power range high neutron flux low setpoint, the high setpoint, or the high rate of neutron flux increase setpoint.

Plant Operating Condition: Both full and zero power cases are analyzed.

Effect of Proposed Change: The trip mechanisms for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

15. MAJOR RUPTURE OF MAIN FEEDWATER PIPE (FEEDLINE BREAK) (UNIT 2 ONLY)

Event Definition: A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. A reactor trip may occur on any of the following conditions:

1. High pressurizer pressure,
2. Overtemperature delta T,
3. Low-low steam generator water level in any steam generator,
4. Safety injection signals from any of the following:
   a. Low steam line pressure,
   b. High containment pressure-low setpoint, and
   c. High steam line differential pressure.

Plant Operating Condition: The plant is initially operating at 102 percent of uprated thermal power.

Effect of Proposed Change: The trip mechanisms for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

16. LOCA AND LOCA-RELATED ANALYSES

Event Definition: A LOCA is the result of a pipe rupture of the RCS pressure boundary. The following LOCA-related analyses have been reviewed for impact by the proposed change:

1. Large and small break LOCA,
2. Reactor vessel and loop LOCA blowdown forces,
3 Post-LOCA long term core cooling subcriticality, and
4. Post-LOCA long term core cooling minimum flow and hot leg switchover to prevent further boron precipitation.

Plant Operating Condition: Full power conditions are assumed.

The UFSAR small break LOCA and large break LOCA analyses only credit protection from the low pressurizer pressure reactor trip. Furthermore, the main turbine trip signals do not monitor any parameter that would provide useful protection from a small break or large break LOCA.

Effect of Proposed Change: The change does not affect the normal plant operating parameters, the safeguards systems actuation or accident mitigation capabilities important to LOCA, or the assumptions used in the LOCA-related accidents. The change does not create conditions more limiting than those assumed in these analyses.

17. CONTAINMENT INTEGRITY EVALUATION (SHORT TERM / LONG TERM)

Event Definition: Containment integrity and subcompartment safety analyses are performed for ice condenser design containments to quantify both the margin in the containment design pressure and ice condenser performance requirement including the minimum ice mass or the maximum flow blockage.

Plant Operating Condition: Full power conditions are assumed.

Effect of Proposed Change: The turbine trip is not credited in the containment integrity analyses. The proposed change does not adversely affect the short term or long term mass and energy releases of the containment analyses. Therefore, the conclusions presented in the UFSAR remain valid with respect to the containment analyses.

18. MAIN STEAMLINE BREAK (MSLB) MASS AND ENERGY RELEASE ANALYSES

Event Definition: Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures.

Plant Operating Condition: Full power conditions are assumed.

Effect of Proposed Change: The turbine trip is not credited in the CNP UFSAR MSLB analyses. The conclusions presented in the UFSAR remain valid with respect to MSLB mass and energy release rates and steam mass release calculations.

SUMMARY

The CNP Unit 1 and Unit 2 UFSAR analyses of record do not credit the direct reactor trip from turbine trip for the protection of fission product barriers. The conclusions of the UFSAR will remain valid following the proposed change for the reactor trip from turbine trip interlock from P-7 to P-8.

The P-7 interlock receives input from the power range neutron flux instrumentation and the turbine first stage pressure. The P-8 interlock only receives input from the power range neutron flux instrumentation. This represents a logic change for the reactor trip on turbine trip function. This change is acceptable because the P-8 interlock will continue to receive reliable input from the power range neutron flux instrumentation, and the accident analyses do not credit the turbine first stage pressure input to the permissive as a trip initiator or as an accident mitigation function.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Indiana Michigan Power Company (I&M) has evaluated whether a significant hazards consideration is involved with the proposed amendment 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

   The proposed change revises the setpoint at which a reactor trip will occur by changing the interlock at which it is enabled from the P-7 interlock, at approximately 10 percent power, to the P-8 interlock, at less than or equal to 31 percent power. The P-7 and P-8 interlocks are not accident initiators and the change to the reactor trip setpoint does not create any new credible single failure. An analysis has shown that a turbine trip without a reactor trip at 31 percent power or below does not challenge the pressurizer power operated relief valves (PORVs), thereby not adversely affecting the probability of a small break loss of coolant accident due to a stuck open PORV. The consequences of accidents previously evaluated are unaffected by this change because no change to any accident mitigation scenario has resulted and there are no additional challenges to fission product barrier integrity.

   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 changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed change to the power level at which a reactor trip on turbine trip is enabled does not adversely affect previously identified accident initiators and does not create any new accident initiators. The change does not affect how the associated trip function operates. No new single failures or accident scenarios are created by the proposed change and the proposed change does not result in any event previously deemed incredible being made credible.

   Therefore, the proposed change does 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

   No safety analyses were changed or modified as a result of the proposed change in reactor trip setpoint. All margins associated with the current safety analyses acceptance criteria are unaffected. The current safety analyses remain bounding. The safety systems credited in the safety analyses will continue to be available to perform their mitigation functions. The proposed change does not affect the availability or operability of safety-related systems and components.

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

Based on the above, I&M concludes that the proposed amendment presents 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

10 CFR 50.36 (c) (2) (ii), stipulates that a technical specification limiting condition for operation must be established for each item meeting one or more of the following criteria:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The change to the enabling interlock for a reactor trip on turbine trip continues to meet this regulation. That is, the reactor trip function remains available as an anticipatory trip following the loss of heat removal capability of the secondary system to minimize the pressure/temperature transient on the reactor.

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 Nuclear Regulatory Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health or safety of the public.

6.0 ENVIRONMENTAL CONSIDERATIONS

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

None

8.0 PRECEDENT

The NRC has approved similar submittals for plants changing the interlock at which the reactor trip on turbine trip is enabled from the P-7 interlock to the P-8 interlock.

Indian Point Unit 3	Accession No. ML003780834
North Anna	Accession No. ML013460457
Salem	Accession No. ML011690022
Braidwood/Byron	Accession No. ML020850675

Enclosure 3 to AEP:NRC:6331

Donald C. Cook Units 1 and 2 Turbine Trip without a Reactor Trip Transient from the P-8 Setpoint Analysis (Proprietary)