INDIANA
MICHIGAN
POWER®
A unit of American Electric Power

Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
AEP.com

May 31, 2006

AEP:NRC:6331-01
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
Application for Amendment to Revise Technical Specification Surveillance Requirement for Channel Calibration of Overtemperature Differential Temperature and Overpower Differential Temperature Reactor Protection System Functions

Dear Sir or Madam:

Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, proposes to modify the Technical Specification (TS) Surveillance Requirement (SR) for channel calibration of the overtemperature differential temperature (OTΔT) and overpower differential temperature (OPΔT) reactor protection system functions. I&M proposes to delete existing Note 1, which requires verification of the Reactor Coolant System (RCS) resistance temperature detector (RTD) bypass loop flow rate. During the fall 2006 Unit 1 refueling outage, I&M plans to remove the RTD bypass piping and install fast response thermowell-mounted RTDs located in the RCS loop piping. I&M also proposes an editorial change to remove the note number from existing Note 2, since it will become the only note for that SR. Associated TS Bases will be changed in accordance with the CNP Bases Control Program.

I&M expects that removal of the RTD bypass piping will result in a reduction of approximately 30 person-rem in radiation exposure to personnel performing work in containment during refueling outages. Removal of the RTD bypass piping will also reduce the potential for RCS leakage and reduce refueling outage costs. I&M intends to perform the same plant modification on Unit 2. I&M plans to request a similar amendment for Unit 2 via a separate submittal following completion of the necessary Unit 2 engineering and analysis work.

Enclosure 1 to this letter provides an affirmation affidavit pertaining to the proposed amendment. Enclosure 2 provides a detailed description and safety analysis to support the proposed amendment, including an evaluation of significant hazards considerations pursuant to 10 CFR 50.92(c), and an environmental assessment. Attachment 1 provides the affected TS page marked to show changes. Attachment 2 provides the TS page with the proposed changes incorporated.

A001

I&M requests Nuclear Regulatory Commission (NRC) approval of the proposed amendment by September 8, 2006, to allow implementation of the RTD bypass plant modification during the fall 2006 Unit 1 refueling outage. I&M requests implementation of the proposed amendment be required prior to entry into Mode 2 during that outage.

I&M acknowledges the established regulatory practice to submit license amendment requests at least one year in advance of the requested approval date. The need for this license amendment was not recognized during the initial planning stages of the associated plant modification. The note proposed for removal by this amendment request did not exist in the CNP TS until the improved standard TS were implemented in September 2005. Consequently, the associated engineering work was scheduled for completion to support implementation of the modification during the fall 2006 Unit 1 refueling outage rather than support submittal of an amendment request a year in advance of that outage. I&M regrets the need to request an expedited NRC approval. Nevertheless, I&M considers that the significant reduction in personnel radiation exposure described above justifies approval in time to allow implementation of the modification during the fall 2006 Unit 1 refueling outage.

Copies of this letter and its enclosures and 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.

This letter contains no new regulatory commitments. 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

JRW/rdw

Enclosures:

1. Affirmation.
2. Application for Amendment to Revise Technical Specification Surveillance Requirement for Channel Calibration of Overtemperature Differential Temperature and Overpower Differential Temperature Reactor Protection System Functions.

Attachments:

1. Unit 1 Technical Specification Page Marked to Show Proposed Change
2. Unit 1 Proposed Technical Specification Page

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

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 31st DAY OF May, 2006

Regan D. Wendzel
Notary Public

My Commission Expires REGAN D. WENDZEL
Notary Public, Berrien County, MI
My Commission Expires Jan. 21, 2009

Application for Amendment to Revise Technical Specification Surveillance Requirement for Channel Calibration of Overtemperature Differential Temperature and Overpower Differential Temperature Reactor Protection System Functions

1.0 DESCRIPTION

Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, proposes to modify the Technical Specification (TS) Surveillance Requirement (SR) for channel calibration of the overtemperature differential temperature (OTΔT) and overpower differential temperature (OPΔT) reactor protection system (RPS) functions. I&M proposes to delete existing Note 1, which requires verification of the Reactor Coolant System (RCS) resistance temperature detector (RTD) bypass loop flow rate. During the fall 2006 Unit 1 refueling outage, I&M plans to remove the RTD bypass piping and install fast response thermowell-mounted RTDs located in the RCS loop piping. I&M also proposes an editorial change to remove the note number from existing Note 2, since it will become the only note for that SR. Associated TS Bases will be changed in accordance with the CNP Bases Control Program.

2.0 PROPOSED CHANGE

Existing Note 1 of Unit 1 TS SR 3.3.1.15 will be deleted. The note number will be removed from existing Note 2 of Unit 1 TS SR 3.3.1.15.

Attachment 1 provides the affected TS page marked to show changes. Attachment 2 provides the TS page with the proposed changes incorporated.

3.0 BACKGROUND

Existing Design

CNP Unit 1 has four reactor coolant loops. In the existing design, an RTD bypass manifold system is used to obtain representative hot leg and cold leg temperatures in each reactor coolant loop. Each loop has separate hot leg and cold leg bypass inlet piping connections and manifolds. A representative hot leg temperature is obtained by mixing flow from three scoop connections. These scoops extend into the flow stream (at locations 120 degrees apart in the cross-sectional plane) on each reactor coolant hot leg. Each scoop has five flow holes which sample the hot leg coolant. Flow for the cold leg bypass manifold is obtained downstream of the reactor coolant pump (RCP) discharge. The hot and cold bypass manifold piping for each loop join to form a common discharge line. The combined flow discharges to the suction side of the RCP. The existing RTD bypass manifold system consists of approximately 500 feet of reactor coolant pressure boundary piping (not including instrument tubing), 44 valves, 38 pipe hangers (including nine snubbers), eight sets of flanges, and eight RTD manifolds.

In the existing design, the RTDs extend directly (without thermowells) into the bypass manifold fluid flow. This minimizes the response time of the RTDs. The RTD outputs are used to calculate the loop differential temperature (ΔT) and loop average temperature (Tavg) signals that are used by the RPS and control systems. The loop ΔT and/or loop Tavg signals are used in the OTΔT and OPΔT functions in the RPS; the Steam Line Isolation on High Steam Flow Coincident with Low-Low Tavg and P-12 Low-Low Tavg Interlock functions in the Engineered Safety Feature Actuation System (ESFAS); and in verifying RCS total flow. In addition, the RTD outputs are used for alarms and indication, rod control, turbine runback, pressurizer level, and other control systems.

The existing RTD bypass manifold system design was developed to resolve concerns with temperature streaming (temperature gradients) within the RCS hot legs. The temperature streaming in the hot leg piping was a result of incomplete mixing of coolant leaving various regions of the reactor core at different temperatures. The bypass manifold system compensates for the temperature streaming by sampling the primary coolant through scoop tubes and mixing the primary coolant within the bypass manifold to develop an average RCS hot leg temperature for the loop. The bypass manifold system also limits the velocity of the coolant flow to the RTDs, and allows RTD replacement without the need to drain the RCS loops.

The coolant velocity through the RTD bypass piping system is low relative to the RCS loop velocity. As a result, the bypass piping, valves, and manifolds become collection points for activated corrosion products. These components tend to become radiological hot spots, which significantly increases the general area radiation levels. Due to their proximity to the RCP and steam generator related work locations, the RTD manifolds are significant contributors to personnel radiation dose during unit outages. To reduce this dose contribution, temporary lead shielding is installed during each outage. Installation and removal of this shielding is labor intensive, negatively impacts other work activities requiring material transport through the containment hatch, and involves radiation exposure to personnel installing the shielding. I&M estimates that the RTD bypass piping system contributes approximately 40 percent (30 person-rem) of the overall dose each refueling outage. Additional concerns associated with the existing system include the large number of components requiring maintenance, repair, surveillance, and testing. Review of industry operating experience has identified the RTD bypass manifold valves as sources of RCS leakage events. Although the existing RTD bypass manifold system has been adequately performing its intended function, elimination of the system would reduce personnel radiation exposure, outage costs, and maintenance concerns.

Modification

I&M plans to eliminate the RTD bypass manifold system from all four RCS loops in Unit 1. All piping and valves associated with this system will be removed. Associated RTD bypass system pipe supports will also be removed. This is a standard industry modification which has been implemented successfully by other Westinghouse plants.

The three hot leg scoops will be modified to accept new thermowells, which will contain new, fast-response RTDs (Model N9004E, manufactured by Weed Instruments Incorporated). A hole will be drilled through the end of each scoop to facilitate flow past the RTD. Water will enter through the existing openings, flow past the RTD, and exit through the new hole at the end of the scoop. The existing cold leg RTD bypass piping nozzle will also be modified to accept an RTD thermowell. The cold leg RTD thermowell will be inserted directly in the coolant flow path.

The RPS will be modified to calculate an average hot leg temperature for each loop, using the three new RTD signals. The electronically-averaged temperature will function similar to the single temperature input provided by the existing manifold RTD. The new fast response thermowell RTD system will provide the same degree of independence and redundancy as the existing system.

4.0 TECHNICAL ANALYSIS

The proposed change deletes an existing TS note that requires verification of the RTD bypass loop flow rate as part of the OTΔT and OPΔT channel calibrations. The OTΔT reactor trip function provides primary protection against departure–from-nucleate-boiling (DNB) during transients in Westinghouse pressurized water reactors. The measured ΔT is used as an indication of reactor power and is compared to the OTΔT setpoint, which varies depending upon the measured Tavg, pressurizer pressure, and the reactor axial flux difference signal. If the measured ΔT exceeds the OTΔT setpoint in more than one loop, a reactor trip signal is generated.

The OPΔT reactor trip function is designed to protect against a high fuel rod power density and thus preclude fuel centerline melting. The measured ΔT is used as an indication of reactor power and is compared to the OPΔT setpoint, which varies depending upon the indicated Tavg. If the measured ΔT exceeds the OPΔT setpoint in more than one loop, a reactor trip signal is generated.

Response Time

Replacement of the existing RTD bypass system with the new fast response thermowell RTD system will affect the RCS temperature measurement response characteristics that are currently modeled as part of the OTΔT and OPΔT reactor trips credited in certain non-loss of coolant accident (LOCA) analyses. As shown in Unit 1 Table 14.1-2 of the Updated Final Safety Analysis Report (UFSAR), the OTΔT and OPΔT RPS functions have a total time delay of 8 seconds assumed in the analyses. The following table shows how this 8-second assumption is maintained by the existing RTD bypass system and how it will be maintained by the new fast response thermowell RTD system.

Response Time Parameters for RCS Temperature Measurement		
Component	Existing RTD Bypass System (seconds)	New Fast-Response Thermowell RTD System (seconds)
RTD bypass piping transport and thermal lag.	4	N/A
RTD response time	2	4
Electronics signal processing, reactor trip signal, trip breaker opening, and rod cluster control assembly gripper release.	2	2
Total Response Time	**8**	**Less than or equal to 8**

The OTΔT and OPΔT reactor trip model used in the non-LOCA analyses includes a 6-second first order lag time for the temperature sensor response plus a 2-second delay for the electronic time response, for a total time of 8 seconds from the time that the setpoint is reached to the loss of stationary gripper coil voltage, i.e., when the rod cluster control assemblies (RCCAs) are free to fall. As shown in the above table, the 6-second first order lag includes the RTD response time, and the bypass pipe coolant transport and thermal heatup time. The 2-second electronic delay accommodates electronic signal processing, reactor trip signal, trip breaker opening, and RCCA gripper release.

Due to the new fast response thermowell RTD system, the individual components that comprise the OTΔT and OPΔT reactor trip response time assumed in the non-LOCA analyses will be altered. Although the individual component times will be different, the total OTΔT and OPΔT reactor trip response time assumption of 8 seconds will be met. As described below, evaluations were performed to assess the impact that a change in the OTΔT and OPΔT component response times would have on non-LOCA accidents analyzed in the UFSAR that rely on OTΔT or OPΔT trips for reactor protection.

Uncontrolled RCCA Bank Withdrawal at Power (UFSAR Unit 1 Section 14.1.2)

An uncontrolled RCCA bank withdrawal at power (RWAP) event can occur due to an improper operator action or a malfunction of the rod control system, and will result in an increase in the core heat flux due to the positive reactivity addition. The event would be terminated by either a high neutron flux or OTΔT reactor trip function. The event is classified as a Condition II event, i.e., an incident of moderate frequency, as defined by the American National Standard ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of stationary Pressurized Water Reactor Plants." The event is analyzed to demonstrate that the DNB design basis is satisfied.

A spectrum of reactivity insertion rates from several different power levels (100 percent (%), 60%, and 10% power) was considered for both beginning and end of life conditions. The cases that result in a reactor trip from the high neutron flux setpoint are unaffected by the change in the OTΔT response time components. For lower reactivity insertion rates, the OTΔT reactor trip function provides the primary protection, as indicated by Unit 1 Figures 14.1.2-7 through 14.1.2-9 in the UFSAR. A CNP-specific sensitivity analysis was performed in which the limiting DNB ratio (DNBR) cases from the existing UFSAR RWAP analyses were run, assuming various combinations of lags and delays with a total response time of 8 seconds. Based on the results of the existing UFSAR RWAP analyses and the sensitivity analysis, it was shown that there is no significant effect on the calculated minimum DNBR. In all cases, the minimum DNBR remained above the safety analysis DNBR limit. Therefore, the DNB design basis will be met and the conclusions in UFSAR Unit 1 Section 14.1.2 will remain valid.

Chemical and Volume Control System Malfunction (UFSAR Unit 1 Section 14.1.5)

The chemical and volume control system malfunction (boron dilution) analysis is performed to demonstrate that sufficient time is available following initiation of the event to allow an operator to determine the cause of the inadvertent dilution and take corrective action before shutdown margin is lost. The event is classified as a Condition II event by ANSI N18.2-1973. This event is bounded by another Condition II event, the RWAP event, with respect to Condition II criteria, such as ensuring that the DNB design basis is satisfied, and maintaining peak primary and secondary pressures less than 110% of design.

A change in OTΔT individual component response time for the boron dilution event presented in the UFSAR would only potentially affect the case analyzed at full power with manual rod control. For this case, the operator action time is measured from the time of reactor trip (on OTΔT) until a loss of the plant shutdown margin. Since a boron dilution transient at full power with manual rod control results in a reactivity insertion essentially equivalent to an RWAP event, a conservative time for reactor trip was selected from the times calculated for the full power RWAP analysis. Based on sensitivity runs generated for the RWAP event, where various combinations of RTD lag and delay response times were modeled, the time of rod motion would be delayed by a maximum of 3 seconds in the full power cases.

The acceptance criterion in the CNP accident analysis for a boron dilution event at full power with manual rod control is that a minimum of 15 minutes be available for operator action. The interval from the time of reactor trip on OTΔT to loss of shutdown margin calculated for the UFSAR licensing basis boron dilution event at full power with manual rod control is 43 minutes for Unit 1. Thus, significant margin is available. A 3-second increase in OTΔT reactor trip response does not affect the results of the licensing basis analyses. Therefore, the operator action time criterion continues to be met, and the conclusions in UFSAR Unit 1 Section 14.1.5 will remain valid.

Loss-of-Load/Turbine Trip (UFSAR Unit 1 Section 14.1.8)

A loss of external electrical load can result from an abnormal variation in the network frequency, a trip of the turbine, or the spurious closure of the turbine stop or control valves or steamline isolation valves. If a failure of the steam dump valve system also occurs, the sudden reduction in steam flow will result in an increase in the pressure and temperature in the steam generators. Heat transfer will be reduced, causing the reactor coolant temperature and pressure to rise. The loss-of-load event is analyzed to confirm that the pressurizer and steam generator safety valves are adequately sized to prevent over pressurization of the RCS and steam generators, and to ensure that the DNB design basis is satisfied. The event is classified as a Condition II event by ANSI N18.2-1973.

The high pressurizer pressure, low steam generator level, and OTΔT reactor trip functions provide protection for a loss-of-load/turbine trip event. Four cases were analyzed for this event. Two limiting cases were analyzed for peak RCS pressure concerns. These two cases assumed no pressurizer pressure control for both beginning and end of life reactivity feedback conditions, and a reactor trip results from the high pressurizer pressure RPS function. Therefore, these cases are not affected by the OTΔT response time modeling.

The other two cases were analyzed to demonstrate that the DNB design basis is satisfied for limiting loss-of-load/turbine trip events. These cases assumed pressurizer pressure control for both beginning and end of life reactivity feedback conditions. In these two cases, the beginning of life (minimum) feedback case is terminated by the OTΔT reactor trip function. For this case, explicit sensitivities were generated for CNP, with modeling of various combinations of lag and delays. The results of these sensitivity studies indicated there would be no significant effect on the calculated minimum DNBR. Therefore, the minimum DNBR remains above the safety analysis limit, the DNB design basis is met, and the conclusions in UFSAR Unit 1 Section 14.1.8 will remain valid.

Steam Line Isolation on High Steam Flow Coincident with Low-Low Tavg and P-12 Low-Low Tavg Interlock ESFAS Functions

As indicated in TS Table 3.3.2-1 (Functions 4.e and 8.c) and UFSAR Table 7.2-7, there are no response time requirements for the Steam Line Isolation on High Steam Flow Coincident with Low-Low Tavg function or the P-12 Low-Low Tavg Interlock ESFAS functions.

Response Time Testing

The RTD manufacturer will perform time response testing of each RTD and thermowell prior to installation at CNP. The RTDs and thermowells must exhibit a response time bounded by the values shown in the preceding table. In addition, response time testing of the RTDs will be performed in-situ in accordance with TS SR 3.3.1.19. This testing will demonstrate that the RTDs can satisfy the response time requirement when installed in the plant.

Instrument Uncertainty Considerations

Instrument uncertainty calculations have been performed for the new fast response thermowell RTD system in Unit 1. The uncertainty calculations include a measurement term to address the effects of hot leg temperature streaming. Temperature streaming will exist in the hot leg due to inadequate mixing of coolant leaving various regions of the reactor core. The use of three flow scoops located at 120 degree increments along the circumference of the hot leg loop pipe reduces the streaming effects. The effects of cold leg streaming are not included in the calculation because it is considered in the safety analysis margin. As described below, the results of the instrument uncertainty calculations were used to determine that the existing allowable values in TS for OTΔT, OPΔT, and Tavg will remain bounding for the new fast response thermowell RTD system.

Reactor Trip System TS Table 3.3.1-1, Functions 6 and 7, and Notes 1 and 2, specify the allowable values for the OTΔT and OPΔT RPS setpoint. I&M calculations have confirmed that the existing OTΔT and OPΔT TS allowable values will bound the instrument uncertainty of the new fast response thermowell RTD system. ESFAS Instrumentation TS Table 3.3.2-1, Functions 4.e and 8.c, specify the allowable values for the Low-Low Tavg setpoint. I&M calculations have confirmed that the existing Low-Low Tavg TS allowable values will bound the instrument uncertainty of the new fast response thermowell RTD system.

Additionally, I&M has evaluated the potential effect of new instrument uncertainties on the RCS total flow rate analysis. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits TS SRs 3.4.1.3 and 3.4.1.4 require periodic verification that RCS total flow rate is greater than or equal (≥) to 341,100 gallons per minute (gpm). RCS flow measurement uncertainty is dependent, in part, on the accuracy of hot and cold leg temperature measurement. I&M's engineering evaluation has determined that the existing TS RCS total flow rate limit of ≥ 341,100 gpm will bound the instrument uncertainty impact of the new fast response thermowell RTD system.

RTD Element Failure

As with the existing RTD bypass system, the failure of an RTD would be identified using existing control board alarms and indicators following installation of the new fast response thermowell RTD system. These alarms and indicators include Tavg deviation alarms, ΔT deviation alarms, Tavg–Tref deviation alarms, and shiftly rounds which verify all required Tavg and ΔT indications. If a deviation alarm for a channel is received, or if a channel check during operator rounds reveals a deviation in one or more channels, the condition is evaluated. If the condition is determined to be caused by a failed RTD, the following actions may be taken.

In the existing RTD bypass system, the hot and cold leg RTD manifolds each contain an active single element RTD and a spare single element RTD. The spare RTD can be connected in the event of a failure of an operating RTD. In the new fast response thermowell RTD system, each

RTD contains a second element that can be connected to the circuit in the event of an operating element failure. Switchover to the spare RTD element can be performed at the appropriate terminal blocks in a junction box in the reactor cable tunnel, which is located outside the containment and is accessible during reactor operation at power.

RCS Pressure Boundary Codes

The piping analysis for the RTD bypass removal modification uses the governing code for the CNP RCS piping, ANSI B31.1, 1967 Edition. The RCS is an Inservice Inspection Class 1 system governed by ASME Section XI, 1989 Edition, for repair and replacement of pressure retaining components and their supports. The new welds will be inspected in accordance with Section XI requirements. The RCS pressure boundary will be leak tested at normal operating temperature and pressure per ASME Section XI, Code Case N416-1.

5.0 REGULATORY SAFETY ANALYSIS

<u>No Significant Hazards Consideration</u>

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with the proposed change 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

The proposed change is deletion of a Technical Specification (TS) note which requires verification of the resistance temperature detector (RTD) bypass flow rate for the Reactor Coolant System (RCS) loop. Removal of the TS requirement to verify RTD bypass flow rate will not involve a significant increase in the probability of occurrence of a previously evaluated accident because the RTD bypass will no longer exist. The bypass loop piping and RTDs will be removed and a new system will be installed. The new system uses new fast response thermowell RTDs that extend directly into the RCS loop flow. Removal of the RTD bypass piping is expected to result in a reduction of approximately 30 person-rem in the radiation exposure to personnel performing work in containment during refueling outages.

Replacement of the existing RTD bypass system with the new system will also not involve a significant increase in the probability of occurrence of a previously evaluated accident. The new system will perform the same control and indication functions, and is expected to be as reliable as the existing system. The accuracy and response time of the new system with

respect to control and indication functions has been evaluated and determined to be acceptable. Accordingly, there is no significant increase in the probability of occurrence of an accident resulting from a control or indication system malfunction. The new system will result in a reduction in the number of RCS pressure boundary welds and valves, thereby reducing the number of potential leakage sources. The new RCS pressure boundaries will be installed, inspected, and tested in accordance with applicable codes and standards. Accordingly, there is no significant increase in the probability of an accident resulting from an RCS pressure boundary malfunction. Therefore, the probability of occurrence of an accident previously evaluated will not be significantly increased.

Consequences of an Accident Previously Evaluated

Removal of the TS requirement to verify RTD bypass flow will not involve a significant increase in the consequences of a previously evaluated accident because the RTD bypass will no longer exist. Replacement of the existing RTD bypass system with the new system will also not involve a significant increase in the consequences of a previously evaluated accident. The new system will perform the same Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS) functions, and is expected to be as reliable as the existing system. The accuracy and response time of the new system with respect to RPS and ESFAS functions has been evaluated and determined to be acceptable. Accordingly, the proposed change will not decrease the reliability or effectiveness of the RPS and ESFAS systems in mitigating accidents or events. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

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

   Response: No.

   Removal of the TS requirement to verify RTD bypass flow will not create the possibility of a new or different kind of accident because the RTD bypass will no longer exist. Replacement of the existing RTD bypass system with the new system will also not create the possibility of a new or different kind of accident. The previously evaluated potential accidents and events associated with the existing system are those initiated by control system malfunctions and those involving loss of RCS pressure boundary integrity. The changes to control systems are limited to changes to the temperature sensors and associated signal processing. There will be no change in the ability to detect an RTD element failure and the ability to correct such a failure will be maintained. The changes to the RCS pressure boundary are limited to removal of existing piping and components and installation of new components. These changes do not introduce any new failure modes in the control systems or the RCS pressure boundary, nor do they induce new failure modes in other structures, systems, or components. 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

   Removal of the TS requirement to verify RTD bypass flow will not involve a significant reduction in a margin of safety because the RTD bypass will no longer exist. Replacement of the existing RTD bypass system with the new system will also not involve a significant reduction in a margin of safety. The margins of safety applicable to the RCS temperature monitoring instrumentation changes are those associated with the instrument response times and instrument uncertainty. Calculations have demonstrated that the margins between the values resulting from the new system and the values assumed in accident analyses will not be significantly reduced. The margins of safety applicable to the RCS pressure boundary changes are those that assure pressure boundary integrity. Compliance with applicable codes and standards for the installation, inspection, and testing of the new RCS pressure boundaries will assure that there is no significant reduction in the associated margins of safety. 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 change 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.

Applicable Regulatory Requirements/Criteria

10 CFR 50.36 requires that each license authorizing operation of a production or utilization facility include TS. The TS are required to include Surveillance Requirements (SRs), which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that the facility operation will be within safety limits, and that the limiting conditions for operation will be met. This amendment deletes an SR that will become obsolete following the implementation of a plant modification to delete the RTD bypass manifold system. I&M has determined that no other TS are affected.

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 and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

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 the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined by 10 CFR 20, or would change an inspection or SR. 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 PRECEDENTS

Precedent amendments have been approved for at least 15 other nuclear power plants. Three of these plants that more recently received amendments solely addressing removal of the RTD bypass were the Byron and Braidwood Nuclear Power Stations (Reference 1) and the North Anna Power Station (Reference 2). The proposed CNP amendment differs from these precedent amendments in that CNP has converted its TS to the NUREG-1431 Improved Standard Technical Specifications and the CNP TS will not have to be changed to incorporate new allowable values, OTΔT or OPΔT time constants, or instrumentation response times.

8.0 REFERENCES

1. Letter from R. R. Assa, NRC, to D. L. Farrar, Commonwealth Edison Company, "Issuance of Amendments – Byron and Braidwood Stations (TAC Nos. M91667, M91668, M91669 and M91670)," dated September 5, 1995, ML020870191.

2. Letter from L. B. Engle, NRC, to W. L. Stewart, Virginia Electric and Power Company, "North Anna Units 1 and 2 - Issuance of Amendments Re: Elimination of Resistance Temperature Detectors and Substitution of Thermowells (TAC Nos. M82838 and M82839)," dated April 22, 1992, ML013480129.

Attachment 1 to AEP:NRC:6331-01

UNIT 1
TECHNICAL SPECIFICATION PAGE
MARKED TO SHOW PROPOSED CHANGE

3.3.1-9

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.11	-----NOTES----- 1. For Function 4, not required to be performed until 12 hours after THERMAL POWER is below the P-10 interlock. 2. For Function 5, not required to be performed until 4 hours after THERMAL POWER is below the P-6 interlock. ----- Perform COT.	184 days
SR 3.3.1.12	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.13	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.14	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.15	-----NOTE~~S~~----- ~~1. This Surveillance shall include verification of Reactor Coolant System resistance temperature detector bypass loop flow rate.~~ ~~2.~~ Normalization of the ΔT is not required to be performed until 72 hours after THERMAL POWER is ≥98% RTP. ----- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.16	Perform COT.	24 months
SR 3.3.1.17	Perform TADOT.	24 months

Attachment 2 to AEP:NRC:6331-01

UNIT 1
PROPOSED TECHNICAL SPECIFICATION PAGE

3.3.1-9

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.11	-------------------------NOTES------------------------- 1. For Function 4, not required to be performed until 12 hours after THERMAL POWER is below the P-10 interlock. 2. For Function 5, not required to be performed until 4 hours after THERMAL POWER is below the P-6 interlock. -- Perform COT.	184 days
SR 3.3.1.12	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.13	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.14	-------------------------NOTE------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.15	-------------------------NOTE------------------------- Normalization of the ΔT is not required to be performed until 72 hours after THERMAL POWER is ≥98% RTP. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.16	Perform COT.	24 months
SR 3.3.1.17	Perform TADOT.	24 months