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June 27, 2007

AEP:NRC:7331-03
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 and Unit 2
Docket Nos.: 50-315 and 50-316
Response to Request for Additional Information Regarding Proposed Amendment Involving Thermowell Mounted Temperature Detectors (TAC Nos. MD3462 and MD3462)

References:

1. Letter from M. A. Peifer, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Application for Amendment to Revise Unit 1 and Unit 2 Technical Specifications to Reflect Replacement of Existing Reactor Coolant System Resistance Temperature Detectors and Bypass Piping with Detectors Mounted in the Primary Loop Piping," AEP:NRC:6331-05, dated November 3, 2006, (ML63320468).

2. Letter from P. S. Tam, NRC, to M. K. Nazar, I&M, "Donald C. Cook Nuclear Plant (DCCNP), Units 1 and 2 - Request for Additional Information Regarding Proposed Amendment Involving Thermowell Mounted Temperature Detectors (TAC Nos. MD3462 and MD3462)," dated March 27, 2007 (ML070811166).

Dear Sir or Madam:

This letter provides Indiana Michigan Power Company's (I&M's) response to a Nuclear Regulatory Commission (NRC) request for additional information regarding an amendment request to revise the Donald C. Cook Nuclear Plant (CNP) Technical Specifications (TS) involving Reactor Coolant System (RCS) resistance temperature detectors (RTDs).

By Reference 1, I&M proposed to amend the CNP Unit 1 and Unit 2 TS to reflect a plant modification that replaces the RCS RTDs and bypass piping with fast response thermowell detectors mounted directly in the RCS loop piping. I&M estimates that removal of the RTD bypass piping system would save approximately 40 percent (30 person-rem) of the overall dose each subsequent refueling outage. The proposed amendment consisted of deletion of a Unit 2 TS note requiring verification of bypass piping flow rates, and changes to the Unit 1 and Unit 2 TS Allowable Values

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for overtemperature differential temperature (OTΔT) and overpower differential temperature (OPΔT) reactor trip system functions. By Reference 2, the NRC requested additional information regarding proposed TS changes.

Enclosure 1 to this letter provides an affirmation affidavit pertaining to the additional information. Enclosure 2 provides I&M's response to the NRC request for additional information. As described in Enclosure 2, I&M has elected to withdraw the proposed changes to the Unit 1 and Unit 2 TS OTΔT and OPΔT Allowable Values.

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 Ms. Susan D. Simpson, Regulatory Affairs Manager, at (269) 466-2428.

Sincerely,

Joseph N. Jensen
Site Vice President

JRW/rdw

Enclosures:

1. Affirmation.
2. Response to Request for Additional Information
3. Westinghouse Letter AEP-06-104

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 27th DAY OF June, 2007

Regan D. Wendzel
Notary Public

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

My Commission Expires ____________

Response to Request for Additional Information

References for information provided by Indiana Michigan Power Company (I&M) are identified on Page 10.

By Reference 1, I&M proposed to amend the Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2 Technical Specifications (TS) to reflect a plant modification that replaces the Reactor Coolant System (RCS) resistance temperature detectors (RTDs) and bypass piping with fast response thermowell detectors mounted directly in the RCS loop piping. I&M estimates that removal of the RTD bypass piping system would save approximately 40 percent (30 person-rem) of the overall dose each subsequent refueling outage. The proposed TS changes consisted of deletion of a Unit 2 TS note requiring verification of bypass piping flow rates, and new Unit 1 and Unit 2 TS Allowable Values for overtemperature differential temperature (OTΔT) and overpower differential temperature (OPΔT) reactor trip system functions. By Reference 2, the Nuclear Regulatory Commission (NRC) requested additional information regarding proposed TS changes. Each question presented in Reference 2 is restated below followed by I&M's response.

NRC Reactor Systems Branch Question A.1

The TS changes associated with the resistance temperature detector bypass line elimination have already been approved for DCCNP-1 (Amendment No. 296, dated October 6, 2006). As with DCCNP-1 (see licensee's letter dated May 31, 2006; Accession No. ML061600449), the response time for the OTΔT trip in DCCNP-2 will be maintained at 8 seconds or less.

The following events could lead to a reactor trip when the calculated OTΔT trip setpoint is reached:

1. *Loss of electrical load/turbine trip*
2. *Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power*
3. *Chemical Volume Control System (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant*
4. *Inadvertent opening of a pressurizer safety or relief valve*

The licensee has provided evaluations of the first three events for both Cook units (licensee's May 31 and November 3, 2006, letters). The fourth event, the inadvertent opening of a pressurizer safety or relief valve, is not in the licensing basis of either Cook unit. This event, like the uncontrolled RCCA bank withdrawal at power, could erode thermal margin (i.e., an OTΔT trip could occur as thermal margin is decreased by a reduction in reactor coolant system (RCS) pressure as well as by an increase in power generation). Both the uncontrolled RCCA bank withdrawal at power event, and the inadvertent opening of a pressurizer safety or relief valve event are important in the determining the constants and coefficients in the OTΔT trip setpoint equation and of the OTΔT trip's dynamic time response characteristics.

The OTΔT trip is designed to protect the plant against departure-from-nucleate-boiling (DNB) during uncontrolled RCCA bank withdrawal at power events that insert reactivity slowly. The high nuclear flux trip provides protection when reactivity is inserted more rapidly. The OTΔT trip, and the low pressurizer pressure trip, protect the plant against DNB during the inadvertent opening of a pressurizer safety or relief valve events. The effectiveness of the OTΔT trip is verified by showing that the reactor trip signal is generated in time to prevent DNB, without taking credit for a reactor trip from the low pressurizer pressure trip logic. This has not been done for either of the Cook units, since these units do not include the inadvertent opening of a pressurizer safety or relief valve event in their licensing bases.

Table 1 shows that the inadvertent opening of a pressurizer safety or relief valve event was specified in the Standard Format in October, 1972 (RG 1.70, Revision 1).

Table 1: Standard Format and Content of Safety Analysis Reports

Event analyses that are not in the Licensing Bases of DCCNP-1 and -2	*RG 1.70 R0 - 2/72*	*RG 1.70 R1 - 10/72*	*RG 1.70 R1 - 10/72*
Reactor coolant pump shaft break	*locked rotor*	*locked rotor*	*T15-1 (3.4)*
Single RCCA withdrawal	*T15-1 (3)*	*T15-1 (3)*	*T15-1 (4.3)*
Inadvertent loading and operation of a fuel assembly in an improper position	*T15-1 (18)*	*T15-1 (15)*	*T15-1 (4.7)*
Inadvertent actuation of the emergency core cooling system that increases RCS inventory		*T15-1 (32)*	*T15-1 (5.1)*
Inadvertent actuation of the CVCS that increases RCS inventory	*T15-1 (4)*	*T15-1 (4)*	*T15-1 (5.2)*
Inadvertent opening of a pressurizer PORV		*T15-1 (13)*	*T15-1 (6.1)*
Radiological consequences of failure of small lines carrying primary coolant outside containment	*T15-1 (26)*	*T15-1 (22)*	*T15-1 (6.2)*

Table 2 shows that DCCNP-1 and DCCNP-2 were licensed 2 and 5 years, respectively, after RG 1.70 incorporated the inadvertent opening of a pressurizer safety or relief valve event.

Table 2: Chronology

June, 1966	*A Guide for the Organization and Contents of Safety Analysis Reports*
July, 1967	*Proposed General Design Criteria*
February, 1971	*10 CFR 50, App A, General Design Criteria*
July, 1971	*10 CFR 50, App A, General Design Criteria*
February,1972	*RG 1.70, Rev 0, Standard Format and Content*
October, 1972	*RG 1.70, Rev 1, Standard Format and Content*
August, 1973	*ANSI N18.2-1973, Nuclear Safety Criteria for Design of PWRs*
October, 1974	*DCCNP-1 was licensed*
January, 1975	*DCCNP-1 achieved initial criticality*
September, 1975	*RG 1.70, Rev 2, Standard Format and Content*
November, 1975	*NUREG-75/087 SRP*
December, 1977	*DCCNP-2 was licensed*
March, 1978	*DCCNP-2 achieved initial criticality*
November, 1978	*RG 1.70, Rev 3, Standard Format and Content*
October, 1986	*DCCNP-2, Cycle 6 SAR*
August, 1989	*"Analysis of D.C. Cook Unit 2, Cycle 8 Reload"*

Table 2 also shows that DCCNP-1 and -2 were licensed more than a year after the issuance of ANSI N18.2-1973, "Nuclear Safety Criteria for Design of PWRs." This standard categorizes the analyzed events according to expected frequency of occurrence, and lists the inadvertent opening of a pressurizer safety or relief valve event as an example of a Condition II (an event of moderate frequency) event. One year after DCCNP-2 was licensed, another revision of RG 1.70 and the Standard Review Plan were issued. Both contained the inadvertent opening of a pressurizer safety or relief valve event. Nevertheless, the licensee continued to maintain that this event, *along with six others listed in Table 1, were not in the licensing bases of DCCNP-1 and -2. The NRC staff accepted this position as recently as 1989 (letter from J. G. Giitter, August 3, 1989). At that time, the licensee and Westinghouse asserted that the seven events were analyzed or evaluated by ANF [Advanced Nuclear Fuel] in response to NRC staff questions regarding the use of ANF methodology in the licensing of ANF-supplied fuel. They were not part of the licensing basis for Westinghouse-supplied fuel. As such, they were not to be considered as part of the licensing basis when the Cook fuel supply contracts reverted to Westinghouse.*

The seven events of Table 1 are not in the current licensing bases of DCCNP-1 and -2. Yet, when issuing an amendment, the NRC staff needs to be able to make the statement that there is "reasonable assurance that the activities authorized by [the] amendment can be conducted without endangering the health and safety of the public" (i.e., the absence of an issue in the current licensing basis is not a cause prohibiting the staff from reviewing that issue where safety may be affected by the proposed amendment). The fact that the current DCCNP-1 and -2 licensing bases do not include the aforementioned seven event evaluations or analyses should not prevent the NRC staff to question whether there is a significant reduction in a margin of safety related to one of these events. The subject amendment application would result in a change to the OTΔT trip. Accordingly, the staff requests an analysis, or equivalent, to provide reasonable assurance that the modified OTΔT trip will not significantly reduce a margin of safety (e.g., thermal margin) during an inadvertent opening of a pressurizer relief or safety valve, an event that could demand a reactor trip through the OTΔT trip logic.

I&M Response to Question A.1

To provide the requested reasonable assurance that the activities authorized by the proposed amendment can be conducted without endangering the health and safety of the public, I&M had Westinghouse perform a comparison of CNP parameters with those of a similar plant that performed the same plant modification to eliminate the RTD bypass. That plant's licensing basis included the inadvertent opening of a pressurizer safety or relief valve. The Westinghouse comparison is provided in Enclosure 3 to this letter. As described in Enclosure 3, Westinghouse determined that the effect of the modification on departure from nucleate boiling DNB margin for the event would be small relative to the available margin. The Westinghouse comparison determined that similar results could be expected if the analysis was performed for CNP.

Regarding the CNP licensing basis, I&M acknowledges that Revision 0 and Revision 1 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," were issued prior to the licensing of CNP Unit 1 and Unit 2. However, compliance with Regulatory Guides is not mandatory unless a commitment to comply with the Regulatory Guide is documented in a plant's licensing basis. The original CNP licensing basis and the current CNP licensing basis do not include a commitment to follow the accident analysis format and content of Regulatory Guide 1.70. Although analyses of seven potential events identified in Regulatory Guide 1.70 (including the inadvertent opening of a pressurizer safety or relief valve), were performed in 1984 as part of a transition from

Westinghouse fuel to ANF, these analyses were not reperformed when CNP subsequently transitioned back to Westinghouse fuel. In a 1989 meeting, I&M informed the NRC that these analyses were not part of the original CNP licensing basis and that they would not be included in the CNP licensing basis following the transition back to Westinghouse fuel. As noted in Question 1 above, the NRC accepted this position and documented their acceptance in Reference 3. CNP has continued to use Westinghouse fuel and, accordingly, its current licensing does not include an analysis of the inadvertent opening of a pressurizer safety or relief valve. I&M, does not consider the discussion in the preceding paragraph of how reasonable assurance of public health and safety is provided modifies the CNP licensing basis to include a plant specific analysis of the inadvertent opening of a pressurizer safety or relief valve.

NRC Reactor Systems Branch Question A.2

A.2 [Original draft question deleted per telephone discussion of March 15, 2007.]

NRC Reactor Systems Branch Question A.3

Please verify that despite the proposed changes to the allowable values for the OPΔT and the OTΔT trip set points, the UFSAR analysis limits will be maintained.

I&M Response to Question A.3

I&M has elected to withdraw the proposed changes to the Unit 1 and Unit 2 OTΔT and OPΔT Allowable Values. The proposed changes that are withdrawn are those documented in Reference 1, Attachments 1A through 2B, Table 3.3.1-1, Note 1 and Note 2, on Pages 3.3.1-15 and 3.3.1-16 of both the Unit 1 and Unit 2 TS. This withdrawal eliminates all TS changes proposed for Unit 1 by Reference 1 and renders the TS changes proposed for Unit 2 identical to the TS changes approved by the NRC for Unit 1 via Reference 4. The description of the "Instrument Uncertainty Considerations" for OTΔT and OPΔT on Page 8 of Enclosure 2 to Reference 1 is hereby superseded by the following description, which is similar to that provided for Unit 1 in support of the amendment approved by the NRC in Reference 4.

OTΔT and OPΔT Instrument Uncertainty Considerations

Instrument uncertainty calculations have been performed for the new fast response thermowell RTD system in Unit 2. 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. 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.

All other portions of the "Technical Analysis" section of Reference 1 remain applicable. The scope of the "Regulatory Safety Analysis" section of Reference 1, including the "No Significant Hazards Consideration" description, has not been expanded. The "Background," "Environment Considerations," "Precedents," and "References" sections of Reference 1 remain valid.

NRC Reactor Systems Branch Question A.4

The calculations performed by the NRC staff show that the changes in the allowable values are within a fraction of a degree Fahrenheit. Show that the new thermowell resistance temperature detectors (RTDs) have the capability to measure this difference in the allowable value.

I&M Response to Question A.4

As described in the response to NRC Question A.3, I&M has elected to withdraw the proposed changes to the Unit 1 and Unit 2 TS OTΔT and OPΔT Allowable Values.

NRC Instrumentation and Controls Branch Question B.1

Enclosure 2, Section 4.0, discusses in general, the instrument uncertainty considerations for the calculations of the allowable value for OTΔT and OPΔT and Enclosure 3 provides the generic D. C. Cook Nuclear Plant setpoint methodology found acceptable by the NRC. Please provide the detailed calculations, including all actual values used for uncertainties, that show justification for the increase in allowable values for OTΔT and OPΔT. Also, please provide the source and/or justification for each uncertainty value used in the calculation.

I&M Response to Question B.1

As described in the response to NRC Question A.3, I&M has elected to withdraw the proposed changes to the Unit 1 and Unit 2 TS OTΔT and OPΔT Allowable Values.

NRC Instrumentation and Controls Branch Question B.2

Enclosure 2, Section 3.0, states that the three hot-leg scoops in each reactor coolant system (RCS) loop will be modified to accept the new thermowells, which will contain the new, fast-response RTDs and that a hole will be drilled through the end of each scoop to facilitate flow past the RTD. How large is the drilled exit hole in comparison to the scoop's water-entry cross-section size? How was it determined that this exit hole size was sufficient to not cause reduced flow through the scoop that could potentially add to a delay in the response time of the measurement of RCS temperature changes or even introduce another uncertainty in the measurement?

I&M Response to Question B.2

The nominal dimensions of the exit hole, the holes that comprise the scoop's water-entry cross-section, the scoop inside diameter, and key thermowell diameters are shown on the attached sketch. These dimensions are standard for Westinghouse four loop plants that have performed the RTD bypass removal plant modification.

To determine the appropriate dimension for the exit hole, Westinghouse performed a test with an RTD in a scoop located in a piping loop with a pump capable of producing a water velocity of 50 feet per second approaching the scoop. A velocity of 50 feet per second is the nominal hot leg reactor coolant velocity. Since the actual hot leg velocity is greater than 50 feet per second, the actual response time is less. A bypass system was arranged to produce a rapid temperature change in the water flowing past the scoop. The temperature transient measured by the RTD was compared with the temperature transient measured by a detector just upstream of the scoop to define the response time difference for the RTD inside the scoop. Several scoop hole sizes were tested, and 0.625 inch diameter was selected as providing the optimal flow characteristics. The test results indicated that the response time of the new system with a thermowell RTD located inside a scoop was less than the response time allowance considered in the safety analysis for the new system. Response time tests at other plants performed after installing the new system confirmed that the actual response time was within the safety analysis allowance.

NRC Instrumentation and Controls Branch Question B.3

[Original draft question deleted per telephone discussion of March 15, 2007.]

NRC Instrumentation and Controls Branch Question B.4

Enclosure 2, Section 3.0, describes in general, the arrangement whereby the three RTDs in an RCS loop will be electronically averaged to obtain a single hot-leg RCS temperature for that loop. Please describe the averaging function. Can a failure of one of the three RTDs in an RCS loop be automatically identified and taken out of the averaging equation by the new electronic averaging circuit?

I&M Response to Question B.4

The average hot leg temperature in each loop is obtained by converting the resistance of each of the three RTDs in the loop to a voltage that represents temperature. The three voltages are added together and divided by three to obtain the average.

Similar to the RTD bypass removal modification performed at the Byron and Braidwood plants and approved by the NRC (Reference 5), the RTD bypass removal modification performed at CNP does not include automatic RTD failure detection. As described in Reference 1, an RTD failure would be identified by operators using existing control board alarms and indicators following installation of the new fast response thermowell RTD system. These alarms and indicators include average reactor coolant loop temperature (T_{avg}) deviation alarms, differential temperature (ΔT) deviation alarms, T_{avg} - $T_{reference}$ deviation alarms, and the TS required shiftly channel checks of T_{avg} and ΔT indications.

NRC Instrumentation and Controls Branch Question B.5

[This question was not discussed in the March 15, 2007, phone call.] Enclosure 3 states that the NRC concluded that the DCCNP allowable value calculation methodology is acceptable in a letter dated June 1, 2005 (Reference 6). However, based on the staff concerns identified in the NRC Regulatory Issue Summary (RIS) 2006-17 (Reference 4), please provide a statement confirming that the setpoints for OTΔT and OPΔT are Limiting Safety System Settings for the variables on which a Safety Limit (SL) has been placed.

I&M Response to Question B.5

As described in the response to NRC Question A.3, I&M has elected to withdraw the proposed changes to the Unit 1 and Unit 2 TS OTΔT and OPΔT Allowable Values.

NRC Instrumentation and Controls Branch Question B.6

[This question was not discussed in the March 15, 2007, phone call.] The NRC letter to the Nuclear Energy Institute, Setpoint Methods Task Force, dated September 7, 2005 (Reference 1), describes setpoint-related technical specifications (SRTS) that are acceptable to the NRC for instrument settings associated with SL-related setpoints. Specifically, Part AA" of the Enclosure to the letter provides limiting condition of operation notes to be added to the TS, and Part AB" includes a check list of the information to be provided in the TS Bases related to the proposed TS changes.

a. *Describe whether and how you plan to implement the SRTS suggested in the September 7, 2005, letter. If you do not plan to adopt the suggested SRTS, then explain how you will ensure compliance with 10 CFR 50.36 by addressing items b and c, below.*

b. *As-Found Setpoint Evaluation: Describe how surveillance test results and associated TS limits are used to establish operability of the safety system. Show that this evaluation is consistent with the assumptions and results of the setpoint calculation methodology. Discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be Ainoperable" or Aoperable but degraded." If the criteria for determining operability of the instrument being tested are located in a document other than the TS (e.g. plant test procedure), explain how the requirements of 10 CFR 50.36 are met.*

c. *As-Left Setpoint Control: Describe the controls employed to ensure that the instrument setpoint is, upon completion of surveillance testing, consistent with the assumptions of the associated analyses. If the controls are located in a document other than the TS (e.g. plant test procedure), explain how the requirements of 10 CFR 50.36 are met.*

References [for NRC Questions B.5 and B.6]

1. *Letter from P. L. Hiland, NRC, to NEI Setpoint Methods Task Force, "Technical Specification for Addressing Issues Related to Setpoint Allowable Values," dated September 7, 2005 (Accession No. ML052500004).*

2. *Letter from B. A. Boger, NRC, to A. Marion, "Instrumentation, Systems, and Automatic Society (ISA) S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated August 23, 2005 (Accession No. ML051660447).*

3. *Letter from J. A. Lyons, NRC, to A. Marion, NEI, "Instrumentation, Systems, and Automation Society S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated March 31, 2005 (Accession No. ML050870008).*

4. *NRC Regulatory Issue Summary 2006-17, ANRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specification,' Regarding Limiting Safety System Setting During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 (Accession No. ML051810077).*

5. *Technical Specification Task Force (TSTF) recommendation for Standard Technical Specification (STS) changes, TSTF-493, AClarify Application of Setpoint Methodology for LSSS Functions," Revision 0, January 27, 2006 (Accession No. ML060270503).*

6. *Letter from J. Donohew, NRC, to M. Nazar, I&M, dated June 1, 2005, Paragraphs G.1.2.a, G.1.2.b, and G.3.2 of the Safety Evaluation for the conversion of the CNP TS to Improved Technical Specifications (Accession No. ML050620034).*

I&M Response to Question B.6

As described in the response to NRC Question A.3, I&M has elected to withdraw the proposed changes to the Unit 1 and Unit 2 TS OTΔT and OPΔT Allowable Values.

References for Information Provided by I&M

1. Letter from M. A. Peifer, I&M, to NRC Document Control Desk, "Application for Amendment to Revise Unit 1 and Unit 2 Technical Specifications to Reflect Replacement of Existing Reactor Coolant System Resistance Temperature Detectors and Bypass Piping with Detectors Mounted in the Primary Loop Piping," AEP:NRC:6331-05, dated November 3, 2006, (ML63320468).

2. Letter from P. S. Tam, NRC, to M. K. Nazar, I&M, "Donald C. Cook Nuclear Plant (DCCNP), Units 1 and 2 - Request for Additional Information Regarding Proposed Amendment Involving Thermowell Mounted Temperature Detectors (TAC Nos. MD3462 and MD3462)," dated March 27, 2007 (ML070811166).

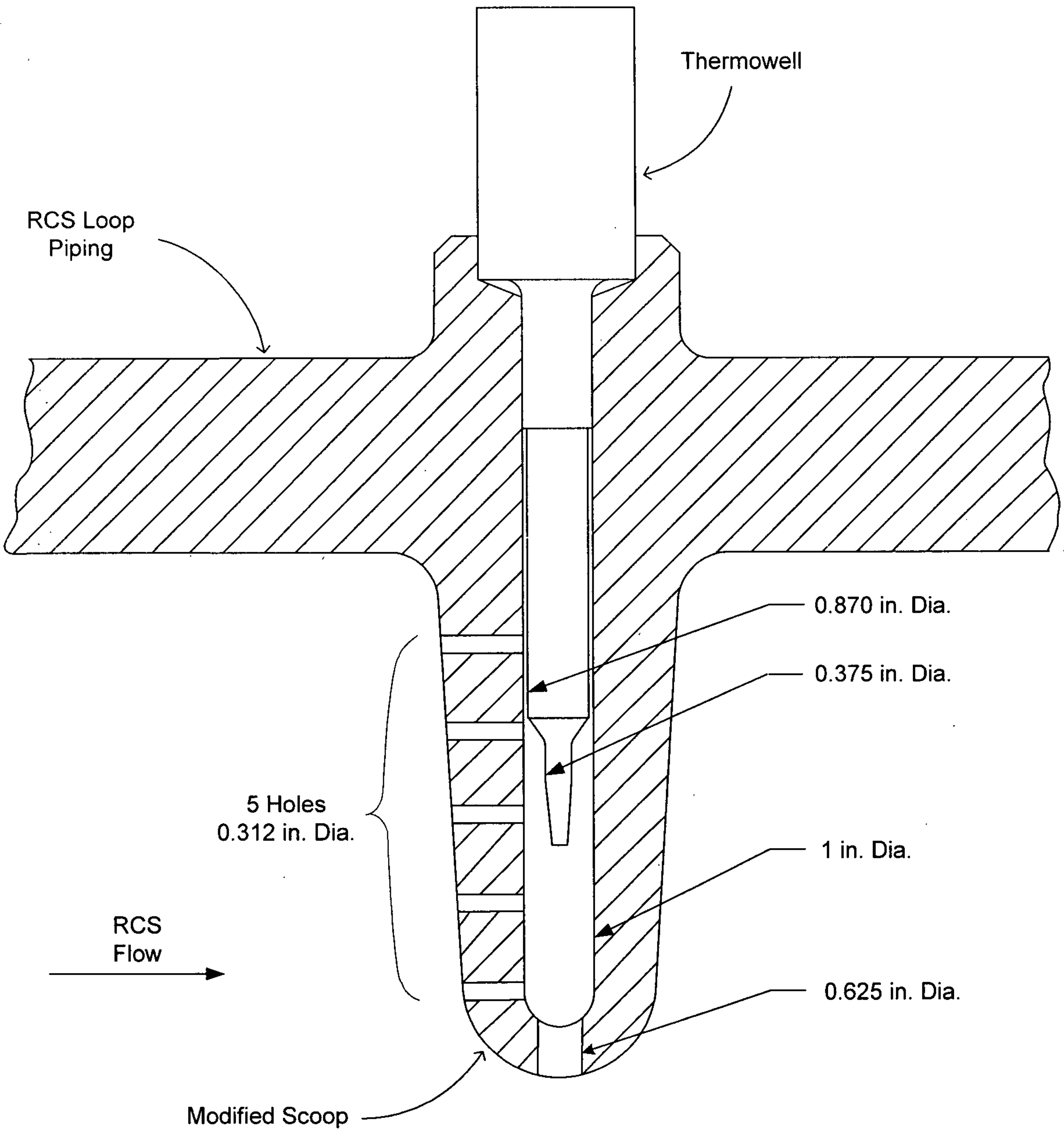
3. Letter from J. G. Giitter, NRC , to M. P. Alexich, I&M, "Analysis of D. C. Cook Unit 2, Cycle 8 Reload," dated August 3, 1989 (no ADAMS Accession number).

4. Letter from P. S. Tam, NRC, to M. K. Nazar, I&M, "Donald C. Cook Nuclear Plant, Unit 1 (DCCNP-1) - Issuance of Amendment Regarding Elimination of the Resistance Temperature Detector (RTD) Bypass Loop (TAC No. MD2106)," dated October 6, 2006 (ML062480328).

5. 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).

SKETCH OF THERMOWELL AND MODIFIED RCS HOT LEG RTD SCOOP

Dimensions are nominal



Enclosure 3 to AEP:NRC:7331-03

Westinghouse Letter AEP-06-104
Pages 1 through 3