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AEP
INDIANA MICHIGAN POWER

October 15, 2002

AEP:NRC:2900-01
10 CFR 50, Appendix J

Docket No. 50-315

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

Donald C. Cook Nuclear Plant Unit 1
RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT REQUEST FOR
APPENDIX K MEASUREMENT UNCERTAINTY RECAPTURE –
POWER UPRATE REQUEST (TAC NO. MB5498)

References: 1) Letter from J. E. Pollock (I&M) to Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request for Appendix K Measurement Uncertainty Recapture – Power Uprate Request," AEP:NRC:2900, dated June 28, 2002

2) Letter J. F. Stang (NRC) to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant, Unit 1 – Request for Additional Information Regarding License Amendment Request, 'Power Uprate Measurement Uncertainty Recapture,' dated June 28, 2002 (TAC NOS. MB5498)," dated October 2, 2002

By Reference 1, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, proposed to amend Facility Operating License DPR-58, including Appendix A, Technical Specifications, to allow a 1.66-percent increase in the licensed core power to 3304 MWt. By Reference 2, the NRC requested additional information regarding the changes proposed in Reference 1.

Attachment 1 to this letter addresses the specific questions transmitted in Reference 2 by either providing the requested information or committing to

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provide the information in the supplemental response. Attachment 2 provides a listing of commitments made in this letter.

Should you have any questions, or require additional information, please contact Mr. Brian A. McIntyre, Manager of Regulatory Affairs, at (269) 697-5806.

Sincerely,

J. E. Pollock
Site Vice President

NH/jen

Enclosures:
1. Notarized oath and affirmation statement

Attachments:
1. Response to NRC Request for Additional Information Regarding Unit 1 Measurement Uncertainty Recapture – Power Uprate Request
2. Regulatory Commitments

c: K. D. Curry – AEP Ft. Wayne
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MDEQ - DW & RPD
NRC Resident Inspector
J. F. Stang, Jr. – NRC Washington DC
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M. K. Scarpello, w/o attachments
T. R. Stephens, w/o attachments
M. G. Williams
T. K. Woods, w/o attachments

AFFIRMATION

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

Indiana Michigan Power Company

J E Pollock

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 15 DAY OF October, 2002

Jennifer L Kernosky
Notary Public

My Commission Expires 5/26/05

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

ATTACHMENT 1 TO AEP:NRC:2900-01

RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUEST FOR ADDITIONAL INFORMATION REGARDING
UNIT 1 MEASUREMENT UNCERTAINTY RECAPTURE - POWER UPRATE REQUEST

This attachment addresses the Nuclear Regulatory Commission (NRC) questions transmitted by Reference 1 and provides the information needed to support NRC review of the license amendment proposed by Reference 2. The information provided in this letter provides additional details of the analyses performed for the Unit 1 Measurement Uncertainty Recapture (MUR) Uprate Program, but does not impact the methodology, results or conclusions of these analyses, as presented in Reference 2.

I&M has determined that the No Significant Hazards Consideration (NSHC) determination provided in Enclosure 2 of Reference 2 is not impacted by the NRC Request for Additional Information (RAI) responses provided in this letter. With the exception of Questions 3, 14, 17, and 25, the responses provided in Attachment 1 expand on the level of detail provided in the license amendment request (Reference 2). In response to Questions 3 and 25, I&M commits to submit the measurement uncertainty calculation to the NRC, and in response to Question 14, I&M commits to submit a supplement to Reference 2, to amend two of the Technical Specification (TS) changes proposed in Reference 2. These commitments do not impact the justifications or conclusions reached by I&M in the NSHC determination. In response to Question 17, I&M agreed to a modification of the plant operating procedure changes that will be implemented as part of the activities associated with the design change that installs the Caldon Leading Edge Flow Meter (LEFM) system. This revised implementation activity has no impact on the uprated core power rating proposed by Reference 2. Therefore, no change to the NSHC determination provided in Enclosure 2 is necessary as a result of the information provided in this letter. Furthermore, I&M has determined that the environmental assessment provided in Enclosure 2 to Reference 2 is not affected by the information transmitted in this letter.

NRC Question 1

Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4 and NSAL 02-5, to document the problems with the Westinghouse designed steam generator water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for steam generator water level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, anticipated transient without scram (ATWS) and steamline break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation and affects the high-high level trip setpoint. NSAL 02-5, issued on February 19, *2002, deals with the initial conditions assumed in the steam generator water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how D. C. Cook Unit 1 accounts for these uncertainties documented in these advisory letters in determining the steam generator water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the loss-of-coolant accident (LOCA) and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, that the current analyses are still adequate with regard to the power uprate.*

Response to Question 1

The Westinghouse Electric Company (Westinghouse) Nuclear Safety Advisory Letters (NSALs) that identified steam generator level concerns (i.e., NSAL 02-3, including Revision 1, NSAL 02-4, and NSAL 02-5) have been entered into the Donald C. Cook Nuclear Plant (CNP) Plant Operating Experience database, and have been evaluated in accordance with the CNP Corrective Action Program. Each of these NSALs applies to plants with Westinghouse-designed steam generators. The CNP Unit 1 steam generators were replaced in 2000 with Babcock & Wilcox Model 51R steam generators. The internals of the Model 51R steam generators differ from the original Westinghouse Model 51 steam generators. Major differences include the separator unit, improved internal feedwater distribution system, use of lattice grid support plates, and improved tubing material. Furthermore, the Model 51R steam generators have major physical design differences in the narrow range level region. Since the CNP Unit 1 steam generators are not of Westinghouse design and differ significantly from a Westinghouse-designed steam generator, especially in the narrow range level region, the identified NSALs are not applicable to CNP Unit 1, and consequently, the concerns addressed in the subject NSALs have no effect on the CNP analyses-of-record.

NRC Question 2

Upon reviewing large-break LOCA models for power uprates, the Nuclear Regulatory Commission (NRC) recently found plants that require changes to their operating procedures because of inadequate hot leg switch-over times and boron precipitation modeling. Discuss how your analyses account for boric acid buildup during long-term core cooling and discuss how your predicted time to initiate hot leg injection corresponds to the times in your operating procedures.

Response to Question 2

Post-LOCA analysis issues pertaining to long-term core cooling, core subcriticality, and core boron precipitation control were resolved as part of the CNP restart effort that concluded in December 2000. Extensive analyses were performed to support containment system modifications described in Reference 3. The TS changes supported by the Reference 3 analyses were approved by Unit 1 License Amendment No. 234 (Reference 4). Section 3.5 of Attachment 10 to Reference 3 documents a detailed discussion of the comprehensive analyses performed to address post-LOCA concerns. In order to provide a reasonable amount of time for performance of the hot leg switchover evolution (i.e., between 2.5 hours and 7.5 hours after the initiation of the LOCA), it was necessary to credit the negative reactivity associated with control rod insertion. Additional analyses (Reference 5) demonstrating the acceptability of crediting control rod insertion were approved by the NRC in December 1999 (Reference 6). These analyses, which form the basis for the hot leg switchover time stipulated in the CNP Emergency Operating Procedures, are confirmed to remain valid and satisfied for each core reload design, and are not impacted by the 1.66 percent power uprate, as discussed in Section II.1.4 of Attachment 3 to Reference 2.

NRC Question 3

Regulatory Issues Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," Section I.1.E, indicates that a calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty should be submitted with the uprate application. Please provide your plant-specific calculations of the total power measurement uncertainty at the plant.

Response to Question 3

Calculation 1-2-O1-03 CALC 2, Revision 1 "Power Calorimetric Accuracy using the Caldon Check Plus Feedwater Flow Measurement System and a modified PPC CALM Program" will be provided under separate cover.

NRC Question 4

RIS 2002-03, Section II.1.5, "Steam Generator Tube Rupture (SGTR) – Thermal-Hydraulic Analysis," indicates that a change in power has negligible effect on the SGTR margin-to-overfill analysis. You further indicate that you performed a sensitivity study to show that an SGTR occurring with a power level increase of 2 percent remains bounded by your "supplemental SGTR analysis." However, the supplemental SGTR analysis was performed at a normal power level. Please describe the methodology of the sensitivity study to indicate why it remains bounded by the supplemental SGTR analysis performed at a lower power level.

Response to Question 4

Based upon Westinghouse's analytical experience with power uprate efforts, the change in power has a negligible effect on the Steam Generator Tube Rupture (SGTR) margin-to-overfill analysis.

A higher power tends to reduce the initial secondary pressure, thereby increasing the break flow rate. This has a small impact on the transient since, after reactor trip, the secondary pressure rises to the power-operated relief valve (PORV) setpoint. The higher power also tends to prolong the transient due to the higher decay heat level, requiring more time for the cooldown and final depressurization stages of the transient. The changes in primary and secondary conditions associated with the increased power also impact the secondary water inventory, tending to result in a lower initial inventory. For a small uprate, such as the proposed 1.66 percent power uprate, these impacts are not significant.

Westinghouse's generic power uprate experiences discussed above were confirmed specifically for the CNP Unit 1 MUR Uprate Program. The models and methods used in the evaluation of the impact of the 1.66 percent uprate on the CNP Unit 1 supplemental SGTR analyses are the same as those used for the current analysis supporting Unit 1 License Amendment No. 256 (Reference 7). Nominal and initial conditions were revised to model a 2 percent increase in nuclear steam supply system (NSSS) power. Values assumed for steam generator pressure and water mass correspond to the increased power condition using models and methods currently licensed for CNP Unit 1. The nominal NSSS power was increased by 2 percent from 3262 megawatts-thermal (MWt) to 3327 MWt. The increased power results in a lower secondary pressure and lower secondary water mass. The nominal steam pressure was reduced from 688 pounds per square inch absolute (psia) to 682 psia. The nominal (and initial) secondary water mass was reduced from 105,461 pounds-mass per steam generator (lbm/SG) to 104,826 lbm/SG. The nominal vessel average temperature was not changed. Reactor Coolant System (RCS) temperatures were calculated by the LOFTTR2 computer code, based upon the revised nominal and initial conditions. By Reference 7, the NRC approved the use of LOFTTR2 for the CNP Unit 1 supplemental overfill analysis.

The analysis modeling the 2 percent uprated power showed a negligible difference in the time the Overtemperature Delta-T reactor trip signal was generated. Operator action modeling was unchanged. Break flow termination was achieved at 3152 seconds, compared to 3144 seconds in the previous analysis. Integrated break flow increased from 160,424 lbm to 161,034 lbm. As noted above, the nominal and initial steam generator water mass is less under the uprated conditions compared to the exiting analysis. The net effect was a slightly less limiting transient relative to steam generator overfill. Specifically, the margin-to-overfill increased from 51 cubic feet (ft^3) to 65 ft^3. Thus, this sensitivity analysis confirmed for CNP Unit 1 that a 2 percent power increase has a negligible impact on the SGTR margin-to-overfill analysis. Therefore, the conclusion presented in Section II.1.5 of Reference 2 remains valid.

NRC Question 5

In RIS 2002-03, Section IV.5.2, "Structural Integrity Evaluation," you state that "Mechanical repair hardware was not evaluated for the D. C. Cook Nuclear Plant Unit 1 steam generators because they are new replacements with no installed repair hardware and minimal tube *plugging (less than 0.03 percent steam generator tube plugging (SGTP)." The NRC staff believes that the number of tube plugs currently installed in the steam generators is irrelevant and the licensee should evaluate the effect of the Measure Uncertainty Recapture (MUR) Power Uprate on the tube plugs.*

Response to Question 5

There are currently four tubes plugged among the four CNP Unit 1 Model 51R steam generators. Specifically, there are two plugs in SG 11, one plug in SG 13, and one plug in SG 14. The type of plugs used are Framatome Inconel 690 rolled plugs. The stress analysis and specifications for these plugs used parameters applicable to the original plant design and uprated power conditions (i.e., 3264 MWt and 3600 MWt NSSS thermal power conditions) that have been considered in the structural integrity evaluation, as discussed in Section IV.5.2 of Attachment 3 to Reference 2. Thus, the plugs in use in the CNP Unit 1 steam generators have been analyzed to conditions that bound the 1.66 percent MUR power uprate conditions. Therefore, the effect of the MUR uprate on the tube plugs has been addressed by the existing bounding analyses.

NRC Question 6

In RIS 2002-03, Section IV.5.2, "Structural Integrity Evaluation," you state that "Results of the analyses performed on the BWI Series 51 steam generators show that all steam generators components continue to meet American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1989 Edition, limits for the 1.66 percent uprate conditions with the reactor coolant system (RCS) pressure at 2100 psia. The primary-to-secondary pressure differential remains below the design value of 1600 psid. For operation with the RCS at 2250 psia, the primary-to-secondary pressure differential remains below the design value of 1600 psid, provided the secondary side steam pressure is limited to 679 psia." Based on the last sentence, the NRC staff believes you are stating the ASME limits will not be met under the uprate conditions when the RCS pressure is 2250 psia and the secondary side steam pressure is not limited to 679 psia. The staff's understanding, based on their review of Table 3, Case 2, is that it is possible that the secondary side steam pressure may be as low as 618 psia. Based on this conclusion, explain why the power uprate conditions are acceptable. If the intent is to control the secondary side steam pressure such that it is limited to 679 psia, describe the vehicle under which this will be performed.

Response to Question 6

The NRC staff's understanding that American Society of Mechanical Engineers (ASME) limits would not be met under the uprate conditions (i.e., RCS pressure at 2250 psia) if the secondary side steam pressure is not limited to 679 psia is correct. The CNP Unit 1 MUR Uprate Program analyzed conditions with the secondary side steam pressure as low as 618 psia. A subset of the activities I&M will perform as part of the design change to install and implement the Leading

Edge Flow Meter (LEFM™) CheckPlus™ system includes procedure revisions that address impacts. The 679 psia full-power steam pressure limitation for operation with reactor coolant pressure controlled to 2250 psia will be included in CNP Engineering Control Procedure ECP-1-O5-01, "Precautions, Limitations, and Setpoints – Unit 1." In addition, the 679 psia limitation will also be incorporated into the Updated Final Safety Analysis Report (UFSAR) as part of the CNP Unit 1 LEFM CheckPlus system design change package. Once it is incorporated into the UFSAR and the Precautions, Limitations, and Setpoints documents, future plant changes will be required to consider this limitation. Although these changes were not specifically addressed in Attachment 4, "Commitments," of Reference 2, identification and resolution of UFSAR and procedural impacts are required under I&M's design change process, as specified in the first commitment in Attachment 4. Therefore, no new regulatory commitments will be initiated to track these procedure/UFSAR changes.

NRC Question 7

In RIS 2002-03, Section IV.5.3, "Tube Vibration and Wear," you stated that "...the projected level of tube wear as a result of vibration would be expected to remain small, and will not result in unacceptable wear." Provide the staff with additional details (e.g., actual possible increase in wear as a result of power uprate conditions). In addition, describe the basis used to conclude that "unacceptable wear" would not occur.

Response to Question 7

The CNP Unit 1 steam generators were designed and analyzed to conditions that bound the 1.66 percent MUR power uprate conditions. Specifically, the steam generator tube vibration and wear evaluations were performed for individual steam generator power values of 816 MWt, 856 MWt, and 900 MWt (which correspond to total NSSS power levels of 3264 MWt, 3424 MWt, and 3600 MWt, respectively). Thus, the conditions assumed for the existing tube wear analyses bound the 1.66 percent power uprate conditions. Evaluations were also performed to estimate the potential for tube wear, as indicated by the determination of a fretting wear damage parameter, as defined in Reference 8. These evaluations show that the potential fretting wear would remain low. Specifically, the calculations for the 816 MWt individual steam generator power (i.e., 3264 MWt total NSSS power) indicate that the fretting wear damage parameter is 2.17E-3 kg-sec$^{-1.3}$, which is less than the Reference 8 limit of 4.0E-3 kg-sec$^{-1.3}$. Evaluations for the 1.66 percent power uprate condition indicate that an increased level of tube wear would result. The increase in the fretting wear damage parameter value was determined to result in a fretting wear damage parameter of approximately 3.3E-3 kg-sec$^{-1.3}$, which continues to remain below the previously established limit of 4.0 E-3 kg-sec$^{-1.3}$. From these evaluations, it was concluded that the increased level of wear that would occur at the uprated operating conditions would not be significant.

<u>NRC Question 8</u>

In RIS 2002-03, Section IV.5.4, "Regulatory Guide 1.121 Analysis," you state "The Regulatory Guide 1.121 analysis establishes minimum wall requirements for transient conditions corresponding to the 30 percent Steam Generator Tube Plugging (SGTP) case, which envelopes the primary-to-secondary pressure gradients for the 0 percent SGTP condition." In this analysis, is the assumed reactor coolant pressure 2250 psia (as seen in Table 3) or 2100 psia? State whether the assumed reactor coolant pressure bounds all possible pressures during operation (i.e., is the most bounding), and if not, analyze the bounding case and provide the results.

<u>Response to Question 8</u>

The Regulatory Guide (RG) 1.121 analysis performed for CNP Unit 1 in support of the MUR Uprate Program determined tube structural limits for a full range of "at power" operating conditions as shown in the following table. The extremes of the full-power average reactor coolant temperature (i.e., High T_{avg} and Low T_{avg}) were considered, as well as reactor coolant pressure values at both 2100 psia and 2250 psia. The analysis assumed a 30-percent steam generator tube plugging level, since this configuration envelopes the primary-to-secondary pressure gradients for the zero plugging condition.

Summary of Tube Structural Limits MUR Power Uprate Program Conditions

Condition	Parameter	High T_{avg} High Pressure	High T_{avg} Low Pressure	Low T_{avg} High Pressure	Low T_{avg} Low Pressure
Normal Operation	P_i (psia)	2250	2100	2250	2100
	P_o (psia)	765	765	690	618
	ΔP (psi)	1485	1335	1560	1482
	t_{min} (in)	0.022	0.020	0.023	0.022
Transient Conditions	P_i (psia)	2702	2787	2723	2824
	P_o (psia)	1133	1133	1133	1133
	ΔP (psi)	1569	1654	1590	1691
	t_{min} (in)	0.018	0.019	0.018	0.019
Faulted	P_i (psia)	2500	2500	2500	2500
	P_o (psia)	15	15	15	15
	ΔP (psi)	2485	2485	2485	2485
	t_{min} (in)	0.018	0.018	0.018	0.018
t_{min} (inches)		**0.022**	**0.020**	**0.023**	**0.022**
Structural Limit (%)		**55.1**	**59.2**	**53.1**	**55.1**

Note:

Structural Limit = $[(t_{nom} - t_{min}) / t_{nom}] \times 100\%$

$t_{nom} =$ 0.049 inch

Legend for "Summary of Tube Structural Limits MUR Power Uprate Program Conditions"

T_{avg} – vessel average temperature

P_i - steam generator tube inside pressure (primary side pressure)

P_o - steam generator tube outside pressure (secondary side pressure)

ΔP - $P_i - P_o$

t_{min} - minimum steam generator tube thickness

t_{nom} - nominal steam generator tube thickness

NRC Question 9

In RIS 2002-03, Section IV.5.3, "Tube Vibration and Wear," you describe the potential effects of the 1.66 percent MUR on steam generator tube wear. Discuss the potential effects of the 1.66 percent MUR on other potential modes of steam generator tube degradation (e.g., axial and/or circumferential cracking, etc.).

Response to Question 9

The RG 1.121 analysis establishes the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The allowable tube repair limit includes an allowance for degradation growth until the next scheduled inspection. The RG 1.121 analysis performed for the proposed 1.66 percent uprate of CNP Unit 1 considered parameter ranges that bound the 1.66 percent uprate conditions (as discussed further in the response to Question 8). Thus, the effects of the 1.66 percent uprate on steam generator tube degradation modes, such as axial or circumferential cracking, have been incorporated into the tube structural limits determined in accordance with RG 1.121.

CNP Unit 1 steam generators are designed and analyzed for a range of parameters and power levels that bound the conditions applicable to the proposed 1.66 percent uprate. Specifically, parameters representative of both original plant design and uprated power conditions (i.e., 3264 MWt and 3600 MWt NSSS power conditions) have been considered in the structural integrity evaluation, as discussed in Section IV.5.2 of Attachment 3 to Reference 2. Therefore, the proposed 1.66 percent power uprate of CNP Unit 1 remains bounded by the current structural design analyses and no new modes of steam generator tube degradation are introduced.

NRC Question 10

Discuss the impact the power uprate will have on the required frequency of steam generator tube inspections.

Response to Question 10

Steam generator tube inspections will be conducted at a frequency that is the more restrictive of either TS 3/4.4.5, "Reactor Coolant System, Steam Generators," or Electric Power Research Institute (EPRI) publication TR-107569-V1R5, "PWR Steam Generator Examination Guidelines," (Reference 9). I&M adopted EPRI publication TR-107569-V1R5 in accordance with the implementation guidance of NEI 97-06, "Steam Generator Program Guideline," (Reference 10). An increase in inspection frequency in terms of the number of steam generators inspected and the sample size of tubes inspected depends on the progression (if any) of degradation. None of the potential degradation mechanisms are significantly affected by the 1.66

percent power uprate conditions; therefore, the required frequency of inspection is also not affected significantly by the proposed Unit 1 MUR power uprate.

NRC Question 11

RIS 2002-03, Section VII.6.4 discusses the Flow-Accelerated Corrosion (FAC) Program. The NRC staff has the following questions related to the FAC Program:

- *Briefly describe the purpose and elements of the FAC Program.*

- *The submittal states that "Flowrates and temperatures for piping components within the scope of the FAC Program remain within the system design specifications." Explain how the system design specifications are used in the FAC Program (e.g., what decisions/assessments are made based on design flowrates and temperatures).*

- *Identify the software utilized as part of the FAC Program to model the piping systems. Identify the source (e.g., design values, actual values, etc.) of the input parameters (e.g., operating pressures, flowrates and temperatures) to the software program.*

- *Please discuss whether any additional systems will need to be added to the FAC Program as a result of the power uprate. Also discuss whether the power uprate will result in any changes to the software input parameters. If software input parameters will be affected, summarize the significance of the overall impact on FAC Program activities.*

Response to Question 11

- Flow-Accelerated Corrosion (FAC) Program – Purpose and Scope

  The purpose of the FAC Monitoring Program is to predict, detect, monitor, and mitigate FAC in plant piping. The scope of the program consists of all piping and components that cannot be demonstrated to be non-susceptible to FAC as documented in the current FAC System Susceptibility Evaluation.

  The main elements of the program are as follows:

  Scope Identification

  Based on an evaluation of plant piping systems, piping and components that are susceptible to FAC are identified and included within the scope of the FAC Program.

Evaluation and Modeling

Once identified, each susceptible line of plant piping is evaluated and addressed to ensure that the probable level of degradation is ascertained and appropriate action is taken. This is normally accomplished by modeling the line using the EPRI software program, CHECWORKS™. Some FAC-susceptible lines and components cannot be modeled in CHECWORKS due to uncertain operating conditions, software limitations, or other factors. These lines are addressed in the "susceptible non-modeled" (SNM) sub-program.

Inspection

Components are selected for inspection, based on CHECWORKS predictions, SNM selection criteria, past inspection results, and operating experience. Selected large-bore components (nominal diameter greater than 2 inches) are inspected during refueling outages using ultrasonic testing. Small-bore components (nominal diameter 2 inches or less) are typically inspected on-line using radiography. While radiography is preferred, ultrasonic testing is also an acceptable methodology for these components.

Data Analysis

The inspection results for CHECWORKS-modeled components are entered into the CHECWORKS model to calibrate the model. In addition, the component's remaining life is calculated by comparing the inspection results to the acceptance criteria. The results of this analysis are compared to the procedural acceptance criteria to determine:

- if a follow-up inspection is required and when it should be scheduled,
- if inspections at additional locations are needed in the current outage, and
- if the component requires repair or replacement – either immediately or during a future outage.

- The operating pressure, temperature, and flowrate are inputs to the CHECWORKS model, which is used to predict the FAC wear rate in the susceptible components. The MUR power uprate will result in changes to these operational values. The values will be revised in the CHECWORKS model to predict future wear rates. These changes will be minimal, and the system operating parameters will remain within the design limits.

- The EPRI CHECWORKS software is used by CNP's FAC Program to model the piping systems. The source of the input parameters is actual cycle-specific values, and the FAC Program directs these values to be changed and maintained in the software if the values change in the field.

- A review of heat balances that reflect the effect of the MUR power uprate on FAC-related parameters indicates that no additional systems, piping, or other components need to be added to the FAC Program as a result of the MUR power uprate. The FAC Program will direct changes to the CHECWORKS input parameters for changes to system flowrates, temperatures, and pressures. A revised plant heat balance calculation indicates that the changes are minimal and the MUR uprate is not expected to significantly affect the current wear predictions of the CHECWORKS software.

NRC Question 12

Provide the details of how the power uprate will effect the steam dump system capabilities.

Response to Question 12

Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam load at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant rated electrical load without a reactor trip. The current design requirement stated in the UFSAR is for the steam dumps, or turbine by-pass system, to have a capacity of approximately 40 percent of full-load steam flow. The 1.66 percent power uprate affects the steam dump capability in several ways. First, the full-load steam flow value increases with the uprate, so for a fixed steam flow through the steam dump valves, the capacity in terms of full-load steam flow is reduced. Secondly, the full-power steam pressure is reduced for increased steam flow conditions, given other parameters remain constant (e.g., T_{avg} and RCS flow rate). The net effect of the 1.66 percent power uprate is a slight reduction in the available steam dump capability, for a given set of RCS parameters.

As indicated in Section VI.2.1 of Reference 2, a final steam dump valve flow capacity analysis is in progress to determine the appropriate steam dump travel stop position. Based upon a Westinghouse evaluation, a capability of approximately 45 percent of full-load steam flow can be achieved with the travel stops removed, even if a conservatively low steam pressure of 618 psia is assumed. The uprated steam pressure will remain above 679 psia, as discussed in Section IV.5.2 of Reference 2, and further clarified in I&M's response to Question 6, above. To satisfy the commitment made by I&M in Reference 2, the steam dump travel stop position will be adjusted to the proper position prior to implementing the 1.66 percent power uprate.

NRC Question 13

Provide the details of how the power uprate will affect air and hydraulic operated valves.

Response to Question 13

A review of heat balances that reflect the effect of the 1.66 percent power uprate on system design function parameters indicates that there is no impact on the ability of air and hydraulic-operated valves (AHOVs) to perform their design function in the systems affected by the MUR power uprate. No additional AHOVs were identified as being impacted by the MUR Uprate Program. As a result, no changes will be required to the AHOV Program due to the MUR power uprate.

NRC Question 14

In the application, the pressure temperature curves on Figure 3.4-2 of the technical specifications (TS) have been changed to reflect the effects from the power uprate. By letter dated May 3, 2002, you stated the pressure temperature curves in the TS did not reflect the most limiting material. Please provide calculations and revised TS pages that reflect both the power uprate and the most limiting material.

Response to Question 14

In an August 13, 2002, telephone conference with the NRC Staff, I&M committed to revise the pressure-temperature (P-T) limit curves applicability limits proposed by Reference 2 to include the effects of the new limiting reactor vessel beltline material, as well as the increased neutron fluence associated with the 1.66 percent power uprate. A supplement to Reference 2 that will revise the proposed P-T curves applicability limits will be provided under separate cover. The revised applicability limits in the license amendment request supplement will reflect the results of a recent computation based on the uprated power level and considering the new limiting reactor vessel beltline material. The license amendment request supplement will also withdraw the proposed change to the reactor vessel surveillance schedule, as implementation of this change is not required for the MUR power uprate.

NRC Question 15

In response to item I.1.D of RIS 2002-03, Attachment 1, Submittal Attachment 3, cites WCAP-8567 for a description of "Improved Thermal Design Procedure" (ITDP) and states that NRC has approved the use of ITDP at Cook Nuclear Plant Unit 1. The reference cited for this approval does not include NRC review and acceptance of this procedure for general use. Please clarify. Please describe the application of ITDP for the requested power uprate.

Response to Question 15

The use of Westinghouse WCAP-8567, "Improved Thermal Design Procedure," was initially approved for CNP Unit 1 in the NRC's Safety Evaluation Report (SER) for Unit 1 License

Amendment 74, dated September 20, 1983 (Reference 11). Reference 11 should be used in place of Unit 1 License Amendment 126 (Reference I.6 of the Unit 1 MUR Uprate Program License Amendment Request, Reference 2), which approved a more recent use of the ITDP methodology at CNP Unit 1.

I&M applied the ITDP methodology for the requested 1.66 percent power uprate by identifying the individual contributors to the accuracy of the thermal power calorimetric. The error components (sensitivities) are combined statistically using the square root of the sum of the squares (SRSS) methodology to calculate the total reactor thermal power accuracy. Typically, error components that are dependent are combined arithmetically into independent groups, which are then statistically combined.

<u>NRC Question 16</u>

The "Sensitivity % Rated Thermal Power (RTP)" column of the submitted table appears to be mislabeled, and appears to represent the product of RTP sensitivity and the uncertainty in the various parameter measurements. The reference in that same table to "Root Mean Squared (RMS)" is interpreted to have been intended to mean "Square Root Sum of the Squares (SRSS)."

<u>Response to Question 16</u>

The "Uncertainty" and the "Sensitivity" columns of Table I-1 are directly related. The "%RTP" is intended to indicate that the units of the "Sensitivity" column are in percent of rated thermal power and not in the process units associated with the "Parameter" column. Total uncertainty was determined by varying each process parameter about a base value and determining the corresponding sensitivity of percent rated thermal power (%RTP). As an example, at the current 100 percent RTP, the feedwater pressure is 725.10 pounds per square inch – gauge (psig). The uncertainty of the feedwater pressure input to the plant process computer (PPC) calorimetric program is determined to be ±15 psig. Thus, if the actual feedwater pressure is 725.10 psig, and an error as large as ±15 psig is possible at the point that the feedwater pressure signal is input to the PPC, the corresponding error in the calculated %RTP could be as large as ±0.001753 %RTP. Thus, Table I-1 shows that the sensitivity of %RTP to a change in feedwater pressure of ±15 psig is ±0.001753 %RTP.

As noted, the reference in this same table to root mean squared (RMS) should have been referenced to SRSS.

<u>NRC Question 17</u>

In your Submittal Attachment, 3 Section I.1.G/H, bullet 5 you state that failure of one plane of leading edge flow meter (LEFM) transducers would not affect power measurement, and cites Caldon Topical Report ER-157P and the associated Safety Evaluation Report (SER) as *justification for this claim. The report and the associated SER do not support this claim. Loss of an entire detection plane in an LEFM CheckPlus system would render it functionally similar to an LEFM Check system. Such a reduced system would not be optimized for single-plane use, and so performance would likely fall short of a properly configured LEFM Check system. ER-157P clearly indicates a significant difference in accuracy between the Check and CheckPlus flowmeters. The topical report and SER indicate that continued operation without reduction in power with one LEFM CheckPlus* <u>*component*</u> *out of service* <u>*might*</u> *be justifiable, but leaves it up to the applicant to provide the justification. Please clarify.*

<u>Response to Question 17</u>

Reference 2, Attachment 3, Section I.1.G/H, Bullet 5, is deleted from this license amendment request by the following discussion. Implementation of the MUR Uprate Program will develop administrative controls that will ensure that loss of a single plane of transducers in the LEFM CheckPlus system is considered an LEFM out-of-service condition. This will eliminate the concern pertaining to the accuracy of the LEFM with the loss of a single plane of transducers. Attachment 4, "Commitments," of Reference 2 included a commitment to install the new LEFM CheckPlus system at CNP Unit 1. As noted in the commitment, the design change for the installation includes development of system out-of-service administrative requirements. Because the development of out-of-service administrative requirements is already being tracked by this commitment, no new regulatory commitments will be initiated to track this activity.

<u>NRC Question 18</u>

The application refers to a serial link between the LEFM and the PPC (Plant Process Computer) and states that the venturi-based instrument will always be calibrated in accordance with the last "good" value from the LEFM. However, there is no discussion of the timing or operation of this link or of the calibration adjustment. Please provide a discussion of: a) the nature and operation of the serial link, b) the schedule by which the venturi-based flowmeter calibration is adjusted, c) the method for adjusting the venturi flowmeter calibration, and d) the means for distinguishing "good" from "bad" LEFM data for the purpose of calibration adjustment of the venturi meter.

<u>Response to Question 18</u>

a. <u>Nature and Operation of the Serial Link</u>

The serial data link consists of the hardware and software used by the PPC to acquire data and status information from the Caldon LEFM CheckPlus system output and to store this information within the PPC for use by other applications.

A serial interface device is used to interface the PPC with the LEFM equipment via serial links. The device is connected to the ethernet local area network (LAN). A separate port on the serial device is connected to each LEFM link via an RS-232 serial cable. Since there are two links from the LEFM, Port 1 of the serial device is connected to CPU-A of the LEFM, while Port 2 is connected to CPU-B of the LEFM. The Data Link software uses specific operating system service calls to access and read the ports of the serial device. The data link does not provide any automatic update between the venturi and LEFM flowmeters. The process for adjustment between the venturi and LEFM power outputs is manual and is described below.

The venturi calorimetric and the LEFM calorimetric are completely separate and are performed independently by the PPC. Each program performs independent calculations to determine core thermal power.

b & c. Venturi Flowmeter Calibration Following LEFM Loss

The venturi instrumentation is not "calibrated" on-line by a linkage to the LEFM. Instead, the venturi calorimetric calculated power is manually adjusted; i.e., calibrated to the last "good" LEFM calorimetric and the corresponding venturi calorimetric at that same time. This is performed by retrieving the last good thermal power computed by the LEFM (P_L) and comparing it to the thermal power computed by the venturis (P_V) at that same time. A correction factor (CF) is then calculated by taking the ratio of the last good LEFM calorimetric power value to the venturi power value at that same time (i.e., $CF = P_L / P_V$). For the 48-hour period proposed in Reference 2, during which operation at the uprated power would be allowed as long as steady-state conditions exist, the corrected calorimetric power would be computed as being equal to the current venturi calorimetric power multiplied by the correction factor ($P_{CORR} = CF \times P_V$).

Plant operating procedures will be revised to ensure that should the LEFM out-of-service condition not be corrected, Operations will reduce plant thermal power level such that the plant is operating at or below the pre-uprate power level limit of 3250 MWt at the time that the 48 hours has elapsed. Attachment 4, "Commitments," of Reference 2 included a commitment to install the new LEFM CheckPlus system at CNP Unit 1. As noted in the commitment, the design change for the installation includes development of operational procedures. Therefore, no new regulatory commitments are required to track this procedure change.

d. Distinguishing "Good" from "Bad" LEFM Data

The last "good" value of the LEFM power calorimetric will be the last retrievable data point with a status of "good." The method of identifying the status of LEFM data by the electronic unit and the alarms to the operator are described in the Caldon documentation located in the CNP vendor documentation program and the software design descriptions for the data link and the calorimetric program.

NRC Question 19

If the plant computer or the plant computer serial link from the LEFM is not operational, automatic power calculations will not be performed. Please show that these conditions, and any other conditions that might interfere with automatic operation, are properly accounted for in the procedures and in appropriate limitations associated with the proposed modification. In particular, show that, despite such conditions, the venturi-based flowmeter calibration will remain sufficiently correlated with the LEFM calibration to support continued operation above the pre-uprate power limit in the event of LEFM failure.

Response to Question 19

The adjustment of the power-level calculated by the venturi upon a loss of the LEFM is explained in the response to Question 18. A PPC failure would be treated as a loss of both the LEFM and the ability to obtain a corrected calorimetric power using the venturis. This would result in reducing plant power to the pre-uprated rated thermal power limit of 3250 MWt. The 48-hour time period would not apply in this specific case, as a manual calorimetric would be required. The manual calorimetric only supports operation at plant power levels up to 3250 MWt.

NRC Question 20

Please show that the time limit established for continued operation above the pre-uprate power limit with the LEFM out of service properly accounts for:

a. *decay of venturi-based flowmeter accuracy from the most recent LEFM-based calibration update to the time of LEFM failure,*
b. *continued operation from the time of LEFM failure to the initiation of power runback, and*
c. *continued operation during power runback, until the indicated power is at or below the pre-uprate power limit.*

Response to Question 20

During past refueling outages, the feedwater venturis were inspected for evidence of fouling. The results of these venturi inspections have consistently indicated that the Unit 1 feedwater venturis do not experience fouling. Based on this evidence, feedwater venturi fouling that would result in degradation of the accuracy of these components is not expected. Thus, venturi fouling that would degrade flowmeter accuracy would not be expected over the 48-hour period that the LEFM is not operational.

As discussed in the response to Question 18, upon LEFM failure, the venturi power calorimetric values would be adjusted (i.e., calibrated to the last valid power output of the LEFM prior to the time of LEFM failure). Expectations of instrument drift vary depending upon the manufacturer's specifications. However, values of drift are typically in the range of tenths of a percent of the calibrated span over 18 to 24 months or more. This typical drift value would not result in any significant drift for the instrumentation associated with the calorimetric measurements over a 48-hour period.

In accordance with plant operating procedures, power reduction will occur to ensure that the plant will be at or below the pre-uprate power limit, 3250 MWt, within 48-hours in the event of a loss-of-LEFM condition. In the case of a loss of the PPC, the operating procedures will ensure that the plant transitions to the pre-uprate power level, as needed to support the manual calorimetric measurement using the venturis.

Therefore, operation over the period that the LEFM is out of service is justified using the venturis, as applicable.

NRC Question 21

Please specify the time allowed from initiation of power runback until core power reaches the pre-uprate power limit, in the event of extended LEFM failure.

Response to Question 21

For the LEFM out-of-service condition, the 48-hour "clock" will start at the time of LEFM failure. Failure will be annunciated in the control room on the PPC screen that displays the calorimetric power level. The LEFM electronic unit and central processing unit (CPU) continuously monitor, test and /or verify the following attributes of the LEFM operation:

- Acoustical processing units
- Analog inputs
- Test paths
- Signal quality
- Path-to-path sound velocity
- Velocity profiles
- Watchdog timer
- Flowrate calculations uncertainty verified against specified System Uncertainty Threshold
- Meter path operation (i.e., signal quality, sound velocity to specified thresholds)
- Meter velocity profile (i.e., changes in hydraulic profile, verified against specified thresholds)

The procedures for power reduction will be in accordance with current operating procedures such that the plant will be operating at, or below, the pre-uprate power level limit of 3250 MWt by the time the 48 hours has elapsed. For the loss-of-PPC case, the 48-hour limit does not apply and power reduction will be in accordance with current operating procedures such that the plant will transition to a power level at, or below, the pre-uprate power level limit of 3250 MWt, as described in response to Questions 18 and 19.

NRC Question 22

RIS 2002-03, Attachment 1, Section I.1.F requests certain information concerning all instrumentation involved in the power calorimetric. The licensee's response to this item addresses only the LEFM. Please provide the requested information for the remaining instrumentation.

Response to Question 22

In addition to the process inputs provided by the LEFM CheckPlus system, the PPC program uses the following process inputs to calculate thermal power:

- Steam Pressure
- Blowdown Flow
- Charging Flow
- Charging Temperature
- Charging Pressure
- Letdown Flow
- Letdown Temperature
- Letdown Pressure
- Pressurizer Pressure
- RCS Loop 4 Cold Leg Temperature
- Volume Control Tank (VCT) Temperature

Blowdown flow measurement is performed by a Caldon ultrasonic measurement system. Calibration of this ultrasonic measurement system is maintained using self-checking and self-adjusting methods. The value and status of the blowdown flow measurement are provided to the PPC. If the status of the blowdown flow measurement or the failure of the blowdown flow system indicates that the status is "bad", this is reflected in the PPC calorimetric program and results in the status of the LEFM calorimetric values also indicating a "bad" status. Control of the ultrasonic measurement system is maintained by the CNP change control process. Hardware control of the ultrasonic measurement system is provided by the CNP design change control process, which conforms to 10 CFR 50, Appendix B, and control of the system software is provided by CNP's software control process.

The remaining process inputs are obtained from analog instrumentation channels that are maintained and calibrated in accordance with required periodic calibration procedures.

Configuration of the hardware associated with these process inputs is maintained in accordance with the CNP change control process.

Instruments that affect the power calorimetric, including the LEFM inputs, are monitored by CNP's System Engineering personnel in accordance with the requirements of I&M's Corrective Action Program. Equipment problems for plant systems, including the LEFM CheckPlus equipment, fall under the site work control processes. Conditions that are adverse to quality are documented under the Corrective Action Program. Corrective Action procedures, which ensure compliance with the requirements of 10 CFR 50, Appendix B, include instructions for notification of deficiencies and error reporting.

Calibration and maintenance are performed by CNP Instrumentation and Controls (I&C) - Maintenance Department personnel using site procedures. Site procedures are developed using the vendor technical manuals for the applicable equipment. All work is performed in accordance with site work control procedures. Routine preventive maintenance procedures include physical inspections, power supply checks, back-up battery replacements, and internal oscillator frequency verification. Corrective actions involving maintenance will be performed by I&C-Maintenance personnel, qualified in accordance with I&M's I&C Training Program.

Plant procedures ensure compliance with the requirements of 10 CFR Part 21.

<u>NRC Question 23</u>

Please confirm that the installation of the new flowmeter will not adversely affect the performance of the existing flow instrumentation.

<u>Response to Question 23</u>

For Unit 1, the Caldon LEFM flow-measuring device is installed at least 124 feet upstream of the venturi and other feedwater flow instrumentation. The path between these instruments also includes at least six 90-degree elbows or tees, three flow valves, and three different pipe diameters. This is a sufficient equivalent length of pipe, in terms of the number of pipe diameters and flow resistance elements, to prevent hydraulic interference between these instruments. Therefore, there is no hydraulic communication between these instruments that would cause interference due to the installation of the new Caldon LEFM flow-measuring device.

<u>NRC Question 24</u>

Please clarify item I.1.G/H of Attachment 3 of your application to Reference 1 to address operation <u>at any power level in excess of</u> the <u>pre-uprate limit</u>, not just <u>exactly at</u> the <u>new</u> limit.

Response to Question 24

I&M concurs with the NRC staff's interpretation that Item I.1.G/H of Attachment 3 to Reference 2 should be interpreted to refer to plant operation at any power level in excess of the pre-uprate limit, up to the new limit.

NRC Question 25

Please provide a copy of the calculation that establishes the thermal power measurement uncertainty, as requested in Item I.1.E of Attachment 1, and reiterated and explained in Items I.2, I.6, and I.7 of Attachment 2, to RIS 2002-003. Item G.6 of RIS Attachment 2 also requests detailed information.

Response to Question 25

Calculation 1-2-O1-03 CALC 2, Revision 1 "Power Calorimetric Accuracy using the Caldon Check Plus Feedwater Flow Measurement System and a modified PPC CALM Program," which establishes the thermal power measurement uncertainty for the proposed 1.66 percent power uprate, will be provided under separate cover.

REFERENCES

1. Letter J. F. Stang (NRC) to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant, Unit 1 – Request for Additional Information Regarding License Amendment Request, 'Power Uprate Measurement Uncertainty Recapture,' dated June 28, 2002 (TAC No. MB5498)," dated October 2, 2002

2. Letter from J. E. Pollock (I&M) to NRC Document Control Desk, "License Amendment Request for Appendix K Measurement Uncertainty Recapture – Power Uprate Request," AEP:NRC:2900, dated June 28, 2002

3. Letter from R. P. Powers, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Technical Specification Change Request Containment Recirculation Sump Water Inventory," C1099-08, dated October 1, 1999

4. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6766 and MA6767)," dated December 13, 1999

5. Letter from R. P. Powers, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, License Amendment Request for Credit of Rod Cluster Control Assemblies for Cold Leg Large Break Loss-of-Coolant Accident Subcriticality," C0999-11, dated September 17, 1999

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

7. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MB0739 and MB 0740)," dated October 24, 2001

8. Pettigrew, M. J, Taylor, C. E. and Subash, N. "Flow Induced Vibration Specifications For Steam Generators and Liquid Heat Exchangers", AECL-11401, Chalk River Laboratories, Chalk River, Ontario, November 1995

9. EPRI TR-107569-V1R5, "EPRI PWR Steam Generator Examination Guidelines," September 1997

10. NEI 97-06, "Steam Generator Program Guideline"

11. Letter from D. L. Wigginton, NRC, to J. Dolan, I&M, issuing CNP Unit 1 License Amendment 74, dated September 20, 1983

ATTACHMENT 2 TO AEP:NRC:2900-01

REGULATORY COMMITMENTS

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

Commitment	Date
Calculation 1-2-O1-03 CALC 2, Revision 1 "Power Calorimetric Accuracy using the Caldon Check Plus Feedwater Flow Measurement System and a modified PPC CALM Program" will be provided under separate cover.	October 18, 2002
A supplement to Reference 2 will be provided under separate cover, proposing to revise the P-T curves applicability limits to 18.6 effective full-power years (EFPY) rather than the 28.4 EFPY proposed by Reference 2. The license amendment request supplement will also withdraw the proposed change to the reactor vessel surveillance schedule, as implementation of this change is not required for the MUR power uprate.	October 18, 2002