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INDIANA MICHIGAN POWER

June 29, 2001

C0601-21
10 CFR 50.90

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Units 1 and 2
RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT FOR
"CHANGES IN STEAM GENERATOR TUBE RUPTURE ANALYSIS
METHODOLOGY"
(TAC NOS. MB0739 AND MB0740)

References:
1) Letter from R. P. Powers (I&M) to Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment for Changes in Steam Generator Tube Rupture Analysis Methodology," C1000-11, dated October 24, 2000.

2) Letter from J. F. Stang (NRC) to R. P. Powers (I&M), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Regarding License Amendment Request," dated May 7, 2001, (TAC Nos. MB0739 and MB0740).

3) Letter from S. A. Greenlee (I&M) to NRC Document Control Desk, "Notification of a Due Date Extension for Response to a Request for Additional Information," C0501-18, dated May 31, 2001, (TAC NOS. MB0739 AND MB0740).

In Reference 1, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, proposed to amend

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Facility Operating Licenses DPR-58 and DPR-74 to change the CNP licensing basis as described in the Updated Final Safety Analysis Report. I&M proposed to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a postulated steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of CNP to a postulated SGTR with respect to SG overfill.

In Reference 2, the NRC informed I&M that additional information was needed to enable the NRC staff to adequately evaluate the proposed amendment. In Reference 3, I&M informed the NRC that the response to the requested information would be provided by June 30, 2001. The attachment to this letter provides the information requested in Reference 2.

There are no new commitments made in this submittal.

Should you have any questions, please contact Mr. Ronald W. Gaston, Manager of Regulatory Affairs, at (616) 697-5020.

Sincerely,

Scott A. Greenlee FOR

M. W. Rencheck
Vice President Nuclear Engineering

/bjb

Attachment

c: J. E. Dyer
MDEQ – DW & RPD
NRC Resident Inspector
R. Whale

ATTACHMENT TO C0601-21

RESPONSE TO NUCLEAR REGULATORY COMMISSION
STAFF REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT FOR CHANGES IN STEAM GENERATOR TUBE RUPTURE ANALYSIS METHODOLOGY

By letter dated October 24, 2000, from R. P. Powers, Indiana Michigan Power (I&M), to the Nuclear Regulatory Commission (NRC) Document Control Desk, I&M submitted a license amendment request to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a postulated steam generator tube rupture (SGTR). The NRC, by letter dated May 7, 2001, from J. F. Stang (NRC) to R. P. Powers (I&M), requested additional information regarding the October 24, 2000, submittal. The information provided below responds to the NRC's request for additional information.

NRC Question

"Section D, 'Plant Specific Submittal Requirement,' of Enclosure 1 of the safety evaluation for WCAP-10698-P-A, states that certain plant-specific input shall be provided when referencing the WCAP for licensing actions. Please provide the required information."

I&M Response

I&M is proposing to utilize limited aspects of the WCAP-10698-P-A methodology and associated computer codes to supplement Donald C. Cook Nuclear Plant's (CNP's) current licensing basis SGTR analysis to directly address postulated steam generator overfill. The proposed change in methodology would be applied only to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected SG in response to a SGTR event. This change in analysis technique is being proposed because it provides a more accurate representation of SG fill than CNP's current licensing basis. CNP's present methodology would be retained for calculating the radiological consequences of the postulated SGTR since the current CNP licensing basis methodology continues to bound the radiological consequences calculated by the new methodology. Since not all aspects of the WCAP-10698-P-A methodology are being adopted by I&M, not all five plant-specific input requirements, as described in Section D of the NRC safety evaluation report (SER) for WCAP-10698-P-A, are applicable to CNP.

The five items described in Section D of the SER for WCAP-10698-P-A are provided below with I&M's response following each item.

(1) **Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis. Demonstration runs should be performed to show that the**

**accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.**

The required operator action times used in the CNP-specific SGTR margin to overfill analysis were verified via simulator demonstration runs to assure the operators could mitigate the accident within a period of time compatible with overfill prevention. The bounding design basis analysis documented in WCAP-10698 is not being used by CNP, as discussed further in item (5), since the margin to SG overfill following a SGTR has been addressed by CNP-specific analyses.

The required operator action times used in the CNP-specific analysis have been incorporated into CNP's Operator Training program to assure continued adequate performance and continued justification for the use of the assumed times. SGTR scenarios were also included in the simulator validation of CNP's Emergency Operating Procedure (EOP) program. Validation includes, for example, verification that the EOP is compatible with plant design and can be performed within acceptable time limits.

(2) **A site-specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in Reference (1).**

As stated in I&M's October 24, 2000, submittal, CNP's current licensing basis calculation for offsite radiological consequences of a postulated SGTR remains bounding. Specifically, the mass transfer values used in the current licensing basis analysis for offsite dose consequences bound those calculated using the new WCAP methodology. Therefore, CNP's present methodology is being retained for calculating the offsite radiological consequences of a postulated SGTR.

(3) **An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.**

An evaluation of the structural adequacy of CNP's main steam lines under water filled conditions is not necessary. The margin to SG overfill analysis following a SGTR determined that an overfilled condition would not occur. This conclusion is supported by CNP operator action timing and plant simulator runs, which have also been validated as discussed in item (1) above.

(4) **A list of systems, components and instruments which are credited for accident mitigation in the plant specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components**

**state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured SG identification, provide the effect on the duration of the accident.**

As indicated in CNP's response to item (1), the license amendment request is intended to apply a more accurate analysis to demonstrate that CNP's current licensing basis of precluding SG overfill is maintained. A list of the systems, components and instruments credited in CNP's current SGTR analysis is provided in the table below.

The actions prescribed in CNP's EOPs for SGTR accident mitigation focus on four key aspects:

- isolating the ruptured SG,
- cooling the reactor coolant system (RCS),
- depressurizing the RCS, and
- terminating operation of the emergency core cooling system (ECCS).

The EOPs for a reactor trip or safety injection, and a steam generator tube rupture direct the operators to use specific equipment. The most challenging SGTR scenario with respect to SG fill includes a coincident loss of offsite power, which results in some of the equipment allowed to be used in the EOPs (consistent with the Westinghouse Emergency Response Guidelines) to be unavailable. Thus, the EOP-prescribed equipment presented in the tables and discussions below are based on offsite power not being available.

| Equipment/ Component Name | ID No. | Safety Grade (Y or N) | If Non-Safety Grade, is Safety Grade Backup Available? (Y or N) | Function | Remarks |
|---|---|---|---|---|---|
| SG Water Level – Narrow Range | BLP-1x0, -1x1, -1x2; where x corresponds to SG's 1 through 4 | Y | N/A | Ruptured SG Identification | |
| AFW Discharge Valve | FMO-211 & -212, -221 & -222, -231 & -232, or -241 & -242 | Y | N/A | Ruptured SG Isolation | |
| Main Feed Pump Trip | Unit 1: 1-OME-84-LPSVE, -LPSVW, -HPSVE, -HPSVW<br>Unit 2: 2-OME-84-SCVE, -SCVW | N | N | Feedwater Isolation | Notes 1 and 6 |

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|---|---|---|---|---|---|
| Main Feed Pump Discharge Valves | FMO-251 and –252 | N | N | Feedwater Isolation | Notes 1 and 6 |
| Feedwater Regulating Valves | FRV-210, -220, -230, -240 | N | N | Feedwater Isolation | Notes 1 and 6 |
| Feedwater Isolation Valves | FMO-201, -202, -203, -204 | N | N | Feedwater Isolation | Notes 1 and 6 |
| Steam Supply Valve to TDAFP | MCM-221, or –231 | Y | N/A | Ruptured SG Isolation | |
| SG Blowdown Isolation Valve | DCR-310, -320, -330, or –340 | N | N | Ruptured SG Isolation | Note 6 |
| SG Blowdown Sample Valve | DCR-301, -302, -303, or –304 | N | N | Ruptured SG Isolation | Note 6 |
| SG Stop Valve | MRV-210, -220, -230, or –240 | Y | N/A | Ruptured SG Isolation | Note 2 |
| SG PORVs (air-operated) | MRV-213, -223, -233, -243 | N | N | Ruptured SG Isolation and RCS Cooldown via Non-Ruptured SGs | Notes 3, 4, and 6 |
| Core Exit Thermocouples | NTR-1 through –65 | Y | N/A | Monitor RCS Cooldown | |
| Pressurizer PORVs (air-operated) | NRV-151, -152, and –153 | N | N | RCS Depressurization | Notes 4 and 6 |
| Pressurizer Water Level | NLI-151, NLP-151, -152, -153 | Y | N/A | ECCS Termination Criteria | Note 5 |

<u>Notes:</u>

1. The isolation of main feedwater is provided post-reactor trip via a feed pump trip, closure of the feed pump discharge valves (FMO-251 & -252), feed regulating valves (FRV-210, -220, 230, and -240), and the feedwater isolation valves (FMO-201, -202, -203, and –204).

2. The SG stop valves drain valve (DRV-407) and steam line warming valves (MS-144 and -143) are also closed, or check closed. These lines are not a significant secondary vent path, and thus timely closure is not critical.

3. The SGTR EOP directs the operators to increase the setpoint of the SG PORV on the ruptured SG from the nominal setpoint of 1025 psig to 1040 psig.

4. The SG PORVs form part of the main steam system pressure boundary upstream of the SG stop valves, and thus are safety-grade. However, the electrical and control air appurtenances for the SG PORVs are non-safety grade. Similarly, the pressurizer PORVs provide part of the safety grade RCS pressure boundary, but the control functions to manipulate these valves are classified as non-safety grade. Two of the

three pressurizer PORVs (NRV-152 and –153) have a continuous back-up source of air in the form of air bottles inside containment. As such, no operator action is required to align the back-up air bottles.

5. During the RCS depressurization, pressurizer level is restored to satisfy, in part, the ECCS termination criteria. Upon satisfying the ECCS termination criteria, the safety injection pumps are secured, all but one centrifugal charging pump (CCP) is stopped, and the boron injection tank is isolated. The remaining CCP is used to establish charging and letdown.

6. The Unit 1 and Unit 2 CNP licensing basis SGTR analysis described in Section 14.2.4 of the UFSAR credits non-safety grade equipment to mitigate the consequences of a SGTR accident. Thus, the use of the non-safety grade equipment, (e.g., SG and pressurizer PORVs) for accident mitigation following a SGTR in the margin to SG overfill analyses is consistent with CNP's licensing bases for both units.

The following radiation monitors could be used for identification of a SGTR accident.

- SG blowdown line,
- SG power-operated relief valve (PORV) line,
- gland steam condenser vent, and
- steam jet air ejector vent.

All four of the radiation monitor instruments listed above are non-safety grade, and are specifically listed in the EOPs as indications that can be used for identification of a SGTR event. These instruments are maintained and tested in accordance with CNP Technical Specifications and Offsite Dose Calculation Manual requirements. In accordance with 10 CFR 50.65, these instruments are also monitored within the scope of CNP's maintenance rule program. These monitors have not exceeded their maintenance reliability criteria.

The SGTR EOP allows identification of the ruptured SG by any of following indications:

a) directing chemistry personnel to sample all SGs for an activity analysis and perform a one-minute beta analysis, and

b) identifying the ruptured SG by one or more of the three indications:
   - an unexpected rise in SG narrow range level, or
   - high radiation from any SG sample, or
   - high radiation from any SG PORV.

The most limiting accident scenario with respect to SG tube overfill is the complete severance of one SG U-tube. The high primary to secondary flows that result from this design basis accident allow the ruptured SG to be promptly identified by the operators due to the unexpected rise in SG narrow range level. Thus, the most timely indication of the design basis SGTR is provided by the narrow range SG water level.

The chemistry sample provides a longer-term confirmatory indication of the ruptured SG for large break scenarios. The chemistry sample is only one of three methods specified in the SGTR EOP for the identification of the affected SG. The operator can use any one of the three methods for identifying a ruptured SG. The EOP actions focus primarily on the other two methods; checking the readings from the four radiation monitors discussed previously to see if the SG tubes are intact, and checking SG levels. The reactor trip EOP does include directions to sample the SGs in the event that the ruptured SG event was not identified by checking the radiation monitors. Thus, the chemistry sample of a ruptured SG provides confirmation, not primary indication, in the case of a design basis break.

In summary, the high primary to secondary flow rate due to a design basis break results in SG water level behavior and radiation monitor alarms that provide relatively quick indication of a ruptured SG. Therefore, the SG chemistry sample analysis time is not a critical aspect of the accident mitigating actions for scenarios that would be challenging with respect to SG fill.

**(5) A survey of plant primary and "balance-of-plant" system design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.**

This item is not applicable to CNP. CNP is not using the bounding plant analysis as described in WCAP-10698. As stated in Reference 1, CNP proposes to apply limited use of the WCAP-10698 methodology (and associated computer code) to supplement the CNP SGTR analysis with a CNP-specific margin to SG overfill analysis. The supplemental margin to SG overfill analysis used plant-specific analysis input values. Since the supplemental analysis is CNP-specific, the WCAP-10698 bounding plant analysis was not used, or needed. Therefore, there is no need to differentiate the CNP design from the generic plant design. Regarding worst single failure assumptions, CNP's current licensing basis SGTR analysis does not include a single failure. Thus, there is no need for CNP to consider a single failure in the SGTR margin to overfill analysis.

Comparison to No Significant Hazards Evaluation

CNP's response to the NRC's request for additional information does not affect the original evaluation performed in accordance with 10 CFR 59.92. The information provided in this response is considered supporting information and does not change the intent of CNP's original submittal.