REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8604250108 DOC. DATE: 86/04/22 NOTARIZED: NO DOCKET #
FACIL: 50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH. NAME AUTHOR AFFILIATION
ALEXICH, M. P. Indiana & Michigan Electric Co.
RECIP. NAME RECIPIENT AFFILIATION
DENTON, H. R. Office of Nuclear Reactor Regulation, Director (post 851125

SUBJECT: Provides listed addl info to close out several outstanding items re Unit 2, Cycle 6 reload. Response to Question 5 re resulting reactor coolant flow, core exit enthalpy & DNBR encl.

DISTRIBUTION CODE: A001D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 12
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	PWR-A ADTS		1	0	PWR-A EB		1	1
	PWR-A EICSB		2	2	PWR-A FOB		1	1
	PWR-A PD4 PD	01	5	5	WIGGINGTON, D		1	1
	PWR-A PSB		1	1	PWR-A RSB		1	1
INTERNAL:	ADM/LFMB		1	0	ELD/HDS3		1	0
	NRR/DHFT/TSCB		1	1	NRR/ORAS		1	0
	REG FILE	04	1	1	RGN3		1	1
EXTERNAL:	24X		1	1	EG&G BRUSKE, S		1	1
	LPDR	03	1	1	NRC PDR	02	1	1
	NSIC	05	1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR 24 ENCL 20

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

AEP:NRC:0916T

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License No. DPR-74
ADDITIONAL INFORMATION RELATED TO THE D. C. COOK UNIT 2 CYCLE 6 RELOAD

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

The purpose of this letter is to close out several outstanding items related to the Unit 2, Cycle 6 reload. The information is provided below.

1. <u>Additional Information Requested by the Reactor Systems Branch on January 8, 1986</u>

   On January 8, 1986, your staff transmitted to AEPSC a request for additional information related to the Unit 2, Cycle 6 reload analyses performed by the Exxon Nuclear Company (ENC).

   The response to questions numbered 1, 2, 3, 4, 7, and 11 were transmitted to you by ENC on March 5, 1986 in their letter RAC:017:86. Responses to questions numbered 6, 8, and 10 were submitted by ENC in their letter RAC:022:86, dated March 14, 1986. ENC's response to Question 10 was a commitment to provide additional analyses related to the control rod ejection accident. These analyses have been completed and were transmitted to you by ENC in their letter GNW:053:86, dated April 14, 1986.

   We transmitted our response to Question 9 in our letter AEP:NRC:0916P, dated March 27, 1986. Our response to Question 5, which concerns natural circulation flow from conditions of 20% power, is included as Attachment 1 to this letter. This response includes an analysis of DNBR, which was performed by ENC at your staff's request.

   Upon your acceptance of the information related to Questions 5 and 10, we believe that all open times identified in your January 8, 1986 letter should be closed.

A001
1/1

2. Additional Information Requested by the Reactor Systems Branch on April 3, 1986

On April 3, 1986 we received an additional six questions related to the Cycle 6 reload. These questions were discussed with your staff on April 7 and 10, 1986. During those discussions we were informed that Questions 1, 2, 3, and 6 would require no written response and would be closed out based on the verbal discussions. The response to Questions 4 and 5 were transmitted to you by ENC in their letter GNW:055:86, dated April 18, 1986. Assuming your acceptance of these responses, no open items remain concerning your April 3, 1986 questions.

3. LOCA Analysis

At the request of the NRC reviewer, Exxon is incorporating additional penalties in their Fuel Cooling Test Facility (FCTF) correlation factors. A reanalysis incorporating these penalties is to be performed. It is anticipated that the results will be submitted to the NRC by May 2, 1986.

4. Revisions to Proposed DNB Technical Specification 3/4.2.5.2.

In our letter AEP:NRC:0916I, dated March 15, 1986, we proposed to add a new Technical Specification (T/S) 3/4.2.5.2 (DNB Parameters-Modes 2 and 3). This proposed T/S extended requirements for DNB-related parameters from the current Mode 1 to include restrictions in Modes 2 and 3 also. During discussions with your staff on April 10, 1986, we were informed of their desire to extend the requirements to include restrictions in Modes 4 and 5. In response to this, we have included in Attachment 2 a revised version of T/S 3/4.2.5.2, which incorporates the additional restrictions. It is noted that incorporation of these changes also required changes to T/S Table 3.2-2, T/S 3.4.1.3, the Index section, and the Action statement of T/S 3.2.5.2. We have included revised versions of these pages in Attachment 2. Changes to the original submittal are indicated by an additional bar in the right-hand margin of the affected pages.

These revised changes have been reviewed by the Plant Nuclear Safety Safety Review Committee (PNSRC) and will be reviewed by the Nuclear Safety and Design Review Committee at their next regularly scheduled meeting.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President
RBK 4/22/86

Attachments

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

ATTACHMENT 1 TO AEP:NRC:0916T

RESPONSE TO QUESTION 5 ON JANUARY 8, 1986 TRANSMITTAL

RESPONSE TO QUESTION 5
OF NRC TRANSMITTAL ON 8 JANUARY, 1986

Question 5

Page 71 of XN-NF-85-28, Supplement 1 states that the P-7 setpoint is 11% with a 9% uncertainty allowance so that loss of flow without reactor trip would be limited to conditions of about 20% power. At other Westinghouse designed plants P-7 limits power under loss of flow conditions to 10% with 1% uncertainty. Justify that 20% power under natural circulation conditions could be accommodated without bulk boiling in the core or loss of DNB margin. Justify the power peaking factors used in the analysis by comparison to Technical Specification limits at 20% power. Provide the resulting reactor coolant flow, core exit enthalpy and DNBR.

Response

Pursuant to our discussions we have reviewed possible scenarios in an effort to identify those which might result in the condition indicated above. Our investigation has determined that the only credible scenario which can result in a loss of all reactor coolant flow below 10% power without a reactor trip is a Loss of Offsite Power. However, this transient will de-energize the electrical buses supplying the Reactor Coolant Pumps and will establish a Blackout condition.

A blackout condition automatically initiates a specific sequence of events including automatic shedding of various loads. The Control Rod Drive Motor Generator (MG) Sets will be shed from the 600 volt safety bus 2 seconds after the blackout is established by opening of the MG set 600 volt supply breakers which are safety grade. The MG sets will then coast down. These MG sets are designed incorporating a flywheel which prevents rod drops resulting from minor upsets in the MG set power supply. There is an interlock associated with the MG set output breaker which will trip the output breaker when the input breaker trips. When this output breaker is tripped, the control rods will fall into the core. The interlock mechanism and the MG output breaker are not safety grade and no credit is taken in the analysis of this accident for their action. For purposes of the analysis, the time span after the MG sets are tripped from their power supply to the point at which the control rods fall into the core is determined by the coast down characteristics of the MG sets and the Rod Control System.

An inquiry to our NSSS vendor, Westinghouse Electric Corporation, concerning the MG coastdown characteristics was answered with the following. "The flywheel is designed to provide power for one second with all rods out and two overlapped banks moving. Considering conservative design this time could be extended to as long as five seconds. Holding the rods stationary takes less power than moving them; therefore, if the control rods are withdrawn and stationary, the flywheel coastdown time could be increased by a factor of 10 to 20."

Based on the above we would expect that the control rods would fall into the core a maximum of 102 seconds following the Blackout. Reactor Coolant

System Flow Coast Down Measurements performed at D. C. Cook Unit 2 have demonstrated that following loss of all four coolant pumps, there is still approximately 8% flow after 102 seconds.

Exxon Nuclear Corporation has performed a calculation for this accident assuming no operator action and the control rods being inserted 102 seconds after the Loss of Flow accident is initiated. Their analysis follows and indicates that DNB is not strongly challenged, and hence after the safety grade trips occur, reactor operators may take positive action.

A Minimum DNBR of 1.8 was calculated using the XCOBRA-IIIC computer code and the EPRI-1 (EPRI-NP-2609, "Parametric Study of CHF Data", Electric Power Institute, Palo Alto, CA, September 1982) critical heat flux correlation at core conditions which conservatively bound those expected to occur for the four pump coastdown event from below the P-7 interlock setpoint. This result indicates that DNB is not strongly challenged during the event. Core boundary conditions employed are: 1) a reactor power of 20% of rated, 2) a coolant flow rate of 6.9% of design, 3) an inlet temperature of 560 degrees F, and 4) a core exit pressure of 2202 psia. The Technical Specification limit on F-delta H for zero power of 1.86 was employed along with the axial power profile given in Figure 15.0.3-1 of XN-NF-85-64(P), Rev. 1.

Question 5 inquires about conditions and power levels related to the P-7 setpoint. P-7 is an interlock which is developed from either the P-10 interlock or the P-13 interlock. P-10 is in turn developed from the Power Range Nuclear Instrumentation System and is calibrated to a setpoint of 10% reactor power. P-13 is developed from a turbine power signal and is calibrated to a setpoint equivalent to 10% turbine power. The phrasing of the question dealing with the 9% uncertainty tolerance indicates that the concern is with the establishment of P-7 as developed from P-10.

The initial condition prescribed; i.e., the reactor critical and power below the P-10/P-7 setpoint, constrains this accident to a transient operating condition, either during turbine roll and generator paralleling operations or during shutdown operations. During these transient conditions, two reactor operators and a supervisor are required to be present in the control room; typically a third operator is also present to assist in the operations conducted at these times. Since operation in these transient modes requires the active participation of the operating staff present and not merely the monitoring of plant conditions, particularly close attention is paid to all conditions and indications. If a total loss of flow occurred, it would be noted and the appropriate action instituted including tripping the reactor. Tripping the reactor is an action specified in the applicable procedures.

We believe that the increased number and awareness of the operators while in these transient modes ensures that the reactor will be tripped very quickly following the Loss of Flow transient and therefore reactor power will not increase substantially during the accident.

Our response to this question is summarized as follows:

1) This prescribed condition constrains this accident to transient operating conditions.

2) The only credible cause of this accident is a Loss of Offsite Power which will result in a Blackout condition.

3) This Blackout will result in a non-safety grade reactor trip approximately two (2) seconds following the Blackout initiation.

4) If no credit is taken for the non-safety grade trip, the control rods will fall into the core a maximum of 102 seconds following the Blackout initiation.

5) There will be approximately 8% coolant flow 102 seconds after the Blackout initiation.

6) Exxon has performed an analysis which demonstrates that DNB is not strongly challenged during this 102 second period following the blackout.

7) The plant operating conditions during the times this accident could occur are such that typically there are three reactor operators and one supervisor present. They would notice a Blackout condition and respond in accordance with approved procedures which specify ensuring the control rods are inserted.

ATTACHMENT 2 TO AEP:NRC:0916T

PROPOSED REVISED TECHNICAL SPECIFICATION PAGES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DNB PARAMETERS - MODES 2, 3, 4 and 5

LIMITING CONDITION FOR OPERATION

3.2.5.2 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

APPLICABILITY: MODES 2, 3*, 4* and 5*

ACTION:

MODES 2 and 3*

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or open the reactor trip system breakers within the next hour.

MODES 4* and 5*

Within one hour either open the reactor trip system breakers or render the control rod drive system incapable of rod withdrawal.

SURVEILLANCE REQUIREMENTS

4.2.5.2 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

TABLE 3.2-2

DNB PARAMETERS

PARAMETER	LIMIT
Reactor Coolant System T_{avg}	$\leq$ 549.2°F. (Reactor Subcritical)
Reactor Coolant System T_{avg}	$\leq$ 576.3°F. (Reactor Critical)
Pressurizer Pressure	$\geq$ 2176 psig

Reactor coolant loop operational requirements are contained in Specifications 3.4.1.1, 3.4.1.2.c and 3.4.1.3.c.

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The coolant loops listed below shall be OPERABLE and in operation as required by items b and c:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal - East, **

6. Residual Heat Removal - West **

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.***

c. At least three of the above reactor coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System**** and immediately initiate corrective action to return the required coolant loop to operation.