PRIORITY 1
ACCELERATED RIDS PROCESSING

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9506130049 DOC.DATE: 95/05/26 NOTARIZED: YES DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana M 05000316
AUTH.NAME AUTHOR AFFILIATION
FITZPATRICK,E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk) See ~~Rpts~~ Proposed Change to T.S.

SUBJECT: Application for amends to licenses DPR-58 & DPR-74.Amends would support operation of Unit 1 at SG tube plugging levels up to 30%,make one Unit 1 TS more nearly like corresponding Unit 2 TS & maintain consistency of Unit 2 criteria.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16+600
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-1 LA	1	1	PD3-1 PD	1	1
	HICKMAN,J	1	1			
INTERNAL:	FILE CENTER 01	1	1	NRR/DE/EMCB	1	1
	NRR/DRCH/HICB	1	1	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	NUDOCS-ABSTRACT	1	1
	OGC/HDS2	1	0			
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

PRIORITY 1 DOCUMENT

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 504-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 12 ENCL 11

Indiana Michigan
Power Company
P.O. Box 16631
Columbus, OH 43216

INDIANA
MICHIGAN
POWER

May 26, 1995 AEP:NRC:1207

Docket Nos. 50-315
50-316

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
LICENSE NOS. DPR-58 AND DPR-74
PROPOSED TECHNICAL SPECIFICATION CHANGES
SUPPORTED BY ANALYSES TO INCREASE UNIT 1
STEAM GENERATOR TUBE PLUGGING LIMIT AND
CERTAIN PROPOSED CHANGES FOR UNIT 2
SUPPORTED BY RELATED ANALYSES

This letter and its attachments constitute an application for amendment to the technical specifications (T/Ss) for Donald C. Cook Nuclear Plant units 1 and 2. Changes are proposed primarily to support operation of Cook Nuclear Plant unit 1 at steam generator tube plugging levels up to 30%. In addition, analyses and evaluations were performed to support increased operating margins for unit 1. Changes to the T/Ss based on those analyses and evaluations are proposed. Some of the proposed changes are proposed for both unit 1 and unit 2. Finally, one miscellaneous change is proposed to make one unit 1 specification more nearly like the corresponding unit 2 specification and one administrative change is proposed to maintain consistency of unit 2 acceptance criteria.

The small break loss of coolant accident analysis which is submitted under this letter was performed using new Westinghouse Electric Corporation models. The new models employ new methods for modeling safety injection to the broken loop and an improved steam condensation model. These models were submitted for NRC review by Westinghouse Electric Corporation under a letter dated December 14, 1994, identified as NTD-NRC-94-4278, to the Document Control Desk from N. J. Liparulo.

A description of the proposed changes and analysis concerning significant hazards consideration pursuant to 10 CFR 50.92 is contained in Attachment 1. Attachment 2 contains the proposed, revised T/S pages. Attachment 3 contains the existing T/S pages marked to reflect proposed changes. Attachment 4 is a summary

A001

description of the proposed T/S changes. It contains a brief summary of each proposed change and directs the reviewer to the supporting documentation. Attachment 5 is a discussion of earlier related submittals. The analyses described in the earlier submittals contain margin, particularly power margin, which is allocated in the most recent evaluations to allow an increase in steam generator tube plugging. Attachment 5 also addresses previously submitted analyses for unit 2 which support the proposed changes to both units. Attachment 6 is a description of analyses performed by Westinghouse Electric Corporation for Cook Nuclear Plant, unit 1.

Some of the changes proposed for unit 1 are also proposed for unit 2. Attachment 7 contains excerpts from descriptions of analyses performed by Westinghouse Electric Corporation for Cook Nuclear Plant, unit 2. These excerpts are from documents previously submitted for review as described in Attachment 5. Attachment 7 is provided for the convenience of the reviewer and supports the unit 2 portion of the proposed changes.

Our letter AEP:NRC:1211, which has previously been submitted for approval, proposes to eliminate surveillance requirements to test emergency core cooling system and auxiliary feedwater pumps via their recirculation flow paths and to reference the inservice testing program, technical specification 4.0.5. Nevertheless, proposed increases in allowable pump head degradation supported by the analyses described in the attachments to this submittal are also included in this document in the current form of these pump surveillances. These technical specification changes are being proposed in the event that review of AEP:NRC:1211 is not completed by the time this submittal is approved. Approval of the proposed changes in AEP:NRC:1211 implies the withdrawal of the proposed changes to the corresponding technical specification pages in this submittal as discussed in Attachment 1.

In addition, time response technical specification table changes supported by the analyses and required for implementation of the proposals in this submittal are not included in this submittal. This was done because our letter AEP:NRC:1210, which has already been submitted, removes the affected time responses from the technical specifications. Approval of AEP:NRC:1210 is required to fully implement this submittal.

Approval of the changes in this submittal is needed by December 1, 1996, to support unit 1, cycle 16 operation.

We believe that the proposed T/S changes will not result in (1) a significant change in the types of effluents or a significant increase in the amount of effluent that might be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

These proposed T/S changes have been reviewed and approved by the Plant Nuclear Safety Review Committee and by the Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to the Michigan Public Service Commission and the Michigan Department of Public Health.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 26th DAY OF May 1995

Rita M Hill
Notary Public

My Commission Expires: 6-28-99

cad

Attachments

cc: A. A. Blind
G. Charnoff
J. B. Martin - Region III
NFEM Section Chief
NRC Resident Inspector
J. R. Padgett

ATTACHMENT 1 TO AEP:NRC:1207

REASONS AND 10 CFR 50.92 ANALYSIS FOR
CHANGES TO THE
COOK NUCLEAR PLANT UNIT NOS. 1 AND 2
TECHNICAL SPECIFICATIONS

.9506130049

INTRODUCTION

The primary purpose of this submittal is to request approval to operate Cook Nuclear Plant unit 1 with steam generator tube plugging (SGTP) levels as high as 30% in each steam generator. Since the analysis needed to support this request involved reanalysis or evaluation of most of the events discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR), we performed the analyses such that additional operating margin was achieved in several areas. In addition, a number of proposed changes which can now be supported for both units, a miscellaneous change, and an administrative change are being proposed. The proposed changes are discussed in greater detail below and in Attachment 4, Summary Description of Proposed Increased Steam Generator Tube Plugging Technical Specifications. Attachment 4 is provided to assist the reviewer. It contains a brief summary of each change and directs the reviewer to the supporting documentation. Attachment 2 contains the proposed technical specification changes. Attachment 3 contains the current technical specification pages marked up to reflect the proposed changes.

The principal analytical support for this submittal is WCAP 14285 which is included as Attachment 6 to the submittal. As discussed in Section 2.0 of WCAP 14285, the new analyses described in WCAP 14285 replace analyses performed earlier to support the operation of Cook Nuclear Plant. The evaluations described in WCAP 14285 are based on those earlier analyses. The earlier analyses are described in WCAP 11902 and WCAP 11902 Supplement 1, references 3 and 10 of Attachment 5. They are referred to as the "Rerating Program" in WCAP 14285. The analyses in the "Rerating Program" contain margin, particularly power margin, which is allocated in WCAP 14285 to increased SGTP. The "Rerating Program," together with WCAP 14285 and previously submitted analyses for unit 2, support the proposed changes to both units. The earlier analyses, associated submittals, and license amendments are discussed in Attachment 5 for the convenience of the reviewer.

Some of the changes proposed for unit 1 are also proposed for unit 2. Attachment 7 contains excerpts from descriptions of analyses performed by Westinghouse Electric Corporation for Cook Nuclear Plant, unit 2. These excerpts are from documents previously submitted for review as described in Attachment 5. Attachment 7 is provided for the convenience of the reviewer and to support the unit 2 portion of the changes which are proposed for both units.

DESCRIPTION OF CHANGES

The proposed changes are discussed in related groups.

Group 1: Changes Directly Related to Increased Steam Generator Tube Plugging

The first group of these changes is directly related to the increase in SGTP allowance. The changes include a decrease in primary flow, and a new upper limit on reactor coolant system Tavg. The existing Tavg limit has not been changed since the original approval to operate unit 1 at reduced temperature and pressure, which is discussed in Attachment 5. This value is significantly conservative to the analyses. The current limit on Tavg is based on evaluations which were performed in support of the reduced temperature and pressure program. The proposed departure from nucleate boiling (DNB) upper temperature limit was calculated from the upper limit of the temperature window on which the analyses and evaluations in Attachment 6 are based. Another impact of the new analyses and evaluations is a reduction in the contained volume of the reactor coolant system. The analyses and evaluations described in Attachment 6 support the proposed, lower value for minimum measured flow (MMF), the proposed value for upper limit on Tavg, and the proposed value for reactor coolant system volume.

The summary in Attachment 4 provides a brief description of each proposed change and a cross reference to the analyses. This group of proposed changes is found in the following technical specifications:

Reduce MMF
- Table 2.2-1
- Table 3.2-1

Increase DNB Temperature Limit
- Table 3.2-1

Reduce Reactor Coolant System Volume
- 5.4.2

Group 2: Changes Proposed to Increase Unit 1 Operating Margin

The second group of changes results from analyses and evaluations designed to increase operating margin. Since most of the events described in Chapter 14 of the UFSAR had to be reanalyzed or evaluated to support the increased steam generator tube plugging limit, the effort to increase some margins was performed at the same time.

The first proposed changes in this group are changes to the

overtemperature delta T (OTΔT) and overpower delta T (OPΔT) reactor trip setpoints. The new setpoints are based on new core thermal safety limits. As a result of temperature streaming in the reactor system coolant system hot legs, there is an inaccuracy in the measurement of Thot in the resistance temperature detector (RTD) bypass lines. This streaming is a function of the core power distribution. Drifts in the Delta T measurements at full power as a function of burnup result from this phenomenon. When the deviations exceed the instrument allowances for hot leg streaming, it is necessary to recalibrate the OTΔT and OPΔT system. The thermal safety limits and associated OTΔT and OPΔT setpoints were reanalyzed to increase the allowance between the safety analysis limits and technical specification setpoints at the licensed reactor power of 3250 MWt. This permits the use of larger allowances for Delta T drift.

The changes which are proposed in this submittal are based on analyses performed by both us and our contractor, Westinghouse Electric Corporation. Westinghouse performed calculations to ensure that the analysis values for the OTDT and OPDT setpoints are consistent with safety limits and that acceptable results are obtained for the affected transients. We performed calculations to ensure that adequate margin exists between the analysis value and the technical specification nominal value of the OTDT and OPDT reactor trip setpoints. The associated allowable values proposed for notes 3 and 4 of technical specification Table 2.2-1 in Attachments 2 and 3 are based on our calculations.

The second proposed change in this group supports an increase in the allowed degradation of the unit 1 auxiliary feedwater (AFW) pumps. Currently, all analyses in the unit 1 license basis assume a head degradation of approximately 12%. However, the feedwater line break reported in the unit 1 UFSAR for information only assumes an AFW pump degradation of only 5%. All unit 2 analyses, including feedwater line break, support 12% head degradation, which is reflected in the unit 2 technical specifications.

Rather than reanalyze the unit 1 feedwater line break transient, an evaluation to show that the unit 2 feedwater line break bounds unit 1 is provided in WCAP 14285. We have chosen to address the this analysis to support the increased degradation for the auxiliary feedwater pump, even though the information only feedwater line break analysis is not part of the unit 1 license basis.

The evaluation referenced above is contingent upon changing the safeguards actuation logic in unit 1 from the old logic to the hybrid logic presently used by unit 2. The plant design change to modify the unit 1 safeguards actuation logic will be made prior to the beginning of cycle 16. This design change will be performed in accordance with the design change process.

Although the proposal to allow increased AFW pump degradation impacts the response times which are currently included in technical specification tables, this submittal does not include proposed changes to these technical specifications. We have made a separate submission in parallel with this submission pursuant to generic letter 93-08 which proposes to relocate response times to the UFSAR. The generic letter 93-08 submittal is identified as AEP:NRC:1210. The time response portion of these increased tube plugging analyses will not be implemented without approval of the generic letter 93-08 submittal.

We have also submitted a proposal (AEP:NRC:1211) to eliminate surveillance requirements to test emergency core cooling system pumps and auxiliary feedwater pumps on recirculation and to reference the inservice testing program. If AEP:NRC:1211 is approved prior to this submittal, technical specification sections 4.7.1.2 and 4.5.2.f are withdrawn.

The next two proposed changes in this group are related to changes made to the input assumptions for the analyses reported in WCAP 14285. Specifically, the residual heat removal and high head safety injection pumps head degradations were increased to 15% and the pressurized safety valve setpoint tolerance was increased to ±3%.

The analyses and evaluations described in WCAP 14285, Attachment 6, support the proposed technical specification changes to increase operating margin as described above.

The summary in Attachment 4 provides a brief description of each proposed change and a cross reference to the analyses. This group of proposed changes is found in the following technical specifications:

Revise Safety Limits and OPDT/OTDT Reactor Trip Setpoint
- Figure 2.1-1
- Table 2.2-1
- B 2.1.1 Reactor Core
- B 2.2.1 Overtemperature Delta T
- B 2.2.1 Overpower Delta T
- B 3/4.4.1 Reactor Coolant Loops

Increase Allowable Unit 1 AFW Pump Degradation to 12%
Table 3.3-3
Table 3.3-4
Table 4.3-2
4.7.1.2 *

Increase Unit 1 Allowable RHR and HHSI Pump Degradation to 15%
4.5.2.f *

Increase Unit 1 Pressurizer Safety Valve Tolerance
3.4.2
3.4.3

* NOTE: Withdraw if submittal AEP:NRC:1211 is approved.

Group 3: Changes Proposed to Increase the Operating Margin of Both Units

The third group of proposed changes affects both units. They are supported for unit 1 by WCAP 14285 and the "Rerating Program." They are supported for unit 2 by WCAP 14285, the "Rerating Program," and earlier analyses specifically performed for unit 2. Specific references to the earlier analyses are provided in Attachment 4.

The first change in this group is a proposal to reduce the required shutdown margin in modes 1, 2, 3, and 4 from 1.6% to 1.3%. This proposal is supported for both units by the steam mass and energy release (SM&E) to containment analysis in WCAP 14285. This analysis was performed to correct some incorrect analysis assumptions. It is supported for both units by the SM&E outside containment addressed in the "Rerating Program." It is supported for unit 1 by the core response steam break analysis in WCAP 14285 and for unit 2 by the core response steam break discussed in the Vantage 5 Reload Transition Safety Report (RTSR). Approval for the unit 2 core response steam break was obtained in Amendment 134 to DPR-74, the unit 2 license.

The second proposed change is an increase in the allowable centrifugal charging pump (CCP) head degradation to 10%. All the analyses reported in WCAP 14285, the "Rerating Program," and the RTSR were performed using CCP curves consistent with this level of head degradation.

We have also submitted a proposal (AEP:NRC:1211) to eliminate surveillance requirements to test CCP pumps on recirculation and to reference the inservice testing program. If AEP:NRC:1211 is approved prior to this submittal, technical specification sections 4.1.2.4.1, 4.1.2.4, and 4.5.2.f are withdrawn.

The third proposed change in this group reduces the minimum RWST temperature to 70°F. The core response large break loss of coolant accident (LBLOCA) analyses reported in WCAP 14285 for unit 1 and the RTSR for unit 2 support this assumption.

The last change in this group affects the technical specification bases. The peak pressure of the long term containment integrity analysis is reported in Basis section 3/4.6.1.4. The new analysis reported in Attachment 6 bounds both units at a power of 3411 MWt. The proposed change to the technical specification bases reflects the new analysis.

The summary in Attachment 4 provides a brief description of each proposed change and a cross reference to the analyses. This group of proposed changes is found in the following technical specifications:

Decrease Required SDM to 1.3%, Both units
- 3.1.1.1
- 4.1.1.1.1
- B 3/4.1.1.1
- B 3/4.1.1.2

Increase Allowable CCP Degradation to 10%, Both units
- 4.1.2.4.1 *
- 4.1.2.4 *
- 4.5.2.f *

Reduce Minimum RWST Temperature to 70°F, Both units
- 3.1.2.7
- 3.1.2.8
- 3.5.5
- B 3\4.5.5

Change Peak Containment Pressure to Reflect New Analysis, Both units
- B 3/4.6.1.4
- B 3/4 6.1.5

* NOTE: Withdraw if submittal AEP:NRC:1211 is approved.

Group 4: Administrative Change

The final group consists of administrative changes. Specifically, the proposal changes the surveillance acceptance criteria for the unit 2 RHR and SI pumps from discharge pressure to differential pressure. No increase in allowable degradation is being proposed for these pumps. This change is proposed because in conjunction with proposing an increase in allowable RHR and SI pump degradation for unit 1 and an increase in

allowable charging pump degradation for both units, the proposed surveillance acceptance criteria for the degradation changes are formulated in terms of differential pressure. Therefore, to maintain consistent surveillance acceptance criteria format, it is proposed to change the acceptance criteria for the unit 2 RHR and SI pumps from discharge pressure to differential pressure.

We have also submitted a proposal (AEP:NRC:1211) to eliminate surveillance requirements to test emergency core cooling system (ECCS) pumps on recirculation and to reference the inservice testing program (IST). If AEP:NRC:1211 is approved prior to this submittal, the change to technical specification 4.5.2.f is withdrawn.

The summary in Attachment 4 provides a brief description of each proposed change and a cross reference to the analyses. This group of proposed changes is found in the following technical specification:

> Change RHR and SI Pump Acceptance Criteria from Discharge Pressure to Differential Pressure, unit 2
>
> 4.5.2.f *

* NOTE: Withdraw if submittal AEP:NRC:1211 is approved.

Group 5: Time Response Changes not submitted

As indicated above in the discussion of group 2 changes, Changes Proposed to Increase unit 1 Operating Margins; potential changes which impact response times are not being submitted. We have made a separate submission (AEP:NRC:1210) in parallel with this submission pursuant to generic letter 93-08 which proposes to relocate response times to the UFSAR. In view of the submittal to move response times to the UFSAR, certain protection response times are also not included in this submittal. The analyses have been modeled with times that include margin relative to the values currently in the technical specifications. The new values will be placed in the UFSAR when the submittal pursuant to generic letter 93-08 is approved.

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

10 CFR 50.92 specifies that the holder of an operating license or construction permit of a nuclear power facility participate in determining whether a change to the T/S's current licensing basis (CLB) involves a significant hazards consideration. Prior to implementation of a change to the CLB, the Nuclear Regulatory Commission must review and make a final determination, pursuant to the procedures in 10 CFR 50.91, that a proposed amendment to

the operating license involves no significant hazards considerations. In order to satisfactorily complete the review, the proposed amendment to the CLB must not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated,

2. create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. involve a significant reduction in a margin of safety.

For the purpose of performing a significant hazards consideration analysis, the four groups of technical specification changes discussed under Description of Changes can be reduced to three groups. In evaluating significant hazards, the first three groups of proposed technical specifications will be considered together. The miscellaneous change and the administrative change will each be considered separately.

DETERMINATION OF NO SIGNIFICANT HAZARDS FOR CHANGES BASED ON ANALYSES AND EVALUATIONS (Groups 1, 2, and 3)

Criterion 1

Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The analyses which were performed to support the first three groups of proposed changes were performed in accordance with approved methodologies and acceptance criteria applicable to Cook Nuclear Plant. The proposed technical specification changes do not involve postulated initiators for analyzed events. Therefore, the probability of accidents can not be affected. The analyses and evaluations performed all met applicable acceptance criteria. Therefore, the consequences of accidents previously evaluated are unaffected.

Criterion 2

Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The analyses which were performed to support the second and third groups of proposed changes address increases in operating margin for accident mitigators. They do not create the possibility of new accidents. The first group of proposed

changes to reduce minimum measured primary flow, increase the DNB temperature limit, and reduce the reactor coolant system volume have been analyzed or evaluated. The proposed DNB limit is consistent with the DNB design and does not constitute an accident initiator. The new volume results from the new value of allowed tube plugging and is consistent with the analysis. It is not an accident initiator.

The impact of the reduced primary flow in the primary system was analyzed or evaluated, as appropriate. All applicable criteria were satisfied. No new or different kind of accident resulted.

Criterion 3

Do the proposed changes involve a significant reduction in a margin of safety?

No. The margin of safety is provided for the primary pressure boundary and other components in part by applicable design codes. The margin of safety for the various accidents and transients is maintained by the analysis acceptance criteria. Since the components remain in compliance with the codes and standards in effect when Cook Nuclear Plant was licensed and applicable acceptance criteria are met, the margin of safety is not reduced by the 30% SGTP program.

DETERMINATION OF NO SIGNIFICANT HAZARDS FOR ADMINISTRATIVE CHANGES (Group 4)

Criterion 1

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change involves the surveillances for mitigating equipment. Therefore, it has no impact on probability. The proposed change also has no impact on the consequences of an accident because the criteria for operable RHR and SI pumps does not change. The change is only in the parameter that will be compared with the required criteria, the differential pressure instead of the discharge pressure.

Criterion 2

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

No. Nothing is changed with regard to accident initiators. The surveillance criteria for the RHR and SI pumps, which are mitigating equipment, is unchanged. The proposed change can have no impact on accident initiators.

Criterion 3

Does the proposal involve a significant reduction in a margin of safety?

No. The proposal does not change the requirements for a pump to be operable. Only the parameter compared to acceptance criteria changes. The underlying criteria is unchanged. Therefore, there is no change in the margin of safety.

CONCLUSION

It is concluded that operation of Cook Nuclear Plant units 1 and 2 with the changes proposed above does not involve any significant hazards as defined in 10 CFR 50.92.

ATTACHMENT 2 TO AEP:NRC:1207

PROPOSED CHANGES TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2
TECHNICAL SPECIFICATIONS

PROPOSED CHANGES TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 1
TECHNICAL SPECIFICATIONS