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SUBJECT: Application for amend to License DPR-58,decreasing min measured flow requirement as found in Table 3.2-1 & revising RCS limit Figure 2.1-1 & Table 2.2-1 functional unit 12 footnote.Rept withheld.

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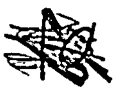

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AEP
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AEP:NRC:1130

Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License No. DPR-58
TECHNICAL SPECIFICATIONS CHANGE FOR UNIT 1 CYCLE 11

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

Attn: T. E. Murley

July 23, 1990

Dear Dr. Murley:

This letter constitutes an application for amendment to the Technical Specifications (T/Ss) for Donald C. Cook Nuclear Plant Unit 1. Due to the possibility of increased steam generator tube plugging, additional margin is sought to the minimum measured flow requirement referenced in T/S 3.2.5 (Reactor Coolant System Total Flow Rate), and associated changes to Safety Limits 2.1.1 (Reactor Core) and 2.2.1 (Reactor Trip System Instrumentation Setpoints). Specifically, we are proposing to decrease the minimum measured flow requirement as found in Table 3.2-1 and to revise Reactor Core Safety Limit Figure 2.1-1 and the Table 2.2-1 Functional Unit 12 footnote to reflect this change.

Currently, Unit 1 has 2.8% margin to its minimum measured flow T/S. If significant additional plugging is required during the upcoming outage, approaching our currently analyzed limit of 10% average plugging, approximately 1.2% in flow would be lost, leaving only 1.6% margin for measurement repeatability. Thus, although the actual flow is expected to be sufficient, measurement fluctuations could result in a failure to meet the T/S requirement.

Unit 1 will undergo extensive steam generator tube testing and repair which may result in additional tube plugging during the upcoming refueling outage. The flow is measured soon after reaching Mode 1, which could be as early as December 19, 1990. Therefore, we request that you respond to this proposal by December 12, 1990.

A001 1/1

A description of the change and our analyses concerning significant hazards are contained in Attachment 1. The proposed revised T/S pages are contained in Attachment 2. Attachment 3 contains an evaluation of the proposed changes from Westinghouse.

For the reviewer's convenience, Attachment 4 contains reports referenced in the Westinghouse evaluation in Attachment 3. Attachment 4 includes:

4A. One copy of WCAP-12568, "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for American Electric Power, D. C. Cook Unit 1 Nuclear Power Station" (PROPRIETARY).

4B. One copy of WCAP-12569, "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for American Electric Power, D. C. Cook Unit 1 Nuclear Power Station" (NON-PROPRIETARY).

4C. A Westinghouse authorization letter, CAW-90-046, Proprietary Information Notice, and accompanying Affidavit.

As Item 4A contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the supporting Westinghouse Affidavit should reference CAW-90-046 and should be addressed to R. A. Wiesemann, Manager of Regulatory & Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

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

These changes have been reviewed by the Plant Nuclear Safety Review Committee and 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 Mr. J. R. Padgett of the Michigan Public Service Commission and the NFEM Section Chief of the Michigan Department of Public Health.

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

Sincerely,

M. P. Alexich
Vice President

MPA/eh

Attachments

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
G. Charnoff
NFEM Section Chief
J. R. Padgett
NRC Resident Inspector - Bridgman
A. B. Davis - Region III

ATTACHMENT 1 TO AEP:NRC:1130

REASONS AND 10 CFR 50.92 SIGNIFICANT HAZARDS EVALUATION

FOR CHANGES TO THE DONALD C. COOK NUCLEAR PLANT UNIT 1

TECHNICAL SPECIFICATIONS

Introduction

This letter requests modification to Technical Specification (T/S) 3.2.5 (Reactor Coolant System Total Flow Rate), and associated changes to Safety Limit 2.1.1 (Reactor Core) and 2.2.1 (Reactor Trip System Instrumentation Setpoints). Specifically, we are proposing to decrease the minimum measured flow (MMF) requirement as found in Table 3.2-1 from 366,400 gpm to 361,600 gpm; to revise Reactor Core Safety Limit Figure 2.1-1 to reflect this change in MMF; and to revise the Table 2.2-1 Functional Unit Number 12 footnote to reflect the new MMF. In addition, the mathematical symbols have been written out as an administrative change to Table 2.2-1.

The MMF requirement assures that reactor coolant system (RCS) flow meets the assumptions used in the NSSS design calculations and the accident and transient analyses. For those departure from nucleate boiling (DNB) transients that are analyzed using the Improved Thermal Design Procedure (ITDP) (Reference 1), the initial RCS flow is assumed to be the MMF. For these analyses, uncertainties associated with the flow measurement are incorporated into the DNBR limit value. For the NSSS design calculations, non-DNB related accident and transient analyses, and DNB transient analyses for which the ITDP is not used, the initial RCS flow assumed is the thermal design flow (TDF), which is 354,000 gpm for Cook Nuclear Plant Unit 1. For these analyses, flow uncertainty is accounted for by the Technical Specification requirement (MMF) being larger than the TDF plus the calculated uncertainty.

The actual calculated RCS flow measurement uncertainty is 2.1% (Reference 1). The current T/S MMF value is based on the original design value of 3.5%. This evaluation removes this margin from the uncertainty value. Thus the new T/S will still assure the TDF is met. Evaluations are required for DNB calculations using the ITDP methodology.

Westinghouse has evaluated the Cook Nuclear Plant Unit 1 safety analyses and NSSS design calculations for the reduced RCS MMF (see Attachment 3). It has been determined that the conclusions of the safety analyses remain valid for the reduction in RCS flow, and the NSSS design calculations are unaffected by the reduction in the MMF requirement.

10 CFR 50.92 Evaluation

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

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

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

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Given the change, the thermal design flow which is assumed in NSSS design and non-ITDP UFSAR Chapter 14 analyses will still be met. Existing departure from nucleate boiling margin has been allocated to offset the change in minimum measured flow in the remaining UFSAR Chapter 14 analyses. Thus, the change is not expected to involve a significant increase in the probability or consequences of a previously analyzed accident.

Criterion 2

The change will not change the design or operation of the plant. Thus, it would not be expected to create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

Criterion 3

The original thermal design flow analyses assumptions for NSSS design and non-ITDP UFSAR Chapter 14 analyses will be met, and existing departure from nucleate boiling margin has been allocated for ITDP UFSAR Chapter 14 analyses. Thus, the subject evaluations have been demonstrated to comply with the licensing basis of the plant and in fact, involve no reduction in previously reported analysis results. Therefore, although the change may be construed as involving a reduction in the margin of safety, this will not be significant from a safety or licensing viewpoint.

We note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes that may reduce in some way a safety margin, but the results of which are within limits established as acceptable. As discussed above, the change is consistent with our licensing basis

and involves no reduction in results of analyses previously reported to the NRC. Therefore, we believe the example cited is applicable and that the changes should not involve significant hazards consideration.

References

1) "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for American Electric Power D. C. Cook Unit 1 Nuclear Power Station" WCAP 12568, April 1990. Provided as Attachment 4 for the reviewer's convenience.

ATTACHMENT 2 TO AEP:NRC:1130

PROPOSED REVISED

TECHNICAL SPECIFICATION PAGES

ATTACHMENT 3 TO AEP:NRC:1130

WESTINGHOUSE EVALUATION FOR A

REDUCTION IN THE TECHNICAL SPECIFICATION

MINIMUM MEASURED FLOW REQUIREMENT

ATTACHMENT 1

DONALD C. COOK NUCLEAR PLANT UNIT 1

SAFETY EVALUATION FOR A REDUCTION IN THE
TECHNICAL SPECIFICATION MINIMUM MEASURED FLOW REQUIREMENT

BACKGROUND

Technical Specification (T/S) 3.2.5 (Table 3.2-1) requires that the Reactor Coolant System (RCS) Total Flow Rate be greater than or equal to an indicated value of 366,400 gpm. This Minimum Measured Flow (MMF) requirement assures that RCS flow meets the assumptions used in the NSSS design calculations and the accident and transient analyses.

For those DNB transients that are analyzed using the Improved Thermal Design Procedure (ITDP) (Table 1) the initial RCS flow is assumed to be the MMF. For these analyses, uncertainties associated with the flow measurement are incorporated into the DNBR limit value. For the NSSS design calculations, non-DNB related accident and transient analyses (Table 2), and DNB transient analyses for which the ITDP is not used (Table 2), the initial RCS flow assumed is the Thermal Design Flow (TDF), which is 354,000 gpm for Cook Nuclear Plant Unit 1. For these analyses, flow uncertainty is accounted for by the Technical Specification requirement (MMF) being larger than the TDF plus the calculated uncertainty.

The purpose of this safety evaluation is to determine the impact of a reduction in the Cook Nuclear Plant Unit 1 Technical Specification MMF requirement on the NSSS design calculations and the accident and transient analyses. The MMF requirement reduction is from 366,400 gpm to 361,600 gpm.

EVALUATION

All of the Cook Nuclear Plant Unit 1 safety analyses and the NSSS design calculations have been reviewed to determine the impact of a reduction in the RCS MMF Flow Technical Specification requirement from 366,400 gpm to 361,600 gpm. This section summarizes the effects of a reduction in the MMF requirement on the safety analyses and the NSSS design calculations.

NSSS Design Calculations

As stated above, the NSSS design calculations assume a minimum RCS flow consistent with TDF. To assure that there is no impact on the TDF, the minimum measured flow requirement must be larger than the TDF plus uncertainties.

The measurement uncertainty associated with MMF is 2.1% (Reference 1). The proposed change to lower the MMF to 361,600 gpm will, therefore, not impact TDF (354,000 gpm).

$$361{,}600 - 0.021 \times 361{,}600 = 354{,}006.4 > 354{,}000$$

Since there is no impact on the TDF, the NSSS design calculations are not affected by the change to the MMF requirement.

Non-DNB Safety Analyses and Non-ITDP DNB Safety Analyses

These safety analyses also assume a minimum RCS flow consistent with TDF. Since there is no impact on the TDF, the non-DNB and DNB non-ITDP safety analyses are also not affected by the change to the MMF requirement.

DNB ITDP Safety Analyses

A reduction in MMF for Cook Nuclear Plant Unit 1 impacts the analysis of the DNB events analyzed with ITDP in three areas: 1) the DNBR design limits and/or margin will be affected, 2) the Reactor Core Safety Limits (T/S Figure 2.1-1) will be affected, which can potentially affect the Overtemperature ΔT and Overpower ΔT reactor trip setpoints (T/S Table 2.2-1) that are based on these core limits, and 3) the initial RCS flow value assumed is reduced. Each of these areas is discussed in detail below.

1. DNBR Calculations

The design limit DNBR values for Cook Nuclear Plant Unit 1 are 1.323 for the thimble cell and 1.334 for the typical cell. These are the minimum DNBR values that are required in order to meet the DNB design basis for analyses that use the ITDP. However, the safety analyses conservatively use a safety limit DNBR of 1.45 for both thimble and typical cells. This allows for DNBR penalties to be offset with available margin between the design and safety limit DNBR values. A reduction in core flow is a penalty with respect to the calculation of DNBR. For Cook Nuclear Plant Unit 1, sufficient margin is available and has been allocated to accommodate the MMF reduction.

2. Reactor Core Safety Limits and Protection Setpoints

The Reactor Core Safety Limits (T/S Figure 2.1-1) represents the loci of points of thermal power, RCS average temperature, and RCS pressure for which the DNBR is equal to the safety analysis limit DNBR (1.45) or the average enthalpy at the vessel exit is equal to the saturated liquid enthalpy (vessel exit boiling). For a given RCS pressure, the lines at high power levels are DNBR limits, while the lines at lower powers are vessel exit boiling limits. The vessel exit boiling limits are lowered from the actual saturated enthalpy line to account for flow uncertainty, since the RCS MMF requirement (T/S 3.2.5) is the same as that used as the basis of the Reactor Core Safety Limits. The current T/S Figure 2.1-1 is based on a MMF of 366,400 gpm.

The DNBR lines remain unchanged for the MMF reduction, since DNBR margin has been allocated as described above. However, the vessel exit boiling lines represent a physical limit (biased for flow uncertainty as described above). A reduction in the RCS flow results in these lines becoming more restrictive. That is, for a given pressure and power, exit boiling will occur at a lower RCS temperature. Thus, the Reactor Core Safety Limits figure in the Technical Specifications should be revised.

The Overtemperature ΔT and Overpower ΔT reactor trips are designed to prevent the Reactor Core Safety Limits from being exceeded. Since the flow is reduced and the vessel exit boiling limits change, the current setpoints for these trip functions were examined. Setpoint calculations were performed using the methodology described in Reference 3. These calculations confirm that the current trip setpoints continue to provide core protection for the reduced MMF. As noted in Reference 2, it is assumed that the reference average temperature (T′ and T′′) in the setpoint equations are rescaled to the full power average temperature each time the cycle average temperature is changed. Also, the reference pressure (P′) in the Overtemperature ΔT setpoint equation is assumed to be set to the appropriate nominal primary system pressure (2100 psia or 2250 psia).

3. Initial RCS Flow Assumption

The transient analysis of DNB events that use the ITDP assume the MMF for the initial RCS flow value. The MMF reduction could affect the results of these analyses in two ways: 1) in the DNBR calculation, for which RCS flow is an important parameter, and 2) by affecting the calculated system transient.

For DNB events analyzed with the ITDP, DNBR margin has been allocated to offset the penalty associated with the MMF reduction. Thus the conclusion that the DNB design basis is met for these events remains valid. See above for details on the use of DNBR margin.

The system transient response is primarily governed by the initiating event and any subsequent control or protection system actuations. The transient results of interest would not be affected by the initial RCS flow assumption unless the reduction in flow was sufficiently large to significantly affect the steady-state core or steam generator heat transfer capability. This is not the case for the small reduction of about 1.3% being considered for Cook Nuclear Plant Unit 1. Therefore, it can be concluded that the transient results previously calculated for the events analyzed assuming MMF remain valid for the reduction in MMF to 361,600 gpm.

To confirm this conclusion, sample sensitivity cases were run for an Uncontrolled RCCA Bank Withdrawal at Power analysis. The sensitivity cases represented a range of reactivity insertion rates, varying only the initial RCS flow assumption (361,600 gpm vs. 366,400 gpm). The results showed insignificant differences in the transient results in parameters such as RCS temperatures, pressures, and time of reactor trip.

The safety analyses that assume the MMF are specifically addressed below.

Uncontrolled RCCA Bank Withdrawal at Power - FSAR 14.1.2, 1990 update

This analysis is documented in Section 3.3.4.4 of Reference 2. DNB margin has been allocated to offset the penalty associated with the MMF reduction. The Overtemperature ΔT reactor trip setpoint is not changed. In addition, the calculated system transient would not be affected by the relatively small flow reduction. The results and conclusions of the previously applicable safety analysis remain valid.

Rod Cluster Control Assembly Misalignment - FSAR 14.1.3, 1990 update

This analysis is documented in Section 3.3.4.5 of Reference 2. DNB margin has been allocated to offset the penalty associated with the MMF reduction, such that the dropped rod limit lines used in the cycle specific DNB evaluation remain valid. In addition, the calculated system transient would not be affected by the relatively small flow reduction. The results and conclusions of the previously applicable safety analysis remain valid.

Loss of Reactor Coolant Flow - FSAR 14.1.6, 1990 update

The loss of flow and the locked rotor rods-in-DNB analyses are documented in Section 3.3.4.7 of Reference 2. The normalized flow coastdown is determined by the pump characteristics and not the initial flow assumption. The system transient is unaffected by this relatively small reduction in MMF. DNB margin has been allocated to offset the penalty associated with the MMF reduction, such that the calculated DNBR at the limiting point in the transient is unaffected. The results and conclusions of the previously applicable safety analysis remain valid.

<u>Loss of External Electrical Load - FSAR 14.1.8, 1990 update</u>

This analysis is documented in Section 3.3.4.8 of Reference 2. DNB margin has been allocated to offset the penalty associated with the MMF reduction. In addition, the calculated system transient would not be affected by the relatively small flow reduction. The results and conclusions of the previously applicable safety analysis remain valid.

<u>Excessive Heat Removal Due to Feedwater System Malfunctions - FSAR 14.1.10, 1990 update</u>

This analysis is documented in Section 3.3.4.10 of Reference 2. DNB margin has been allocated to offset the penalty associated with the MMF reduction. In addition, the calculated system transient would not be affected by the relatively small flow reduction. The results and conclusions of the previously applicable safety analysis remain valid. Note that the zero power case was recently reanalyzed. The analysis documented in Reference 4 is applicable to Cook Nuclear Plant Unit 1. The system transient analysis used to calculate the maximum reactivity insertion rate would not be significantly affected by small changes in the initial RCS flow. The DNB evaluation for the zero power case did not use the ITDP.

<u>Excessive Increase in Secondary Steam Flow - FSAR 14.1.11, 1990 update</u>

This analysis is documented in Section 3.3.4.11 of Reference 2. DNB margin has been allocated to offset the penalty associated with the MMF reduction. In addition, the calculated system transient would not be affected by the relatively small flow reduction. The results and conclusions of the previously applicable safety analysis remain valid.

CONCLUSIONS

The Cook Nuclear Plant Unit 1 safety analyses and NSSS design calculations have been evaluated for the reduction in the RCS MMF given in the Technical Specifications from 366,400 gpm to 361,600 gpm. It has been determined that the conclusions of the safety analyses remain valid for the reduction in the RCS flow and the NSSS design calculations are unaffected by the reduction in the MMF requirement.

Based on the above conclusions, the reduction in the MMF requirement does not involve an increase in the probability or consequences of an accident previously evaluated or involve a reduction in a margin of safety.

The MMF requirement reduction is an operational relaxation, not a plant hardware modification. As concluded above, the MMF requirement reduction does not affect the NSSS design calculations. Therefore, the MMF requirement reduction does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Based on the above, the proposed Cook Nuclear Plant Unit 1 Technical Specification changes described below related to the MMF requirement reduction do not involve a significant hazards consideration.

RECOMMENDED TECHNICAL SPECIFICATION CHANGES

1. Safety Limit 2.1.1 - The Reactor Core Safety Limit Figure (2.1-1) was revised to reflect the change in Reactor Coolant System (RCS) Minimum Measured Flow.

2. Safety Limit 2.2.1 - The Reactor Trip System Instrumentation Setpoints were revised to address the change in Minimum Measured Flow [specifically, Table 2.2-1 Functional Unit 12's footnote (Loss of Flow) was revised to reflect the new Minimum Measured Flow].

3. Tech Spec 3/4.2.5 - Table 3.2-1 was revised to reflect the reduced MMF requirement.

Marked up Technical Specification pages are attached.

REFERENCES

1. "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for American Electric Power D. C. Cook Unit 1 Nuclear Power Station," WCAP-12568 (Proprietary), WCAP-12569 (Non-Proprietary), April 1990.

2. "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," WCAP-11902, October 1988. Submitted to NRC by AEP:NRC:1067, October 14, 1988.

3. "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8746-A, March 1977 (original version), September 1986 (approved version).

4. 90AE*-G-0089, W/AEP2-0083, "Reanalysis of Feedwater Flow Malfunction at Zero Power for Donald C. Cook 2 VANTAGE 5 Fuel Transition," May 17, 1990.

TABLE 1

DNB TRANSIENTS ANALYZED USING ITDP

1. Uncontrolled RCCA Withdrawal at Power
2. Rod Cluster Control Assembly Misalignment / Dropped Rod
3. Loss of Reactor Coolant Flow
4. Loss of External Electrical Load
5. Excessive Heat Removal due to Feedwater System Malfunctions
6. Excessive Increase in Secondary Steam Flow

TABLE 2

NON-DNB RELATED ACCIDENTS AND TRANSIENTS AND
DNB RELATED TRANSIENTS WHICH DO NOT USE ITDP

1. Large Break LOCA
2. Small Break LOCA
3. LOCA Hydraulic Forcing Functions
4. Post-LOCA Hot Leg Recirculation Time
5. Post-LOCA Long Term Core Cooling
6. Steam Generator Tube Rupture
7. Steamline Break Mass/Energy Releases
8. * Startup of an Inactive Loop
9. * Uncontrolled RCCA Withdrawal from a Subcritical Condition
10. Chemical and Volume Control System Malfunction
11. Locked Rotor
12. Loss of Normal Feedwater Flow
13. Loss of All AC Power to the Plant Auxiliaries
14. * Rupture of Steam Pipe
15. Rupture of Control Rod Drive Mechanism Housing
16. Containment Analyses
17. * Excessive Heat Removal due to Feedwater System Malfunctions (zero power case)

* DNB related events