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SUBJECT: Application for amends to Licenses DPR-58 & DPR-74,deleting Tech Specs 3.5.4.1 & 3.5.4.2 re boron injection sys & changing Table 3.3-5 safety injection response times to be consistent w/new analyses assumptions.

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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
TECHNICAL SPECIFICATION CHANGE REQUEST
BIT BORON CONCENTRATION REDUCTION

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

Attn: T. E. Murley

March 26, 1991

Dear Mr. Murley:

This letter and its attachments constitute an application for changes to the Technical Specifications (T/Ss) for Donald C. Cook Nuclear Plant Units 1 and 2. Specifically, we request that T/Ss 3.5.4.1 and 3.5.4.2, which state the requirements for the boron injection system (including the boron injection tank [BIT] and its associated heat tracing), be deleted. In addition, we are requesting a change to the "safety injection (ECCS)" response times specified in T/S Table 3.3-5 to be consistent with the new analyses assumptions.

Deletion of T/Ss 3.5.4.1 and 3.5.4.2 will allow removal of the BIT. As discussed in NRC Generic Letter 85-16, there have been incidents at operating reactor plants in which boric acid has crystallized in the internals of vital safety-related pumps and piping, thereby rendering those systems inoperable. The NRC Staff concluded in Generic Letter 85-16 that there are inherent safety risks in the present system of using high concentrations of boron and further indicated that, based on improved analysis methods, the BIT could be removed. Consequently, we are making this submittal to implement the changes in the T/Ss identified above.

This T/Ss change is being requested for implementation during the next regularly scheduled refueling outages. To support the schedule, we request that the amendments be approved for both units during the first half of 1992. We will keep NRC project management informed of any schedule changes through routine project review meetings.

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Attachment 1 to AEP:NRC:1140

10 CFR 50.92 Analysis for Changes
to the Donald C. Cook Nuclear Plant
Units 1 and 2 Technical Specifications

I. Introduction

As discussed in NRC Generic Letter 85-16, there have been incidents at operating plants in which boric acid has crystallized in the internals of vital safety-related pumps and piping, thereby rendering those systems inoperable. Heat tracing is presently necessary to maintain the boron injection tank (BIT) and associated piping at a sufficiently high temperature to prevent precipitation of the 12% by weight boric acid solution. Furthermore, the safety-related nature of the BIT requires that the heating systems be redundant.

The required solubility temperature imposes a continuous load on the heaters, and the potential for low-temperature alarm actuation and heater burnout exists. Violation of the T/S on concentration in the BIT poses availability problems in that recovery is required within a very short time. If the concentration is not restored within one hour, the plant must be taken to the hot standby condition and borated to the equivalent of 1 percent delta k/k at 200 degrees Fahrenheit. Thus, this requirement has a potentially serious impact on plant availability. In addition, the high boric acid concentration makes recovery from a spurious safety injection signal (which results in injection of the BIT fluid into the reactor coolant system) both time consuming and costly.

Analysis for calculating the consequences of a steam line break have provided results that demonstrate a reduced need for the highly concentrated boron (20,000 ppm to 22,500 ppm) injection. Based on the inherent safety risks in the present system and the analysis results, we are making this submittal requesting the necessary T/Ss changes to support deactivation of the BIT.

Similar license amendments have been approved by the NRC for the North Anna Power Station, H. B. Robinson Steam Electric Plant, Indian Point Unit No. 2, Surry Power Station, V. C. Summer Nuclear Station, Sequoyah Nuclear Plant, and Diablo Canyon Units 1 and 2.

II. Description of Physical Modifications to the Plant

A. Current System Design Basis

The BIT holds 900 gallons of water with a boron concentration in excess of 20,000 ppm, approximately a 12 weight percent solution. Tank heaters and pipe heat tracing are provided to maintain a minimum solution temperature of greater than or equal to 135°F. Recirculation from the boric acid storage tanks (BASTs) to the BIT is maintained continuously via boric acid transfer pumps to ensure that the BIT is full of concentrated boric acid at all times and to prevent boric acid stratification. The BIT is isolated from the reactor coolant system (RCS) during normal plant operation.

During a safety injection, the suction of the charging pumps is diverted from the normal suction at the volume control tank (VCT) to the refueling water storage tank (RWST) and discharged through the BIT to the RCS. Concurrently, isolation valves in the recirculation line to the BAST close.

The operability of the boron injection system currently ensures that sufficient negative reactivity is injected into the core to offset the increase in positive reactivity caused by RCS cooldown due to a steam line break.

B. <u>Design Changes to the Boron Injection System</u>

The BIT will remain in place connected to the ECCS injection piping system, but filled with water containing only a nominal boron concentration instead of 12 percent boric acid by weight. The BIT recirculation piping to/from the BAST and attendant equipment (piping, valves, instruments, heat tracing, etc.) will generally remain in place, but will be disconnected at appropriate locations. All controls that are deactivated will be removed to lessen distractions and make additional panel space available.

An alternate solution considered was the entire removal of the tanks, piping, heat tracing, etc. Due to the obvious cost savings in labor and radiation exposure between removing and disposing of these contaminated components versus abandoning in place, the practical course was determined to be disconnecting the BITs and their ancillary components from interfacing systems. Furthermore, abandoning in place allows for easy restitution if a future need arises to return the BITs to service.

The following system design changes will be made to the boron injection system:

1. Piping Modifications

   The BIT/BAST recirculation piping and the BIT flushing lines will be cut and capped to isolate the 12% boric acid system from the BIT.

2. Valve Impacts

   (a) BIT/BAST recirculation valves will remain in place with their air supplies disconnected (these are fail-closed valves). The control switches for these valves will be removed from the main control room panels.

   (b) Manual valves in the BIT recirculation path will remain in place, in the closed position.

3. Instrumentation Impacts

   (a) Flow instruments 1- and 2-IFA-250 will remain in place with their alarm wiring disconnected. The main control room annunciator windows, which warn the operator of low BIT recirculation flow, will be "blanked" and made available as spares.

   (b) The wiring for BIT temperature alarms will be disconnected and their annunciator windows, warning the operator of low BIT temperature, will be "blanked" and made available as spares.

4. Heat Tracing

   (a) Heat tracing circuits applicable to the BIT recirculation piping will be de-energized and abandoned in place. The thermostats/alarmstats will be disconnected and removed. Annunciators applicable to these heat trace circuits will be "blanked" and made available as spares.

   (b) The BIT strip heaters will be de-energized and abandoned in place. The temperature controllers will be de-terminated and removed.

III. Description of Proposed Technical Specification Changes

A. T/S Index page VII

Section 3/4.5.4 is being deleted and the index therefore indicates that pages 3/4 5-9 and 3/4 5-10 are intentionally left blank.

B. T/S Index page XIII

Delete section 3/4.5.4 (Boron Injection System) from the bases.

C. Table 3.3-2

Table 3.3-2 is being changed to reflect the overpower delta T reactor trip response time supported by the analysis of steam line break mass and energy release outside containment. In addition, for Unit 1, the orientation of the second page of the table is being changed to make the pages in the table consistent.

D. Table 3.3-5, Engineered Safety Features Response Times

Additional notes were added to Table 3.3-5 to address the response time for charging pump suction switchover from the VCT to the RWST. The note reference for function 3g was changed from an "*" to a "++" to be consistent with the note changes described. However, it should be noted that the response of essential service water is not in any way associated with the alignment of the ECCS system for safety injection. The time responses for the essential service water included in Table 3.3-5 reflect the response time with and without offsite power available.

Table 3.3-5 engineered safety feature (ESF) actuation system response times were increased to reflect the safety analyses supporting the elimination of the BIT. The reactor trip on SI response time for all ESF trips has been increased from 2.0 to 3.0 seconds. This increase is supported by the safety analyses.

In addition, we also wrote out "less than or equal to" where the symbol was previously used.

E. T/S 3/4 5.4, Boron Injection System

Technical Specifications 3.5.4.1 and 3.5.4.2 covering the BIT and its associated heat tracing will be deleted. These pages will be left intentionally blank.

F. Unit 1 T/S 5.5, Design Features/ECCS

This T/S states that the ECCS shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR. An exception statement will have to be added to note the reduction in the boron concentration of the BIT.

G. Bases Section 2.2.1

The discussion in this section pertaining to overpower delta T is being revised to reflect the new analysis.

H. Bases Sections 3/4.3.1 and 3/4.3.2, Protective and Engineered Safety Features (ESF) Instrumentation

A paragraph is being added to this section that discusses the changes being made to Table 3.3-5. The addition of this paragraph is resulting in a large amount of text being shifted on the pages that follow in Bases Section 3/4.3. Consequently, we are submitting those pages as well.

I. Bases Section 3/4.5.4, Boron Injection System

This section describes the basis for T/S 3/4.5.4 noted above and therefore will also be deleted.

J. Bases Section 3/4.5.5, Refueling Water Storage Tank

This section of the bases is being revised to reflect the new analysis.

IV. Justification for Change

The only accident analyses that are significantly affected by boron reduction and deactivation of the BIT are the steam line break transients. These transients are affected with respect to both core integrity and mass and energy releases inside and outside of containment. The analyses for most of these transients have previously been submitted to the NRC and are summarized below.

It should be noted that the assumption of 0 ppm boron concentration is conservative. In reality, the BIT will be filled with water that has a nominal boron concentration.

A. Steam Line Break Core Response

The reanalysis of this event with a BIT boron concentration of 0 ppm for Unit 1 is documented in WCAP-11902, Section 3.3.4.13, which was previously submitted in our letter AEP:NRC:1067 dated October 14,

1988. The NRC Safety Evaluation Report dated June 9, 1989, for Amendment No. 126 to Facility Operating License No. DPR-58 (Unit 1) from Mr. J. F. Stang to Mr. M. P. Alexich provided NRC approval of this analysis. This analysis has been incorporated into Chapter 14 of the Updated Final Safety Analysis Report (UFSAR).

The reanalysis of this event for Unit 2 with a BIT boron concentration of 0 ppm is documented in Attachment 4, Appendix B of our letter AEP:NRC:1071E dated February 6, 1990. The NRC Safety Evaluation Report dated August 27, 1990, for Amendment Nos. 148 and 134 to Facility Operating License Nos. DPR-58 and DPR-74 (Units 1 and 2, respectively) from Mr. T. G. Colburn to Mr. M. P. Alexich provided the NRC approval of this analysis.

B. <u>Steam Line Break Mass and Energy Inside Containment</u>

The reanalysis of this event with a BIT boron concentration of 0 ppm for both Units 1 and 2 is documented in WCAP-11902, Supplement 1, Section S-3.3.4.1 (Attachment 4 to this letter). This analysis provides a series of mass and energy release rates which are evaluated for containment integrity in Item D. This section of WCAP-11902, Supplement 1 was previously submitted in Attachment 5 to our letter AEP:NRC:1071E dated February 6, 1990. The NRC Safety Evaluation Report dated August 27, 1990, for Amendment Nos. 148 and 134 to Facility Operating License Nos. DPR-58 and DPR-74 (Units 1 and 2, respectively) from T. G. Colburn to Mr. M. P. Alexich provided the NRC approval of this analysis. Attachment 4 to this letter contains a complete copy of WCAP-11902, Supplement 1 entitled "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report" for your review.

C. <u>Steam Line Break Mass and Energy Outside Containment</u>

The reanalysis of this event with a BIT boron concentration of 0 ppm for both Units 1 and 2 is documented in WCAP-11902, Supplement 1, Section S-3.3.4 (Attachment 4 to this letter). As discussed in WCAP-11902, Supplement 1, Westinghouse regenerated the outside containment mass and energy releases for 68 separate cases considering different break sizes, units, initial power levels, and auxiliary feedwater flow rates. The evaluation of the affect of these mass and energy release rates on equipment outside of containment is summarized in Item E.

D. Steam Line Break Containment Integrity

The reanalysis of this event with a BIT boron concentration of 0 ppm for both Units 1 and 2 is documented in WCAP-11902, Supplement 1, Section 3.4.2.1 (see Attachment 3 to this letter). This section of WCAP-11902, Supplement 1 was previously submitted in Attachment 5 to our letter AEP:NRC:1071E dated February 6, 1990. The NRC Safety Evaluation Report dated August 27, 1990, for Amendment Nos. 148 and 134 to Facility Operating License Nos. DPR-58 and DPR-74 (Units 1 and 2, respectively) from Mr. T. G. Colburn to Mr. M. P. Alexich provided the NRC approval of this analysis.

E. Steam Line Break Outside of Containment

The reanalysis of this event supporting a BIT boron concentration of 0 ppm and associated response timing changes for both Units 1 and 2 is documented in Attachment 5. This analysis utilizes the spectrum of mass and energy releases summarized in Item C to develop the temperature response of the main steam enclosures as well as important equipment within those enclosures. The limiting case was determined to be a Unit 2 break of 1.2 square feet from 70% initial power. For this case, as discussed in Attachment 5, the instrument surface temperatures were determined to remain below the qualification temperatures. Therefore, reducing the BIT concentration to 0 ppm is acceptable.

V. No Significant Hazards Analysis

We believe that operating with the BIT boron concentration requirement eliminated will not adversely impact public health and safety.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, the 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 an accident previously analyzed or evaluated, or

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

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

Criterion 1

The deactivation of the BIT affects the steam line break transients with respect to core integrity, mass and energy release to containment, and mass and energy release outside containment for instrument qualification. With the assumption that the BIT remains installed without heat tracing and with boric acid concentration reduced to 0 ppm, analyses show that the departure from nucleate boiling design basis is met and no consequential fuel failures are anticipated. Additionally, temperatures and pressures reached in containment would be below the containment design limits. All instrument surface temperatures remain below the qualification temperatures. Therefore, the equipment inside and outside containment necessary to mitigate the consequences of an accident would function as intended. Therefore, no significant increase in the probability or consequences of a previously analyzed accident would occur.

Criterion 2

The BIT is a component of the safety injection system whose sole function is to provide concentrated boric acid to the reactor coolant to mitigate the consequences of a postulated steam line break. The deactivation of the BIT will therefore affect the steam line break transients, but will not create the possibility of a new or different type of accident.

Criterion 3

The analyses performed for the deactivation of the BIT indicate that the departure from nucleate boiling design basis continues to be met. Additionally, the temperatures and pressures reached in containment would fall below the containment design limits. Finally, in Attachment 5 we have shown that the impact on instrumentation of mass and energy release outside containment meets acceptance criteria. Since the design bases contain the required margins of safety, no significant reductions in margins of safety will occur.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14780) of amendments considered not likely to involve significant hazards considerations. The sixth of these examples refers to changes that either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within limits established as acceptable.

The analyses that are referenced in this submittal have been demonstrated to comply with the licensing basis of the plant. In fact, all but the mass and energy release outside containment analysis and its impact on instrumentation have already received NRC approval in SERs previously issued to Cook Nuclear Plant. Thus, we believe the example cited is applicable and that the changes should not involve significant hazards consideration.

The requested T/Ss changes are supported by the attachments to this letter. Attachment 1 contains a description of the changes and our evaluation concerning significant hazards considerations. The proposed revised T/Ss pages are included in Attachment 2. Attachment 3 consists of the existing T/Ss pages marked-up to reflect the changes proposed in this amendment request. Attachment 4 is a copy of WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report." Attachment 5 contains an analysis entitled "Main Steam Line Break Outside Containment Analysis Summary."

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 amounts of any effluent that may be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

The proposed changes have been reviewed 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 Mr. J. R. Padgett of the Michigan Public Service Commission and to the Michigan Department of Public Health.

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

Sincerely,

M. P. Alexich
Vice President

ldp

Attachments

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
J. R. Padgett
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