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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Second suppl to 820407 application for Cycle 4 reload & uprate license amend.Further changes to App A Tech Specs necessary to allow Cycle 4 operation w/Exxon Nuclear Co fuel at 3,411 MWt.

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September 30, 1982
AEP:NRC:0637E

Donald C. Cook Nuclear Plant, Unit No. 2
Docket No. 50-316
License No. DPR-74
UNIT 2 CYCLE 4 TECHNICAL SPECIFICATIONS CHANGE REQUEST

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

Dear Mr. Denton:

This letter and its Attachments constitute the second supplement to the "Application for Unit 2, Cycle 4 Reload and Uprate License Amendment" contained in our letter No. AEP:NRC:0637A dated April 7, 1982. The first supplement to our application is contained in letter No. AEP:NRC:0637B, dated July 8, 1982.

This letter requests further changes to the Unit 2, Appendix A Technical Specifications (T/S). These changes are necessary to allow Cycle 4 operation with Exxon Nuclear Company (ENC) fuel at the slightly higher reactor thermal power of 3411 MWt.

Two Attachments are included in this letter in support of our application. Attachment No. 1 contains a description of the proposed T/S changes. Attachment No. 2 contains the revised T/S pages. All changes in Attachment No. 2 are indicated by a vertical line on the right-hand side of the page. The T/S changes contained in Attachment No. 2 to this letter have been reviewed and approved by the Plant Nuclear Safety Review Committee (PNSRC) and by the AEPSC Nuclear Safety and Design Review Committee (NSDRC).

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As a result of an issue raised by another fuel supplier in 1981, an additional LOCA analysis is being performed with ENC's EXEM/PWR evaluation model to calculate peak cladding temperature with maximum Emergency Core Cooling System (ECCS) safeguards in operation. As indicated in our letter No. AEP:NRC:0637D, dated September 8, 1982, the results of this additional analysis are scheduled to be submitted to the NRC Staff by November 1, 1982. ENC has informed us that certain T/S values will change as a result of this additional analysis. In the interest of avoiding a duplication of effort, those T/S changes which are dependent on the ECCS reanalysis results are not presented here.

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Upon completion of the ECCS reanalysis, the final version of the affected T/S will be submitted.

Payment for NRC processing of the aforementioned requests has previously accompanied our "Application for Cycle 4 Reload and Uprate License Amendment" contained in letter No. AEP:NRC:0637A wherein we interpreted this application for a license amendment to constitute a Class IV Amendment as defined in 10 CFR 170.22.

The T/S changes requested in Attachments No. 3, 4 and 6 to our letter No. AEP:NRC:0637A, the requests contained in this letter, and the requests to be submitted once the ECCS reanalysis described above is completed, constitute all the T/S changes necessary for the operation of Unit 2 during Cycle 4. Please note that the T/S change requests contained in Attachments No. 3 and 4 to AEP:NRC:0637A are still outstanding.

Our review of the Unit 2, Appendix B T/S has concluded that no changes need to be submitted for them.

In our letter No. AEP:NRC:0637B dated July 8, 1982, we informed you that ENC had documented the results of their ECCS analysis as it relates to ENC fuel, and of the more limiting transients for the Cook Plant, in Reports No. XN-NF-82-35 ("Donald C. Cook Unit 2 LOCA ECCS Analysis Using EXEM/PWR Large Break Results") and No. XN-NF-82-32 ("Plant Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3425 MWt"), respectively. ENC has transmitted to the NRC twenty-five (25) copies of each of these reports along with twenty-five (25) copies of Report No. XN-NF-82-37, entitled "D. C. Cook Unit 2, Cycle 4 Safety Analysis Report" under separate cover. In addition, ENC has recently issued Supplement 1 to Report No. XN-NF-82-37 and has transmitted twenty-five (25) copies of this supplement under separate cover to the NRC. We request that you incorporate all of the above-mentioned reports onto Docket No. 50-316 for Unit No. 2 of the Donald C. Cook Nuclear Plant.

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,

R. S. Hunter
Vice President

/os
cc:(attached)

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cc: John E. Dolan - Columbus
M. P. Alexich
R. W. Jurgensen
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
Joe Williams, Jr.
NRC Resident Inspector at Cook Plant - Bridgman

ATTACHMENT NO. 1 TO AEP:NRC:0637E
DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2
DESCRIPTION OF UNIT 2 CYCLE 4 TECHNICAL SPECIFICATION
CHANGE REQUESTS

Change No. 1

Pages 2-2 and 2-3; Figures 2.1-1 and 2.1-2

Core safety limit lines for the D. C. Cook Unit 2 reactor have been regenerated to include the effects of Exxon Nuclear Company's (ENC) XNB CHF correlation and a $F^N_{\Delta H}$ limit of 1.60. The safety limit lines define regions of reactor operation in terms of the system pressure, vessel average coolant temperature, and core power level. Operation within the safety limit lines ensures that the vessel hot leg temperature will be less than the saturation temperature and that the minimum departure from nucleate boiling ratio (MDNBR) will be greater than the safety limit value of 1.17 as calculated with the XNB critical heat flux correlation. These core safety limit lines are shown in Figure 2.1-1 for four-loop operation and in Figure 2.1-2 for three-loop operation.

It should be noted that the Core Safety Limit lines presented herein reflect the change in calculated DNB margin resulting from use of a measured non-mixing vane spacer loss coefficient in the XNB critical heat flux calculation. Previous ENC-generated Core Safety Limit lines, such as those presented in ENC Report No. XN-NF-82-32(P) entitled "Plant Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3425 MWt", employed a loss coefficient calculated from the generalized de Stordeur equation.

Change No. 2

Page 2-8; Table 2.2-1

The analyzed increase in $F^N_{\Delta H}$ and the relative thermal margin deficit for ENC fuel require that the Overtemperature ΔT trip setpoint be reduced. The trip setpoint equation is identical to that currently given in the T/S. However, certain parameters which appear in this equation are required to be changed. The bias constant K_1 in the trip setpoint equation for 4-loop operation has been reduced in magnitude by the equivalent of 4% in power. The pressure gain K_3 for both 3-loop and 4-loop operation has also been adjusted to better correlate the necessary pressure dependence of the trip setpoint. The change to the limitation on indicated T_{AVG} at Rated Thermal Power has previously been submitted.

The Overtemperature ΔT trip setpoint employed in the Plant Transient Safety (PTS) analysis conservatively represents the recommended changes. The validity of the recommended setpoint equation for 4-loop operation has been further tested with steady state thermal margin calculations over the temperature, pressure, and power range of interest. Both the PTS analysis reported in XN-NF-82-32(P), "Plant

Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3425 MWt", and the steady state MDNBR calculations, fully support the inclusion of the recommended 4-loop Overtemperature ΔT trip equation in the reactor protection system.

The gain constant in the Overtemperature ΔT trip equation for 3-loop operation is the same as that in the 4-loop trip equation. The current T/S bias constant K_1 for 3-loop operation has been retained. The adequacy of the 3-loop Overtemperature ΔT trip equation, incorporating the new gain constant, has been tested with steady state thermal margin calculations over the temperature, pressure, and power ranges of interest. The results of these test calculations confirm the conservatism of the Overtemperature ΔT trip equation for 3-loop operation.

Change No. 3

Page B 2-1; Specification 2.1.1

The basis for reactor Core Safety Limits has been revised to reflect ENC's thermal margin methodology. The use of the XNB critical heat flux correlation and its associated safety limit is specified. The fourth paragraph has been deleted, since ENC has not employed a statistical combination of uncertainties in its thermal margin analyses for D. C. Cook Unit 2. Appropriate uncertainties in the thermal margin analysis have been treated deterministically.

ATTACHMENT NO. 2 TO AEP:NRC:0637E
DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2
REVISED TECHNICAL SPECIFICATION PAGES

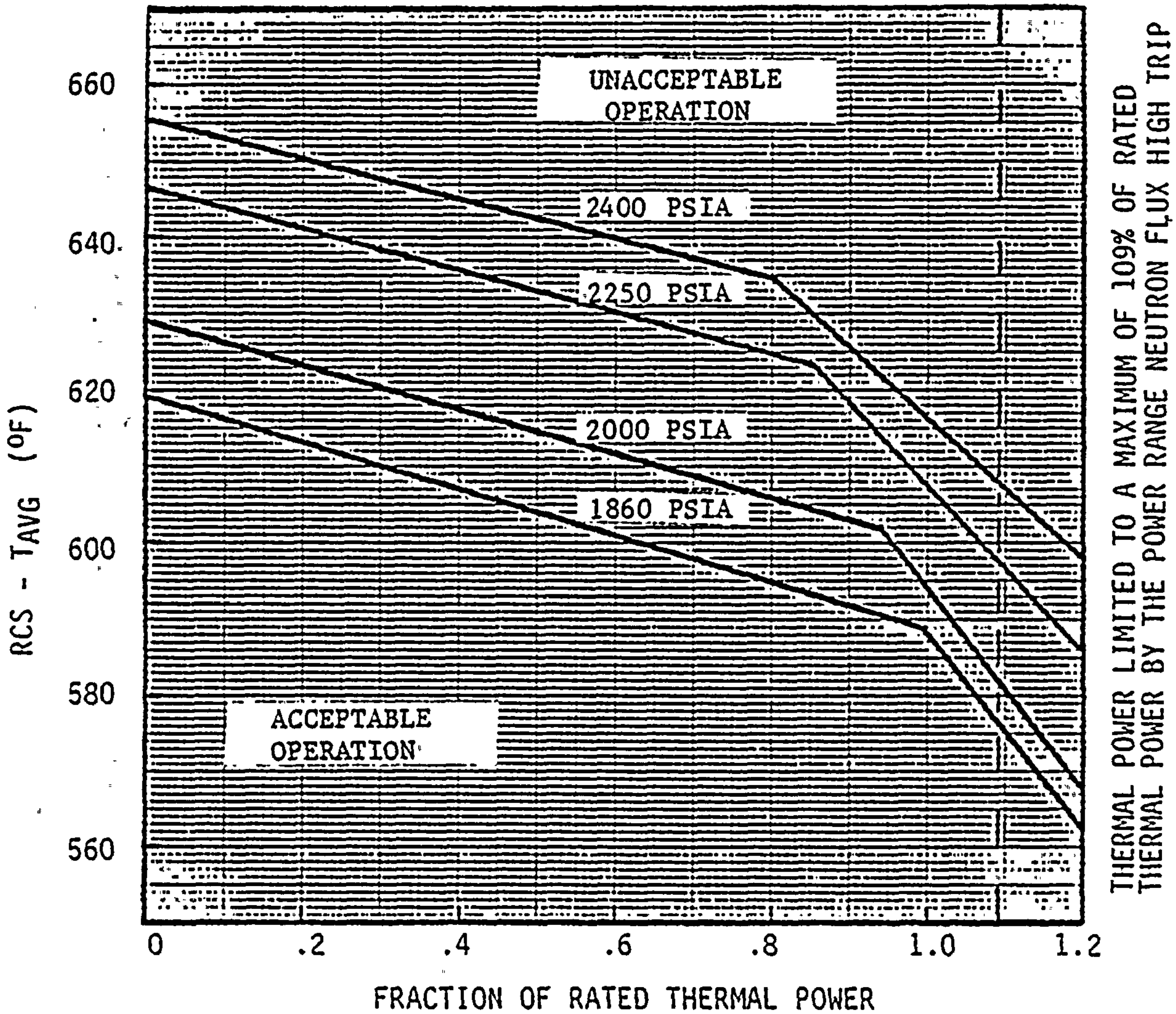

Figure 2.1-1 Reactor Core Safety Limits - Four Loops in Operation

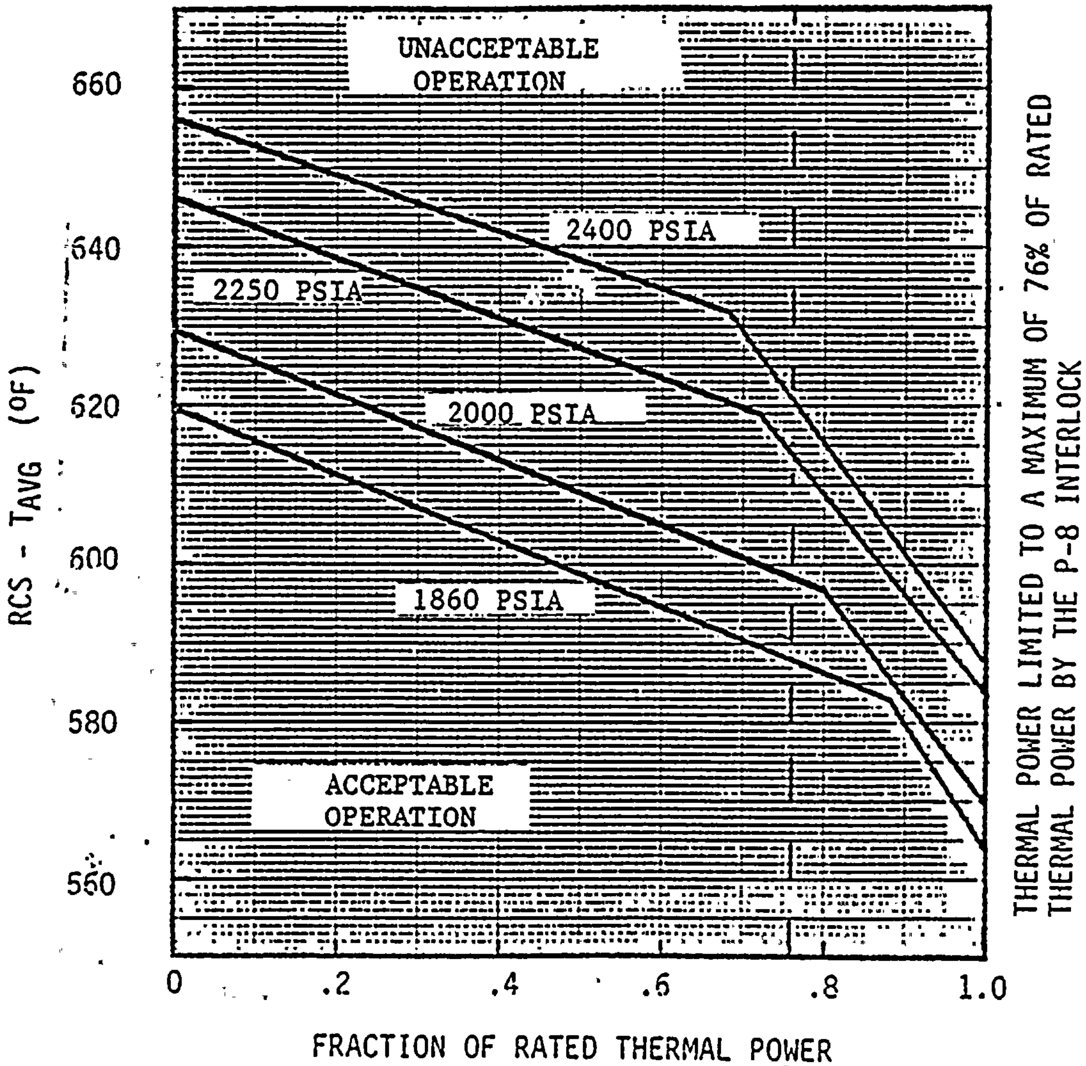

Figure 2.1-2 Reactor Core Safety Limit - Three Loops in Operation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops	Operation with 3 Loops
$K_1 = 1.267$	$K_1 = 1.116$
$K_2 = 0.01607$	$K_2 = 0.01607$
$K_3 = 0.000926$	$K_3 = 0.000926$

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 40 percent and + 3 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds - 40 percent, the ΔT trip setpoint shall be automatically reduced by 1.8 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.2 percent of its value at RATED THERMAL POWER.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the XNB correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the correlation DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Uncertainties in primary system pressure, core temperature, core thermal power, primary coolant flow rate, and fuel fabrication tolerances have been included in the analyses from which Figures 2.1-1 and 2.1-2 are derived.