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JUL 2 1 1981

Docket No. 50-315

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
Post Office Box 18
Bowling Green Station
New York, New York 10004

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 26, 1981.

This amendment revises the F_Q peaking factor limit.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

cp

Enclosures:
1. Amendment No. 48 to DPR-58
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

OFFICE	ORB#1:DL	ORB#1:DL	C-ORB#1:DL	AD:OR:DL	OELD		
SURNAME	CParrish;ds	SMiner	SVarga	TNovak	RBlack		
DATE	6/30/81	7/7/81	7/ /81	7/ /81	7/16/81		

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated March 26, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 21, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 2-3	3/4 2-3
3/4 2-6	3/4 2-6
3/4 2-15	3/4 2-15
3/4 2-16	3/4 2-16
3/4 2-17	3/4 2-17
3/4 2-19	3/4 2-19
3/4 2-20	3/4 2-20
*B3/4 2-1	*B3/4 2-1
B3/4 2-2	B3/4 2-2

*Included for convenience

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The indicated AFD shall be considered outside of its $\pm$ 5% target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the $\pm$ 5% target band shall be accumulated on a time basis of:

a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. A penalty deviation of one-half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target axial flux difference of each OPERABLE excore channel shall be determined in conjunction with the measurement of $F_Q^M(z)$ as defined in Specification 4.2.2.2.C. The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z,\ell)$ shall be determined to be within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z,\ell)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship at the time of the target flux determination.

$$F_Q^M(Z) \leq \frac{F_Q^L(Z)}{PxEp(Z)} [K(Z)]/[V(Z)] \qquad \text{for} \quad P > .5$$

$$F_Q^M(Z) \leq \frac{2F_Q^L(Z)}{Ep(Z)} [K(Z)]/[V(Z)] \qquad \text{for} \quad P \leq .5$$

where: $F_Q^M(Z) = F_Q(Z,\ell)$ at ℓ for which $\frac{F_Q(Z,\ell)}{T(E)}$ is a maximum

$F_Q^L(Z) = F_Q^L(E_\ell)$ at ℓ for which $\frac{F_Q(Z,\ell)}{T(E)}$ is a maximum

$F_Q^M(Z)$ and $F_Q^L(Z)$ are functions of core height, Z, and correspond at each Z to the rod ℓ for which $\frac{F_Q(Z,\ell)}{T(E\ell)}$ is a maximum at that Z.

V(Z) is the function defined in Figure 3.2-3, K(Z) is defined in Figure 3.2-2, T(Eℓ) is defined in Figures 3.2-3a and 3.2-3b, P is the fraction of RATED THERMAL POWER. Ep(Z) is an uncertainty factor to account for the reduction in the $F_Q^L(E\ell)$ curve due to an accumulation of exposure prior to the next flux map.

$$E_p\,(Z) = 1.0 \qquad 0 \leq E\ell \leq 12.0$$

$$E_p\,(Z) = 1.0 + [.0039XF_Q^M(Z)] \qquad 12.0 \leq E\ell \leq 34.5$$

$$E_p\,(Z) = 1.0 + [.0085XF_Q^M(Z)] \qquad 34.5 \leq E\ell \leq 42.2$$

d. Measuring $F_Q^P(Z,\ell)$ in conjunction with a target flux difference determination, according to the following schedule:

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.10]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)\ F_p}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns at 100% or APL (whichever is less) of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi\ell}^{Meas} / T(E\ell)}{[F_{ij}(Z)]_{Max}}$$

R_{ij} and its associated σ_j may be calculated on a full core or a limiting fuel batch basis as defined on page B3/4 3-3 of basis.

e. $F_{Qi\ell}^{Meas}$ is the limiting total peaking factor in flux map i. The limiting total peaking factor is that factor with least margin to the $F_Q^L(E\ell)$ curve defined in Figure 3.2-3a for Exxon Nuclear Company fuel and in Figure 3.2-3b for Westinghouse fuel.

LIMITING CONDITION FOR OPERATION (Continued)

$T(E\ell)$ is the ratio of the exposure dependent $F_Q^L(E)$ to 2.10 and is defined in Figure 3.2-3a for fuel supplied by Exxon Nuclear Company and in Figure 3.2-3b for fuel supplied by Westinghouse Electric Corporation.

f. $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a limiting total measured peaking factor without uncertainties or densification allowance of $F_{Qi\ell}^{Meas}$.

σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1}\sum_{i=1}^{n}(\overline{R}_j - R_{ij})^2\right]^{1/2}}{\overline{R}_j}$$

The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system respectively.

The factor 1.03 is the engineering uncertainty factor.

g. F_p is an uncertainty factor for Exxon fuel to account for the reduction in the $F_Q^L(E)$ curve due to an accumulation of exposure prior to the next flux map. The following F_p factor shall apply:

$F_p = 1.0$	$0 \leq E\ell \leq 12$
$F_p = 1.0 + [.0015 \text{ X } W]$	$12 < E\ell \leq 34.5$
$F_p = 1.0 + [0.0030 \times W]$	$34.5 \leq E\ell \leq 42.2$ where W is the number of effective full power weeks (rounded up to the next highest integer) since the last full core flux map.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: Mode #1 above the percent of RATED THERMAL POWER indicated by the relationship.

$$APL = \min \text{ over } Z \text{ of } \frac{F_Q^L(E_\ell)\ K(Z)}{F_Q(Z,\ell) \times V(Z) \times Ep\ (Z)} \times 100\% \qquad P > .5$$

where $F_Q(Z,\ell)$ is the measured $F_Q(Z,\ell)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty, at the time of target flux determination from a power distribution map using the movable incore detector.

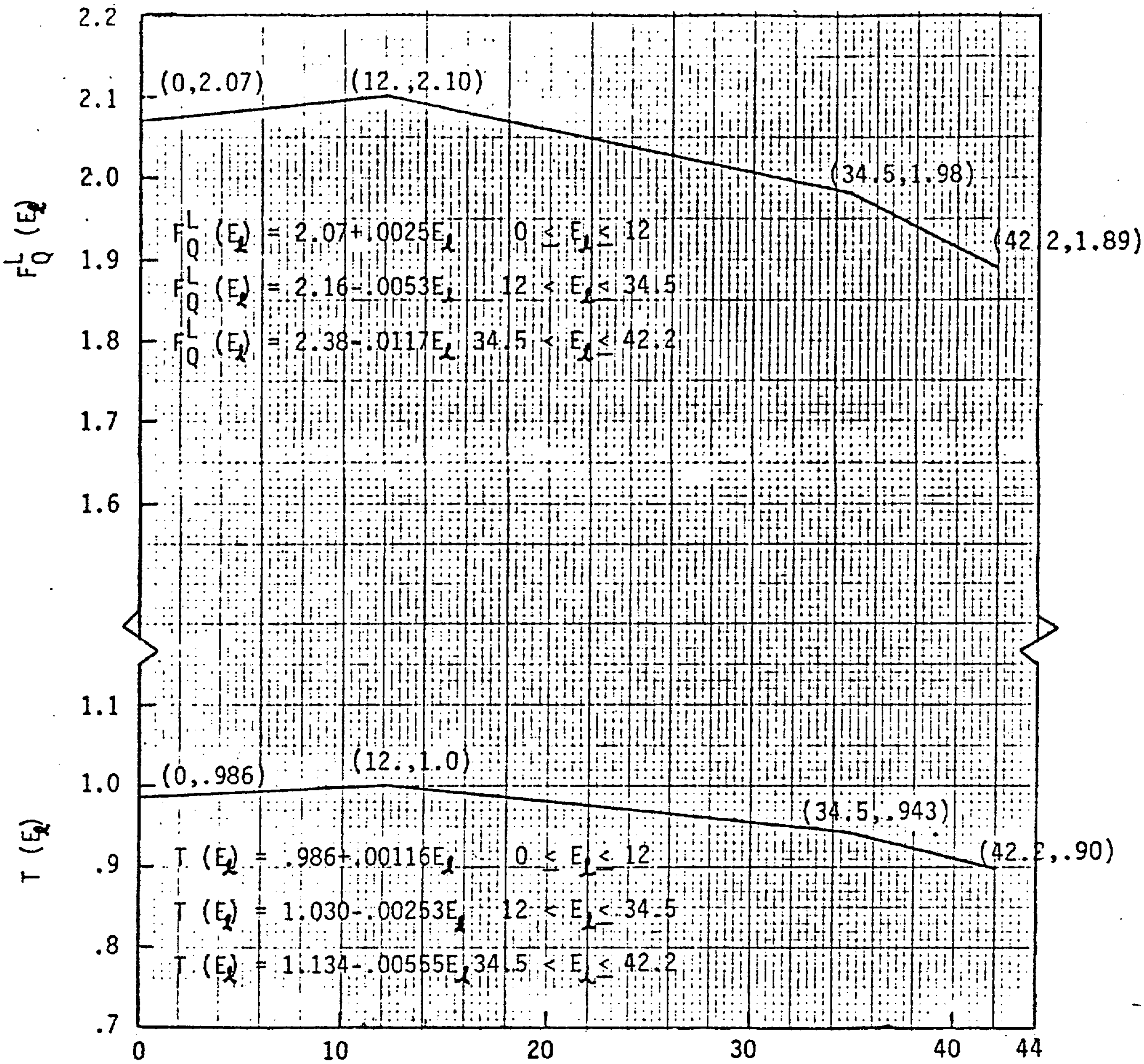
(1) Lower core region 0 to 10% inclusive
(2) Upper core region 90% to 100% inclusive

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER 1 percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next 2 hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to APL or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes.

The APDMS may be out of service: 1) when incore maps are being taken as part of the Augmented Startup Test Program or 2) when surveillance for determining power distribution maps is being performed.



Peak Pellet Exposure In MWD/KG

Figure 3.2.3a

Exposure Dependent F_Q Limit, $F_Q^L(E_\ell)$, and Normalized Limit $T(E_\ell)$ as a Function of Peak Pellet Burnup for Exxon Nuclear Company Fuel.

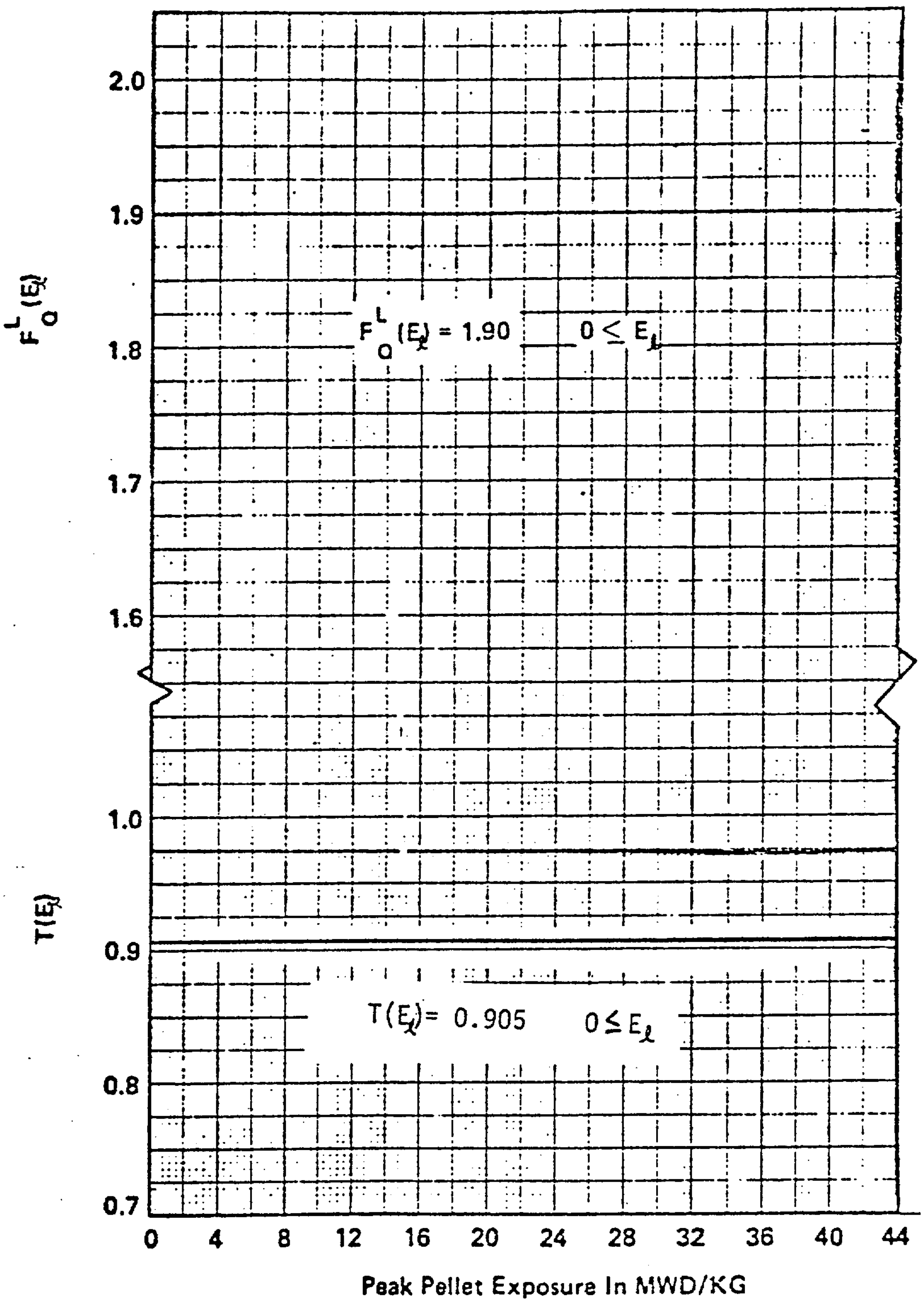


Figure 3.2 - 3b

F_Q Limit, $F_Q^L(E_\ell)$, and Normalized Limit $T(E_\ell)$ as a Function of Peak Pellet Burnup for Westinghouse Fuel

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z,\ell)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels above 50% of RATED THERMAL POWER. For THERMAL POWER levels below 50% of RATED THERMAL POWER, deviations the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER. During operation at THERMAL POWER levels between 15% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

The upper bound limit, 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, on AXIAL FLUX DIFFERENCE assures that the $F_Q(Z,\ell)$ envelope of 2.10 times K(Z) x T(Eℓ) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The lower bound limit (50% of RATED THERMAL POWER) is based on the fact that at THERMAL POWER levels below 50% of RATED THERMAL POWER, the average linear heat generation rate is half of its nominal operating value and below that value, perturbations in localized flux distributions cannot affect the results of ECCS or DNBR analyses in a manner which would adversely affect the health and safety of the public.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

The bases and methodology for establishing these limits is presented in topical report XN-NF-77-57. "Exxon Nuclear Power Distribution Control for PWR's-Phase II" and Supplement 1 to that report.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-315

Introduction

In a letter dated March 26, 1981 to H. R. Denton, Indiana and Michigan Electric Company (I&MEC) requested changes to the D. C. Cook Nuclear Plant, Unit No. 1 Technical Specification to increase the F_Q core peaking factor limit. Enclosed with the letter was EXON Nuclear Company's (ENC) report #XN-NF-81-07 which provides their ECCS reanalysis for D. C. Cook Unit No. 1 using their recently approved ECCS Evaluation Model, WREM-11A (PWR).

A. Summary of ENC Report XN-NF-81-07

This report presents results of a reanalysis of the most limiting large break LOCA identified in the previous D. C. Cook Unit 1 ECCS Analysis[1] using the Exxon Nuclear Company's (ENC) the recently approved[2] ENC-WREM-IIA evaluation model.[3] The previous analysis was performed using ENC's predecessor version of ENC-WREM IIA, namely; ENC-WREM II, so that differences between the two analyses should reflect only changes made in updating the evaluation model, plant design parameters, or data corrections.

The ENC-WREM-IIA model consists of: the blowdown code RELAP4-EM/ENC28,[4] the containment code ICECON[5], the reflood code REFLEX,[3] and the hot channel heatup code TOODEE 2[6]. In addition to minor changes in the blowdown, and heatup code, the principal changes to the earlier version of the evaluation model has been the replacement of NSSS supplier containment pressure transients with ENC's ICECON code for calculation of containment back pressure during reflood, and the replacement of the RELAP 4-EM/FLOOD code with ENC's REFLEX code for the reflood transient calculation.

The most limiting break in terms of peak clad temperature (PCT) determined in the previous ENC analysis was the double-ended equivalent cold-leg split (DECLS) break assumed to occur between the primary loop pump discharge and the reactor vessel. Results of the reanalysis for this break for beginning-of-life (BOL) fuel conditions are presented in the report, and show that a PCT of 2199°F is predicted for this break when operating with a total power peaking factor (F_Q) of 2.07.

The effects of fuel burnup on the allowable F_Q over fuel life was also analyzed to provide the basis for operating technical specifications. The influence of fuel burnup on allowable F_Q was determined by recomputing the hot channel response during blowdown using the hot channel model in RELAP4-EM/ENC28 with revised fuel element parameters for various burnup states determined by the GAPEX[7] code. Thermodynamic boundary conditions used for the core in the hot channel calculations were taken from the BOL blowdown transient calculation from RELAP4-EM/ENC28. For the reflood transient, a similar procedure was used. Hot channel response was recomputed in TOODEE2 using the revised fuel characteristics with burnup from GAPEX, and the average core thermodynamics from the BOL reflood calculation in REFLEX served to provide thermodynamic boundary conditions.

Results obtained from these calculations for five fuel burnup states from BOL to 42.2 GWD/MTM were reported on Table 1.1 and the allowable total peaking factor, F_Q as a function of burnup was shown graphically on Figure 1.1 of the topical report. These results indicated that following a peak F_Q of 2.10 at 12.0 GWD/MTM, that the peaking factor drops off to 1.89 at 42.2 GWD/MTM due to fuel burnup effects. PCTs for the limiting DECLS break over fuel life when operating at the F_Q limits varies within 1°F to 23°F of the 10 CFR 50.46 limit.

B. Evaluation

This reanalysis of the D. C. Cook Unit 1 limiting LOCA has been performed entirely with the recently approved ENC ECCS evaluation model, differentiating it from the previous reload analysis which was performed with an earlier version of the model. In addition, some parametric differences have been introduced in this analysis that have contributed to the change in allowable F_Q from 1.95 to 2.07 at BOL while affecting PCT by only 3°F (2196 to 2199°F).

The principal evaluation model changes that have occurred between ENC-WREM II and ENC-WREM IIA have been the replacement of the reflood and containment transient calculations by ENC developed codes. Independent studies of the effects on evaluation model predictions made for the model review, and evaluation by the staff changes have shown that these modifications have produced some changes (but not significant) between WREM II and IIA for a given break. The two code changes have resulted in allowing small increases in F_Q . As a result of new data on the ECCS injection Δ P penalty in REFLEX, PCT is reduced for a given break, and the use of the ICECON code for the containment back pressure calculation during reflood produces slightly higher pressure predictions than the previous process which in turn allows more rapid reflood rates due to reduced pressure gradients in the steam relief path to the break. Additionally, the TOODEE2 Code has also been updated in WREM-IIA to include steam cooling allowances not previously accepted by the staff. All these changes in combination have acted to reduce PCT $\sim$ 20 to 40°F thereby allowing F_Q to be increased while conforming to the requirements of 10 CFR 50.46.

EMC has reviewed their neutronics methods used for determining the moderator density reactivity controlling reactor shutdown during a large break LOCA, and have introduced a more accurate and more negative value for this parameter in WREM-IIA which results in a reduced power transient in the initial 50 seconds of the LOCA blowdown transient. The peak reduction in power computed with the modified reactivity coefficient amounts to just over 1% of rated power between 7 and 10 seconds after the break. After 50 seconds, the fission product decay energy is the only significant energy source, and both power decay transients are identical. The reduced energy deposition to the coolant during blowdown resulting from this change would have an insignificant effect on PCT for large break transients, but would be expected to show increasing importance as the core voiding rates decrease for smaller breaks, and would act in the direction to reduce PCTs for the smaller breaks.

A final change for this analysis consisted of a correction to the accumulator outlet piping dimension which also produced an insignificant difference in the accumulator discharge transient for the large break transient. This change would have a decreasing significance with reduced break size.

Based on the use of the more recently accepted evaluation model and the influence on PCT that would be produced by the parametric differences between the previous and present analyses, it is concluded from our evaluation that use of the limiting break, a DECLS, obtained from the last reload analysis as the limiting break for the present analysis is acceptable, and that the method and assumptions used in determining the allowable F_Q over fuel life for this break is also acceptable.

Results

The reanalysis of the D. C. Cook Unit No. 1 limiting break using the ENC-WREM IIA-ECCS Evaluation model shows that the F_Q can be raised to the levels shown on Figure 1.1 and is therefore acceptable for plant operation throught the next and subsequent fuel cycles if all analytical assumptions remain valid. The technical specification changes proposed by I&MEC in their March 26, 1981 letter are associated with the new F_Q limits which result from the acceptable LOCA reanalysis. We have reviewed these changes and find they properly implement the new F_Q limits and are therefore acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 21, 1981

References

(1) Exxon Nuclear Company, Donald C. Cook Unit 1 LOCA Analyses Using the ENC WREM-Based PWR ECCS Evaluation Model (ENC WREM-II), XN-76-51, October 1976 and Supplements; Flow Blockage and Exposure Sensitivity Study for D.C. Cook Unit 1 Reload Fuel Using ENC WREM-II Model, XN-76-51 Supplement 1, January 1977; XN-NF-76-51(P) Supplement 2, January 1978; XN-NF-76-51(P) Supplement 3, March 1978.

(2) Letter, Thomas A. Ippolito (NRC) to Warren S. Nechodom (ENC, Topical Report Evaluation, dated March 30, 1979.

(3) Exxon Nuclear Company, Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA, XN-NF-78-30, August 1978, and XN-NF-78-30 Amendment 1, February 1979.

(4) Letter, G.F. Owsley (ENC) to D.F. Ross (NRC), Description of RELAP4-EM ENC28B, dated October 30, 1978.

(5) Exxon Nuclear Company, ICECON: A Computer Program Used to Calculate Containment Break Pressure for LOCA Analysis (Including Ice Condenser Plants), XN-CC-39 Rev. 1, November 1977.

(6) Letter, G.F. Owsley (ENC) to D.F. Ross (NRC), Updates of TOODEE2 Program, dated April 1, 1980.

(7) Exxon Nuclear Company, GAPEX: A Computer Code for Predicting Pellet-to-Clad Heat Transfer Coefficients, XN-75-24, August 31, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-315

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 to Facility Operating License No. DPR-58, issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of Donald C. Cook Nuclear Plant, Unit No. 1 (the facility) located in Berrien County, Michigan. The amendment is effective as of the date of issuance.

The amendment revises the F_Q peaking factor limit.

The application for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 26, 1981, (2) Amendment No. 48 to License Nos. DPR-58 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Maude Reston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 21st day of July, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing