April 8, 1985

Docket No. 50-316

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ORB#1 Gray file (4)

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 67 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 28, 1984 and supported by Exxon letters dated July 7 and August 7, 1984.

The amendment revises the Technical Specifications to allow operation with a nuclear enthalpy rise hot channel factor of 1.55 (1.49 measured).

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/s/DLWigginton

David L. Wigginton, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 67 to DPR-74
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD/OR:DL
CParrish	DWigginton/ts	SVarga		GLainas
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 8, 1985

Docket No. 50-316

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c/o American Electric Power Service Corporation
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Division of Licensing

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2. Safety Evaluation

cc: w/enclosures
See next page

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated August 28, 1984, supportal by Exxon Nuclear letters dated July 7 and August 7, 1984, 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-74 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 67 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 67 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
B 2-2	B 2-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-4a	---------
B 3/4 2-4b	---------

RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figures 3.2-4 and 3.2-5 for 4 and 3 loop operation, respectively.

For: Westinghouse Fuel , for: Exxon Nuclear Company Fuel

$$R = \frac{F^N_{\Delta H}}{1.48\ [1.0 + 0.2\ (1.0 - P)]}\ , \qquad R = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$$

where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and $F^N_{\Delta H}$ = measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ and flow, without additional uncertainty allowance, shall be used.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (as applicable):

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER Limit required by ACTION items a.2 and/or b above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation as shown on Figure 3.2-4 or 3.2-5 (as applicable) for RCS flow rate and R prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

   2. A nominal 75% of RATED THERMAL POWER, and

   3. Within 24 hours of attaining $\geq$ 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-4 or 3.2-5 (as applicable):

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.48 [1 + 0.2 (1-P)] \quad \text{(Westinghouse Fuel)}$$

$$F^N_{\Delta H} = 1.49 [1 + 0.2 (1-P)] \quad \text{(Exxon Nuclear Company Fuel)}$$

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

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 calculated DNBR in the core at or above design 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 certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ 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.

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3411 MWt. The limit on $F_Q(Z)$ is 2.04. The limit on $F^N_{\Delta H}$ is 1.49. The analyses supporting the Exxon Nuclear Company limits are valid for an average steam generator tube plugging of up to 5% and a maximum plugging of one or more steam generators of up to 10%. In establishing the limits, a plant system description with improved accuracy was employed during the reflood portion of the LOCA Transient. With respect to the Westinghouse supplied fuel the minimum projected excess margin of at least 10% to ECCS limits will more than offset the impact of increase steam generator tube plugging.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is 1.97 times the average fuel rod heat flux for Westinghouse supplied fuel and 2.04 times the average fuel rod heat flux for Exxon Nuclear Company supplied fuel.

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

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-4 and 3.2-5, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. The form of this relaxation for DNBR limits is discussed in Section 2.1.1 of the basis.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of 3.5% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-316

1.0 Introduction

To support Cycle 5 operation of D. C. Cook 2, the licensee, Indiana and Michigan Electric Company, provided, in references 1 and 2, LOCA analyses which demonstrated conformance to the requirements of 10 CFR 50.46. These analyses were based upon a total peaking factor, F_Q^T, of 2.04 with a nuclear enthalpy rise hot channel factor, $F_{\Delta H}^N$, of 1.415. In reference 3, we found these analyses to be acceptable.

In reference 4, the licensee submitted a requested change to Technical Specification 3.2.3 to allow operation with a $F_{\Delta H}^N$ of 1.55. The increased $F_{\Delta H}^N$ was requested in order to allow continued full power operation of D. C. Cook 2 during Cycle 5. To support the changes in the Technical Specification, the licensee provided revised LOCA analyses, reference 5. This SER presents our evaluation of these submittals.

2.0 Evaluation Model

Prior to Cycle 5 operation of D. C. Cook 2, the licensee provided the LOCA analysis documented in reference 1. That analysis was performed using the revised Exxon Nuclear Company (ENC) ECCS evaluation model. This model is called EXEM/PWR and is documented in reference 6.

During our review of the EXEM/PWR model, we found that most of the model changes proposed were in compliance with Appendix K to 10 CFR 50. However, we concluded that insufficient documentation was provided to substantiate the correction factors employed on the reflood heat transfer coefficients to account for the effect of mixing vanes and local rod peaking. We determined these factors could not be used in the D. C. Cook 2 LOCA analyses.

As a result of the staff's determination, ENC proposed, in reference 7, a revised method to account for local rod peaking effects on the reflood heat transfer coefficients. This revised method, which was developed specifically for application to Cycle 5 operation of D. C Cook 2 was reviewed and found acceptable by the staff. This revised method was incorporated into the evaluation model utilized in the LOCA analysis and documented in reference 2 which supported operation of Cycle 5 for D. C. Cook 2. Our determination that the revised model utilized for D. C. Cook 2 satisfies Appendix K to 10 CFR 50 is documented in reference 3.

Since the LOCA analyses performed prior to Cycle 5 operation of D. C. Cook 2, ENC has proposed a further revision to their EXEM/PWR ECCS evaluation model. This model change is documented in reference 8 and was used for the revised LOCA analysis discussed herein. This model change proposes a method of applying the EXEM/PWR heat transfer correlations to axial power distributions different from the 1.66 chopped cosine axial power distribution used in the FLECHT tests which formed the basis for the EXEM/PWR correlations.

To apply this methodology, adjustments were made in both the REFLEX and TOODEE2 codes. Within the REFLEX code, differences in axial power distribution relative to the FLECHT tests are accounted for by modifying the initial core average values for QMAX and TINIT. These values are used as input to the carryout rate fraction and quench front correlations. The specific methods used to define these parameters are given in reference 8.

The calculated core reflooding rate, carryout rate fraction and quench front propagation from REFLEX are input to the TOODEE2 code for calculating the fuel rod heat transfer coefficients and cladding temperature response. The same approach used to adjust QMAX and TINIT for the REFLEX code is utilized for the TOODEE2 input except that the hot rod values are adjusted. These modified input parameters are then used directly within the EXEM/PWR reflood heat transfer correlations. An equivalent elevation is then used, based upon conserving the integral power between the fuel rod and the FLECHT rod, to apply the calculated heat transfer coefficients. That is, if the FLECHT rod at 8 feet has the same integral power as the fuel rod at 8.5 feet, the heat transfer coefficients calculated at 8 feet for the FLECHT rod are applied at the 8.5 foot elevation on the fuel rod. The specific ENC method utilized to conserve integral power is discussed in reference 8.

To demonstrate the appropriateness of their model, ENC provided benchmarks to the FLECHT skewed power low flooding rate heat transfer tests 11428, 14331 and 16110. These data were obtained from reference 9. The ENC comparisons showed that the proposed method yielded higher cladding temperatures, and hence lower heat transfer coefficients, than observed in the FLECHT tests. Thus, the ENC methodology is conservative.

In addition to evaluating the information provided by ENC, the staff reviewed some of the FLECHT data to assure that the ENC methodology is conservative. Comparisons were made between FLECHT cosine tests 02414 and 03113 and the skewed power shape tests 15305 and 11003 using the proposed ENC method. These comparisons further illustrated the conservatism of the ENC method. Thus, we find the ENC methodology to be acceptable.

In summary, we find the EXEM/PWR ECCS evaluation model, as applied for D. C. Cook 2, Cycle 5 to be wholly in conformance with Appendix K to 10 CFR 50.

3.0 LOCA Analysis

In reference 5, the licensee provided a revised LOCA analysis for Cycle 5 operation of D. C. Cook 2. The analysis was performed using the same input assumptions as those in reference 2, except the nuclear enthalpy rise hot channel factor, $F_{\Delta H}^{N}$, was increased from 1.415 to 1.55. The axial power profile was change to reflect a total peaking factor, F_{Q}^{T}, of 2.04. In addition, the EXEM/PWR methodology of reference 8 was applied for this analysis.

The analysis used the Cycle 5 core configuration, 85% ENC fuel, and was performed for core burnups of 2 MWD/kg, 10 MWD/kg and 47 MWD/kg. The results of the analysis are given on Table 1. As shown, the analysis showed a peak cladding temperatures of 2014°F, a maximum local zirconium metal-water reaction of 4.7%, and core wide maximum metal-water reactions less than 1%.

We have reviewed the analyses provided by the licensee and have concluded that they satisfy the criteria of 10 CFR 50.46 and were performed with an evaluation model wholly in conformance with Appendix K to 10 CFR 50. Thus, we find that use a nuclear enthalpy rise hot channel factor of 1.55 is acceptable for Cycle 5 operation of D. C. Cook 2.

In addition to the analyses performed for Cycle 5, the licensee also provided LOCA analyses assuming a core configuration of 100% ENC fuel. This core configuration is representative of that expected for Cycle 6 and beyond. The results of the LOCA analyses demonstrated that a core configuration of 100% ENC fuel would be capable of meeting the 10 CFR 50.46 limits. However, these analyses were based upon the Cycle 5 ECCS evaluation model. The staff has not yet judged that the Cycle 5 evaluation model is appropriate for future core reloads. Thus, we will require that the licensee provide revised LOCA analyses prior to Cycle 6 operation.

4.0 Technical Specification Change

The current Technical Specification 3.2.3, entitled "Power Distribution Limits, RCS Flowrate and Nuclear Enthalpy Rise Hot Channel Factor," imposes a measured $F_{\Delta H}^{N}$ limit of 1.36 at 100% power. This value is 4% less than the value of 1.415 utilized in the LOCA analysis of reference 2, which was performed to initially support Cycle 5, in order to account for measurement uncertainty on $F_{\Delta H}^{N}$. In addition, the current Technical Specification places an $F_{\Delta H}^{N}$ limit of 1.49 at 100% power in order to protect DNBR events. Operation of Cycle 5 is restricted by the most limiting of the two $F_{\Delta H}^{N}$ limits as a function of power level.

In reference 4, the licensee submitted changes to Technical Specification 3.2.3 to implement the increased $F_{\Delta H}^N$ of 1.55 assumed in the LOCA analysis. Allowing for the 4% meaurement uncertainty, this yields an $F_{\Delta H}^N$ of 1.49, which is the same as the $F_{\Delta H}^N$ limit for DNBR protection for non-LOCA events. The licensee modified the Technical Specification to delete the previous $F_{\Delta H}^N$ limits for LOCA considerations as the $F_{\Delta H}^N$ limit for non-LOCA events is more restrictive as a function of power level.

We have reviewed the revised Technical Specification 3.2.3 and find it acceptable.

Summary

Based upon the foregoing discussions, we find:

a. The revised LOCA analysis was performed using a model wholly in conformance with Appendix K to 10 CFR 50.

b. The revised analysis shows that continued operation of D. C. Cook 2 Cycle 5 with an $F_{\Delta H}^N$ of 1.55 will meet the requirements of 10 CFR 50.46.

c. The licensee has implemented appropriate Technical Specification changes consistent with the revised LOCA analysis.

Therefore, we conclude that the proposed Technical Specification changes are acceptable.

7.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

8.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Dated: April 8, 1985

Principal Contributors:

R. Jones

6.0 References

(1) XN-NF-84-21(P), "Donald C. Cook Unit 2, Cycle 5, 5% Steam Generator Tube Plugging Limiting Break LOCA/ECCS Analysis," Exxon Nuclear Company, Inc., Richland, WA 99352, February 1984.

(2) XN-NF-84-21(P), Revision 1, "Donald C. Cook Unit 2, Cycle 5, 5% Steam Generator Tube Plugging Limiting Break LOCA/ECCS Analysis," Exxon Nuclear Company, Inc., Richland, WA 99352, May 1984.

(3) Memorandum, R. Houston to G. Lainas, "D. C. Cook 2, Cycle 5," June 6, 1984.

(4) Letter, M. P. Alexich (IMECo) to H. R. Denton (NRC), "Application for Unit 2 Technical Specification Changes For Cycle 5," Docket No. 50-316, August 28, 1984.

(5) XN-NF-84-21(P), Revision 2, "Donald C. Cook Unit 2, Cycle 5, 5% Steam Generator Tube Plugging, Limiting Break LOCA/ECCS Analysis," Exxon Nuclear Company, Inc., Richland, WA 99352, July 1984.

(6) XN-NF-82-20(P), Rev. 1, August 1982; Supplement 1, March 1982; and Supplement 2, March 1982, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Inc., Richland, WA 99352.

(7) Letter, J. C. Chandler (ENC) to H. R. Denton (NRC), Subject: Additional Information Regarding Unit 2 Cycle 5 LOCA ECCS Analysis, May 7, 1984.

(8) XN-NF-82-20(P), Rev. 1, Supplement 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates: Adjustments To FLECHT Based Heat Transfer Coefficients," Exxon Nuclear Company Inc., Richland, WA 99352, July 1984.

(9) WCAP-9183, "PWR FLECHT Skewed Profile Low Flooding Rate Test Series Evaluation Report," Westinghouse Nuclear Electric Systems, Pittsburgh, PA, November 1977.

TABLE 1

1.0 DECLG Break Fuel Response Results for Cycle 5

Peak Rod Average Burnup (MWD/kg)	2.0	10.0	47.0
Initial Peak Fuel Average Temperature (°F)	2151	2060	1629
Hot Rod Burst			
• Time (sec)	69.5	70.5	78.5
• Elevation (ft)	7.0	7.0	7.75
• Channel Blockage Fraction	.25	.28	.47
Peak Clad Temperature			
• Time (sec)	287	288	269
• Elevation (ft)	9.63	9.63	9.38
• Temperature (°F)	2007	2014	1993
Zr-Steam Reaction			
• Local Maximum Elevation (ft)	9.63	9.63	9.38
• Local Maximum (%)	4.6	4.7	4.5
• Core Maximum	<1.0	<1.0	<1.0