Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

December 21, 1982

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

Subject: Dresden Station Unit 2
Proposed Amendment to Appendix A
Technical Specifications to
Support Operation with Fuel
Supplied by Exxon Nuclear Company
NRC Docket No. 50-237

References (a): T. J. Rausch letter to H. R. Denton dated January 11, 1982.

(b): J. D. Hegner letter to L. DelGeorge dated April 29, 1982.

Dear Mr. Denton:

Pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend Appendix A, Technical Specifications, to Provisional Operating License DPR-25 for Dresden Unit 2. This amendment is being submitted to allow the use of fuel assemblies designed and manufactured by Exxon Nuclear Company Inc. (ENC) for the ensuing Cycle 9 reload and future reloads at Dresden Unit 2.

A001 s
1/40 Non-Prop
1/6 Prop

Attachment 6 to this letter provides the changes proposed to the Technical Specifications and Bases. A detailed description of these changes, along with a general discussion of the Dresden 2 Cycle 9 Reload is provided in Attachment 1.

change: ε
PDR ; NP
LPDR ; NP
NSIC ; NP
NTIS ; NP

These proposed changes have received On-site and Off-site review and approval.

limited Dist of Prop Encls

Attachments 2, 3 and 4 to this letter provide the Dresden 2 plant specific reload, transient and LOCA analysis report prepared by ENC. Attachments 2 and 3 contain information proprietary to the Exxon Nuclear Company. As such, they are accompanied by an affidavit (Attachment 5) signed by ENC, 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.

Rec'd w/check: $4000.00

Accordingly, it is respectfully requested that the information which is proprietary to Exxon Nuclear Company, Inc. 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 this application for withholding or the supporting ENC affidavit should be addressed to R. B. Stout, Manager Licensing and Safety Engineering, Exxon Nuclear Company, 2101 Horn Rapids Road, P.O. Box 130, Richland, Washington 99352.

In Reference (a), Commonwealth Edison requested nearly identical changes to operate Dresden Unit 3 with fuel supplied by ENC. Amendment 63 to the Dresden 3 License DPR-25 authorizing these changes was transmitted by Reference (b). Because this request is similar to that previously reviewed and authorized in Reference (b), it does not involve a complex issue; therefore this constitutes a 10 CFR 170 Class III request. As such, a $4,000 fee remittance is enclosed.

Please address any questions you may have to this office.

Three (3) signed originals and thirty-seven (37) copies of this letter with Attachments 1, 4, 5, 6, 7, 8 and 9 are provided for your use. In addition, six (6) copies of this letter with proprietary Attachments 2 and 3 are also being provided at this time.

Very truly yours,

Thomas J. Rausch

Thomas J. Rausch
Nuclear Licensing Administrator

lm

cc: Region III Inspector - Dresden

Attachment (separate sheet)

SUBSCRIBED AND SWORN to
before me this 21st day
of December, 1982

Rosalie A. Prenta
Notary Public

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Attachments (1): Dresden 2 and Cycle 9 Reload Discussion and Description of Technical Specification Changes

(2): Dresden 2, Cycle 9 Reload Analysis Report XN-NF-82-77(P), Rev. 1 dated November 1982

(3): Dresden 2, Cycle 8 Plant Transient Analysis Report, XN-NF-82-84(P), Rev. 1 dated November 1982

(4): Dresden Unit 2 LOCA Analysis Using the Exem Evaluation Model MAPLHGR Results, XN-NF-82-88, Rev. 1, dated November 1982

(5): Affidavit of R. B. Stout Attesting to the proprietary nature of XN-NF-81-75(P), dated December 1982

(6): Proposed Technical Specification Changes to DPR-19.

(7): Dresden 2, Cycle 9 Reload Analysis Report, XN-NF-82-77 (Non Proprietary), Rev. 1, dated November 1982.

(8): Dresden 2, Cycle 9 Plant Transient Report, XN-NF-82-84 (Non Proprietary), Rev. 1, dated December 1982

(9): Errata and Addenda Sheet No. 7 to Dresden and Quad Cities LOCA Analyses, NEDO 24146

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ATTACHMENT 1

Dresden 2 Cycle 9 Reload Discussion and

Description of Technical Specification Changes

Dresden 2 Cycle 9 will represent the first reload of Exxon fuel in Unit 2, and the first use of 9x9 fuel in a Jet Pump BWR. The following safety evaluation addresses the fuel design, reload analyses and Technical Specification changes supporting operation of D2C9 Reload XN-1. The evaluation is divided into four sections as follows:

I. Reload Fuel and Core Design
II. Transient and Accident Analyses
III. Technical Specifications
IV. Summary

Sections I and II are based on the Dresden Station Unit 2 Cycle 9 Reload Analysis, XN-NF-82-77 (Attachment 2), the Plant Transient Analysis for Dresden 2 Cycle 9, XN-NF-82-84 (Attachment 3), and the Dresden Unit 2 LOCA Analysis using the ENC EXEM/BWR Evaluation Model MAPLHGR Results, XN-NF-82-88 (Attachment 4). Section III describes the proposed Technical Specification changes required for Cycle 9. Following the Section IV summary is a list of references primarily consisting of ENC Topical Reports on their generic Jet Pump BWR methodology.

I. RELOAD FUEL AND CORE DESIGN

Dresden 2 Reload XN-1 will consist of 220 ENC 8x8 reload assemblies designated as type XN8D3.02 and 4 ENC 9x9 lead assemblies (LA's) designated as type D2 9x9 2.97. The core loading will consist of the following:

Number of Bundles	Fuel Type*
92	GE 8x8-2.50%
24	GE 8x8-2.62%
384	GE P8x8R-2.65%
220	XN-1 8x8-2.83%
4	XN-1 9x9-2.78%

* Bundle average enrichments

A. Fuel Mechanical Design

The mechanical design of the 8x8 reload fuel is described generically in Reference 1. In general, design criteria are established to limit the stress, strain and overall duty on the fuel rod or bundle during normal and transient operation. In addition, the fuel is designed to be mechanically compatible with other reactor internals, fuel handling equipment and existing fuel.

The four 9x9 LA's have been designed to be mechanically and hydraulically compatible with the co-resident 8x8 fuel and reactor internals. The D2 9x9 2.97 fuel design is a 9x9 array with 80 fuel rods (6 containing Gadolinia) and one spacer capture water rod. The active fuel length is 145.24 inches which includes a 6 inch blanket of natural U at both top and bottom. Enriched fuel pellets have a 1% dish volume while the natural fuel pellets have a 0.5% dish volume. Fuel rod pitch is maintained via seven Zircaloy-4 spacers with Inconel springs. Lower tie plates are drilled to improved reflood capability and employ a spring seal at the tie plate channel interface to limit coolant leakage to the bypass region as a result of channel side wall deformation (bulge) with exposure.

The 9x9 LA fuel design utilizes Zircaloy-2 cladding with the exception of 6 Zr-4 clad fuel rods in one of the 4 9x9 LA's. Zr-4 differs from Zr-2 in that Zr-4 contain no nickel while the other alloying metals exist in about the same concentrations in both Zr-2 and Zr-4. Zr-4 is already used in BWR channels and spacers as well as for fuel pin cladding in PWRs.

B. Thermal Hydraulic Design

The primary thermal hydraulic design criteria for 8x8 XN-1 fuel are identified in Reference 1. As discussed in the D3C8 NRC SER (Amendment 63 issued April 29, 1982), 8x8 XN-1 fuel is thermal hydraulically compatible with GE fuel.

Because XN-1 9x9 fuel is also thermal hydraulically compatible with XN-1 8x8 and GE 8x8 fuel, the core thermal hydraulic response is expected to be similar to previous reloads.

Analyses made during the calculation of the Fuel Cladding Integrity Safety Limit demonstrated that a MCPR Safety Limit of 1.05 provides assurance that at least 99.9% of the 8x8 and 9x9 fuel rods in the core would be expected to avoid boiling transition during steady state operation at the safety limit. Refer to Reference 2 for further discussion of the methodology.

C. Fuel Centerline Melting and Cladding Strain

One of the ENC's primary thermal hydraulic design criterion for D2 reload XN-1 fuel is that fuel design and operation will be such that fuel centerline melting is not expected for anticipated operational occurences (transients) throughout the life of the fuel. To demonstrate compliance with this criterion, ENC has performed transient overpower analyses for a fuel rod history (peak LHGR vs. exposure) which represents a conservative upper bound on peak rod power over the life of the fuel bundle. The results indicate that substantial margin to centerline melting is assured for both 8x8 XN-1 and 9x9 XN-1 fuel. The cladding strain at 120% overpower condition was calculated and determined to be less than the ENC design criteria of .2% plastic strain for 8x8 XN-1 fuel.

For previous reloads, GE provided an LHGR design limit, which was incorporated in the Technical Specifications as an operating limit, to ensure margin during transients to the 1% plastic strain Safety Limit assumed in the GE licensing basis. In addition, a Technical Specification provision for reducing the APRM scram and rod block setting by FRP/MFLPD was incorporated to ensure margin to 1% plastic strain during transients initiated from reduced core flow with excessive peaking (i.e. peaking which would result in an LHGR in excess of the operating limit if recirc flow were increased to rated). For D2C9 this approach and the corresponding Technical Specification provisions will be maintained for GE fuel. For XN-1 reload fuel, the fuel design is such that margin to centerline melt and the design criterion of 0.2% plastic strain is assured for overpower conditions throughout the life of the fuel as demonstrated by the Fuel Design Analyses. Since this inherently protects against 1% plastic strain and assures margin to the expected threshold for strain induced cladding failure, an LHGR Safety Limit is not necessary and has not been specified for ENC fuel. As a result, no Technical Specification LHGR Operating Limit or APRM scram adjustment is required. However, to ensure applicability of the Fuel Design Analysis, proposed Technical Specifications 3.1.B and 4.1.B.2 will require a daily surveillance on power distribution for ENC fuel. In most cases, operation within the MCPR and MAPLHGR limits will ensure that the power distribution for ENC fuel remains within the assumptions of the Fuel Design Analysis. However, this surveillance will ensure under all conditions that the peak LHGR for ENC fuel is procedurally controlled to provide margin to centerline melt for overpower conditions initiated from rated power or reduced flow.

D. Nuclear Design

The 8x8 XN-1 fuel design consists of 63 fuel rods and one water rod. The average assembly enrichment is 2.83% which includes a six inch natural U blanket at both top and bottom. The average enrichment of the central region (excluding blanket) is 3.02%.

Five burnable poison rods containing a Gd_2O_3-UO_2 mixture are utilized to reduce initial bundle reactivity. The specific neutronic design parameters and pin enrichment distribution are provided in Section 4 of Attachment 2. The 9x9 XN-1 fuel design consists of 80 fuel rods and one water rod. The average assembly enrichment is 2.78% which also includes a six inch natural U blanket at both top and bottom. The average center zone enrichment is 2.97%. The 9x9 fuel design contains six poison rods utilizing a Gd_2O_3-UO_2 mixture to reduce initial bundle reactivity.

Core Reactivity - The calculated BOC9 cold core keff values at all rods out and all rods in are 1.111 and 0.958 respectively. The shutdown margin with the strongest control rod out was determined to be 1.1% Δk with the most reactive point in the cycle being BOC. Therefore R is merely equal to 0.04% Δk to account for inverted tubes in the control blades. The Standby Liquid Control System (which is designed to inject a quantity of boron that produces a concentration of no less than 600 ppm of boron in the reactor core in less than 100 minutes) was calculated to provide a shutdown margin of 4.4% Δk for cold conditions with all control rods in their full power positions. The calculated shutdown margins for the strongest rod out and SBLC system are both well in excess of their respective Technical Specification Requirements, 0.25%+R and 3%.

Core Stability - The decay ratio for the D2C9 core was determined to be 0.46 at the intersection of the natural circulation line and the 100% Flow Control Line.

Fuel Storage Vault/Pool Criticality - Technical Specification 5.5 requires that the keff of the spent fuel pool be $\leq$ .95 and that of the new fuel storage vault $<$.90 when dry ($<$ .95 when flooded). In NEDE-24011, GE states that these criteria will be met for GE fabricated racks if fuel bundle reactivities are limited to koo $\leq$1.31 for the rack dimensions utilized in the Dresden spent fuel pool and $\leq$ 1.30 for the rack dimensions utilized in the new fuel vault, where k_{oo} is calculated in an infinite array of similar fuel in the core configuration (as opposed to the storage configuration). GE has calculated koo's for their fuel designs and demonstrated that the criterion is satisfied for all GE fuel. ENC has calculated k_{oo} for 8x8 and 9x9 XN-1 reload fuel and for a comparable GE fuel design. Based on the comparison in Appendix A of Attachment 2 and the criteria from NEDE-24011, it is concluded that adequate margin to the Technical Specification keff limits exists for storage of both 8x8 and 9x9 XN-1 reload fuel in the vault and pool (for GE designed racks).

For the high density fuel storage racks designed by Nuclear Services Corporation, criticality analyses have been performed for ENC fabricated fuel with a center zone enrichment of 3.02 w/o and demonstrated that the 0.95 keff requirement is met. Based on bundle reactivity comparison provided by ENC, the high density fuel storage racks meet the 0.95 keff requirement for both ENC 8x8 and ENC 9x9 fuel when gadolinia is considered.

II. TRANSIENTS AND ACCIDENTS

A. Anticipated Operational Occurrences (Transients)

In order to determine operating limits for D2C9, ENC has considered eight categories of core-wide potential transients (as described in Reference 3) and provided analyses results for the following three transients to determine the thermal margin for D2C9.

- Generator Load Rejection without Bypass (LRw/oB)
- Feedwater Controller Failure (FWCF)
- Loss of Feedwater Heating (LOFWH).

The other core-wide transients are inherently non-limiting or bounded by one of the above. In adition, two local events, Rod Withdrawal Error and Fuel Loading Error, were analyzed as described in Reference 4 and determined to be non-limiting. The results of the core-wide and local transient analyses are provided in Attachment 2 (XN-NF-82-77, D2C9 Reload Analyses) and Attachment 3 (XN-NF-82-84, D2C9 Plant Transient Analysis Report). The 9x9 XN-1 fuel design was incorporated into the analysis of these events by choosing limits that maintain the same bundle power for 9x9 fuel as for 8x8 fuel and then verifying the limits by transient analyses. The Generator Load Rejection without Bypass was determined to be the limiting event for D2C9, resulting in a ΔCPR of 0.26 for 8x8 fuel and .30 for 9x9 which, when combined with the 1.05 Safety Limit, requires a MCPR operating limit of 1.31 for all 8x8 fuel types and a MCPR operating limit of 1.35 for 9x9 XN-1 fuel.

<u>Core-Wide Transients</u> - The plant transient model used to evaluate the LRw/oB and FWCF events was ENC's COTRANSA code (Reference 3) which incorporates a one-dimensional neutronics model to account for shifts in axial power shape resulting from rapid pressurization and void collapse. The LRw/oB event was found to be the most limiting event and therefore analyzed statistically while the non-limiting FWCF was analyzed deterministically (using bounding values as input parameters). The LOFWH event was analyzed deterministically with ENC'S PTSBWR code (Reference 3) which uses a point-kinetics neutronics model since rapid pressurization and void collapse do not occur for this event. Both codes utilize a multi-node steam line model to accomodate pressure waves in the steam line.

One of the statistically varied inputs to the LRw/oB analysis was the control rod speed during the reactor scram. Actual scram time data from previous cycles on Dresden 2 were used to generate the scram time distribution assumed in determining the Δ CPR distribution for this event. In order to assure the applicability of the LRw/oB analysis to cycle 9 operation, compliance with the assumed scram time distribution must be verified throughout cycle 9 as required by proposed T.S. 4.3.C.3. If the current cycle scram speeds deviate from the assumed distribution, an adjustment to the MCPR operating limit may be required. The method for checking compliance and adjusting the MCPR operating limit is provided in proposed Technical Specifications 4.3.C.3 and 3.5.K.

Local Transients

As shown in Attachment 2, the results of the Fuel Loading Error and Rod Withdrawal error were bounded by the LRw/oB event and are therefore non-limiting. Based on the RWE results, the rod block monitor setpoint will be increased from the current value of 107% to 110% to provide additional flexibility in utilizing the allowable power/flow operating region above the 100% flow control line. The Δ CPR for the RWE event with a 110% full flow RBM setpoint was 0.13. The Δ CPR for the fuel loading error event was 0.14. All of the Δ CPRs are less than the limiting value of 0.26 calculated for the LRw/oB event.

Reduced Flow Operation

ENC has provided MCPR operating limits for manual and automatic flow control reduced flow operation in Attachment 2. These values are based on the analyses provided in Reference 16 for D3C8 which ENC has indicated are applicable to D2C9.

ASME Overpressurization Analysis

In order to demonstrate compliance with the ASME Code Overpressurization criteria of 110% of design vessel pressure, the MSIV closure event with failure of the MSIV position scram was analyzed with ENC's COTRANSA code. The maximum pressure observed in the analysis was about 1349 psig or 108% of reactor vessel design pressure. The corresponding steam dome pressure was about 1325 psig, for a vessel differential pressure of 24 psi. This includes the effects of the ATWS RPT which was assumed to initiate at a nominal pressure setpoint of 1240 psig. The ASME limit for peak vessel pressure of 1375 psig (110% of design pressure) is therefore equivalent to a dome pressure limit of 1351 psig (1375-24). The Technical Specification Safety Limit of 1325 psig is based on dome pressure and therefore conservatively assumes a 50 psi vessel dp (1375-1325). The proposed safety limit of 1345 psig dome pressure is based on

a 30 psig vessel dp which removes excess conservatism while continuing to bound expected differential pressure behavior, especially when the lack of forced flow imposed by RPT is considered. The choice of 1345 psig thus assures compliance with the ASME criterion of 1375 psig peak vessel pressure while also maintaining consistency with the D3R7C8 pressure safety limit.

B. Postulated Accidents

In support of D2C9 operation, ENC has reanalyzed the Loss of Coolant Accident (LOCA) to determine MAPLHGR limits for XN-1 fuel and the Rod Drop Accident (RDA) to demonstrate compliance with the 280 cal/gm Technical Specification limit. The results of these analyses are presented in Section 6 of Attachment 2. The methodology for the RDA analysis is described in Reference 4 and that for the LOCA analysis is provided in References 6 thru 13.

Loss of Coolant Accident - Reference 6 describes ENC's generic jet pump BWR3 LOCA break spectrum analysis which defined the limiting break for BWR 3's to be a double-ended guillotine break in the recirculation piping on the suction side of the pump. The analysis of this event for Dresden 2 is provided in Attachment 4 and summarized in Attachment 2. Operation within the MAPLHGR limits of Section 6.1 for ENC 8x8 fuel and Table A.2 for the 9x9 LA's (Attachment 2) will ensure that the peak cladding temperature remains below 2200°F, local Zr-H_2O reaction remains below 17% and core-wide hydrogen production remains below 1% for the limiting LOCA event. The LOCA analysis of Attachment 4 was performed for an entire core of ENC 8x8 reload fuel and therefore provide MAPLHGR's for ENC fuel only. The MAPLHGR limits for D2 9x9 XN-1 LA's were established to maintain nodal powers equivalent to the ENC 8x8 fuel assemblies. Confirmatory ECCS analyses were performed to verify that the 9x9 MAPLHGR limits maintain peak clad temperatures and local oxidation fractions within 10CFR50 Appendix K limits.

As discussed previously, ENC reload fuel is hydraulically and neutronically compatible with G.E. fuel. Therefore, the existing G.E. LOCA Analysis (Reference 14) and MAPLHGR limits will remain applicable during D2C9 and future cycles with GE/ENC mixed cores.

Rod Drop Accident - ENC's methodology for analyzing the Rod Drop Accident (RDA) is described in Reference 4 and utilizes a generic parametric analysis which calculated the fuel enthalpy rise during postulated RDA's over a wide range of reactor operating variables. For D2C9, Section 6 of Attachment 2 shows a value of 111 cal/gm for the maximum deposited fuel rod enthalpy during the worst case postulated RDA. This value is well below the Technical Specification limit of 280 cal/gm. To ensure compliance with the RDA analysis assumptions, control rod sequencing below 20% core thermal power must comply with G.E.'s Banked Position Withdrawal Sequencing constraints (Reference 15).

III. TECHNICAL SPECIFICATIONS

Attachment 6 provides proposed Technical Specification changes to support D2C9 operation with ENC 8x8 and 9x9 fuel. The following sections highlight the major areas requiring revision and identifies the associated sections of the Technical Specifications.

A. GENERAL

Throughout the Technical Specifications and bases, sections have been revised to reflect the appropriate Exxon Methodologies and references and delete General Electric methods and references where necessary. Also, for each revised specification as identified below, the corresponding section of the bases has been revised as required.

B. LHGR

As described previously, no LHGR Safety Limit or Operating Limit is specified for ENC fuel. Operation within the MCPR and MAPLHGR limits, and the power distribution assumptions of the Fuel Design Analyses will protect against fuel centerline melting and thereby protect against strain-induced cladding failures. All Technical Specification sections referring to LHGR or FLPD have been revised to apply only to GE fuel. New specifications have been proposed which require surveillances on ENC fuel to ensure applicability of the Fuel Design Analysis.

In addition to the above, all references to 7x7 fuel and the power spiking penalty have been deleted since there will be no 7x7 fuel in D2C9 and sufficient margin to LHGR limits exist to accommodate the expected power spiking due to fuel densification.

T.S. Section	Description
1.K	Definition of FLPD revised to apply to GE fuel only.
1.1.A.1/2.1.B.1 3.1.B/4.1.B Table 3.2.3 Note 2	APRM Scram and Rod Block equations revised to provide MFLPD/FRP adjustment for GE fuel only. For ENC fuel, a requirement to ensure compliance with the Fuel Design Analysis has been added.
3.5.J/4.5.J	Revised to require LHGR limit and surveillance for GE fuel only. Deletes reference to 7x7 fuel and power spiking.

C. MCPR

T.S. Section	Description
1.1.A	MCPR Safety Limit changed to 1.05
3.5.K	MCPR LCO changed to 1.31 for all 8x8 fuel types and 1.35 for the 9x9 LA's. Revised to indicate new curves for determining MCPR limit during operation at reduced flow and to require adjustment of the limit if scram times fall outside the distribution assumed in the transient analysis.
Figure 3.5.2	Replaced with new figures for determining MCPR limits during operation at reduced flow.

D. Reactor Coolant System Pressure Safety Limit (Section 1.2)

Changed from 1325 to 1345 psig. Previous value assumed a vessel pressure drop of 50 psi. New valve is conservative compared to the actual pressure drop as determined by analysis.

E. RBM Setting (Table 3.2.3)

Changed from $.65W_d+42$ (107% at full flow) to $.65W_d+45$ (110%) based on results of RWE analysis.

F. Section 3.5.D.3.a

This section which allowed operation with only 4 ADS valves during D2 Cycle 7 has been deleted since it is no longer applicable. Analytical support for such operation will have to be purchased from Exxon and licensed later if desired.

G. MAPLHGR

Section	Description
3.5.I/4.5.I	Revised to distinguish GE MAPLHGRs which are functions of nodal exposure from ENC MAPLHGRs which are dependent on bundle average exposure.
Figure 3.5.-1	Add MAPLHGR curve for ENC fuel types XN 8D3.02 and D2 9x9 2.97 while deleting curves for 7x7 fuel. Extend MAPLHGR curves for GE fuel types P8DRB265H,

Section	Description
Figure 3.5.-1 (Cont'd)	8DRB265L, P8DRB282 and P8DRB265L out to 40,000 MWD/ST. In addition, since Errata and Addenda Sheet No. 7 to NEDO-24146 provided extra margin for P8DRB265L over 8DRB265L, a separate curve is now provided. E&A Sheets for the changes to the GE fuel type MAPLHGR curves are provided as Attachment 9. Table 4P of E&A Sheet No. 7 shows MAPLHGR values for P8DRB265L as applying only to Quad Cities. General Electric was contacted regarding this situation and indicated that Table 4P does apply to Dresden as well as Quad Cities.

H. Scram Time Surveillance

Specification 4.3.C.3 has been added to require verification after each set of scram timing data that the current scram speeds fall within the distribution assumed in the transient analyses.

I. License Condition M

The previously approved provisions to allow operation with one recirculation loop out of service are being revised slightly to refer to correct Technical Specifications references and to only specify .03 CPR adders as opposed to actual MCPR values. These changes are strictly for increased clarity and to preclude having to change values every cycle.

TJR/lm

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REFERENCES

1. XN-NF-81-21(A), "Generic Design Report-Mechanical Design for Exxon Nuclear Jet Pump BWR Fuel Assemblies," dated October, 1981.

2. XN-NF-524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors" dated November, 1979.

3. XN-NF-79-71 Revision 2 "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors" dated November, 1981.

4. XN-NF-80-19(A), Volume 1 (Supplements 1 and 2), "Exxon Nuclear Methodology for Boiling Water Reactors Neutronics Methods for Design and Analysis" dated May 1980.

5. XN-NF-81-22(P), September 1981
Generic Statistical Uncertainty Analysis Metholology

6. XN-NF-81-71(A), October 1981
Generic Jet-Pump BWR3 LOCA Analysis Using the ENC EXEM Evaluation Model.

7. XN-NF-82-88, "Dresden Unit 2 LOCA Analysis Using the ENC EXEM/BWR Evaluation Model-MAPLHGR Results" dated November, 1981 (Attachment 4).

8. XN-NF-80-19(A), Volume 2, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
EXEM: ECCS Evaluation Model, Summary Description

9. XN-NF-80-19(A), Volume 2A, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
RELAX: A RELAP4 Based Computer Code for Calculating Blowdown Phenomena

10. XN-NF-80-19(A), Volume 2B, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
FLEX: A Computer Code for Jet Pump BWR Refill and Reflood Analysis

11. XN-NF-80-19(A), Volume 2C, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
Verification and Qualification of EXEM

12. XN-CC-33(A), Revision 1, November 1975
HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option

13. XN-NF-81-58(P), August 1981
RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model

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REFERENCES (Con't)

14. NEDO-24146A Revison 1, "Loss of Coolant Accident Analyses-Quad Cities 1/2, Dresden 2/3" dated April 1979.

15. NEDO-21231, "Banked Position Withdrawal Sequence" dated January 1977.

16. XN-NF-81-84(P), "Dresden Unit 3 Analysis for Reduced Flow Operation", November 1981.

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ATTACHMENT 5

A F F I D A V I T

STATE OF Washington)
ss.
COUNTY OF Benton)

I, Richard B. Stout, being duly sworn, hereby say and depose:

1. I am Manager, Licensing and Safety Engineering, for Exxon Nuclear Company, Inc. ("ENC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with ENC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the documents (1) XN-NF-82-77(P), entitled "Dresden Unit 2 Cycle 9 Reload Analysis," and (2) XN-NF-82-84(P), entitled "Plant Transient Analysis for Dresden Unit 2 Cycle 9," referred to as "Documents". Information contained in these Documents has been classified by ENC as proprietary in accordance with the control system and policies established by ENC for the control and protection of information.

4. The Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by ENC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as being proprietary and confidential.

5. The Documents have been made available to the United States Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document not be disclosed or divulged.

6. The Documents contain information which is vital to a competitive advantage of ENC and would be helpful to competitors of ENC when competing with ENC.

7. The information contained in the Documents is considered to be proprietary by ENC because it reveals certain distinguishing aspects of safety analysis methods which secure competitive economic advantage to ENC for fuel design optimization and improved marketability, and includes information utilized by ENC in its business which affords ENC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Documents.

8. The disclosure of the proprietary information contained in the Documents to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it extremely valuable insights into safety analysis methods, and would result in substantial harm to the competitive position of ENC.

9. The Documents contain proprietary information which is held in confidence by ENC and is not available in public sources.

10. In accordance with ENC's policies governing the protection and control of information, proprietary information contained in the Documents has been made available, on a limited basis, to others outside ENC only as required and under suitable agreement providing for non-disclosure and limited use of the information.

11. ENC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. These Documents provide information which reveals safety analysis methods developed by ENC over the past several years. ENC has invested millions of dollars and many man-years of effort in developing the analysis methods revealed in the Documents. Assuming a competitor had available the same background data and incentives as ENC, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as ENC.

13. Based on my experience in the industry, I do not believe that the background data and incentives of ENC's competitors are sufficiently similar to the corresponding background data and incentives of ENC to reasonably expect such competitors would be in a position to duplicate ENC's proprietary information contained in the Documents.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

Richard B Stout

SWORN TO AND SUBSCRIBED

before me this 7 day of

Dec, 1982

Vicky Brown

NOTARY PUBLIC