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

January 11, 1982

RECEIVED
JAN 18 1982
US NUCLEAR REGULATORY COMMISSION
DOCUMENT MANAGEMENT BR
TIDC

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

Subject: Dresden Station Unit 3
Proposed Amendment to Appendix A
Technical Specifications to
Support Operation with Fuel
Supplied by Exxon Nuclear Company
NRC Docket No. 50-249

References (a): J. S. Abel letter to D. G. Eisenhut dated February 20, 1981.

(b): J. S. Abel letter to D. G. Eisenhut dated March 5, 1981.

Dear Mr. Denton:

Pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend Appendix A, Technical Specifications, to Facility Operating License DPR-25 for Dresden Unit 3. This amendment is being submitted to allow the use of fuel assemblies designed and manufactured by Exxon Nuclear Company Inc. (Enc) for the ensueing Cycle 8 reload and future reloads at Dresden Unit 3.

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 3 Cycle 8 Reload is provided in Attachment 1.

A001
1/40 NP
1/6 Prop

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

change: L E
PDR 1 NP
LPDR 1 NP
NTIS 1 NP
NSIC 1 NP

Attachments 2, 3 and 4 to this letter provide the Dresen 3 plant specific reload, transient and LOCA analysis reports prepared by ENC. Attachment 4 contains information proprietary to the Exxon Nuclear Company. As such, it is 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.

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 G. F. Owsley, Manager of Reload Fuel Licensing, Exxon Nuclear Company, 2101 Horn Rapids Road, P.O. Box 130, Richland, Washington 99352.

In Reference (a), Commonwealth Edison provided notification of our intent to operate Dresden Unit 3 with fuel supplied by ENC. The 10CFR 170 Class III amendment fee of $4,000 for this proposal was provided previously in Reference (b).

Please address any questions you may have to this office.

Three (3) signed copies of this letter with Attachments 1, 2, 3 5 and 6 are provided for your use. In addition, six (6) copies of this letter with proprietary Attachment 4 and the affidavit of Attachment 5 are also being provided at this time. The non-proprietary version of Attachment 4 will be transmitted under separate cover.

Very truly yours,

Thomas J. Rausch

Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

cc: RIII Inspector - Dresden

Attachments (1): Dresden 3 Cycle 8 Reload Discussion and Description of Technical Specification Changes.

(2): Dresen 3 Cycle 8 Reload Analysis Report, XN-NF-81-76, Rev. 1 dated December 1981.

(3): Dresden 3 Cycle 8 Plant Transient Analysis Report, XN-NF-81-78, Rev. 1 dated December 1981.

(4): Dresden Unit 3 LOCA Analysis Using the Exem Evaluation Model MAPLHGR Results, XN-NFG-81-75(P) dated November 1981.

(5): Affidavit of James N. Morgan Attesting to the proprietary nature of XN-NF-81-75(P), dated November 12, 1981.

(6): Proposed Technical Specification Changes to DPR-25.

SUBSCRIBED AND SWORN to
before me this 11th day
of January, 1982.

Rosalie A. Pienta

Notary Public

0168T

50-249

DRESDEN 3 Cycle 8 Discussion

AND

Description Of Technical Specification Changes

Received wth ltr dtd 01/11/82

ATTACHMENT 1

Dresden 3 Cycle 8 Discussion
and Description of Technical Specification Changes

Dresden 3 Cycle 8 will represent the first reload of a Jet Pump BWR utilizing fuel fabricated by Exxon Nuclear Company (ENC). The following discussion addresses the fuel design, reload analyses and Technical Specification changes supporting operation of the D3C8 Reload utilizing XN-1 Enc fuel. The discussion is divided into four sections as follows:

I. Reload Fuel and Core Design
II. Transient and Accident Analyses
III. Technical Specifications
IV. List of References

Sections I and II are based on the Dresden Station Unit 3 Cycle 8 Reload Analysis, XN-NF-81-76 (Attachment 2), the Dresden 3 Cycle 8 Plant Transient Analysis Report, XN-NF-81-78 (Attachment 3) and the Dresden Unit 3 LOCA Analysis MAPLHGR Results, XN-NF-81-75 (Attachment 4). Section III describes the major proposed Technical Specification changes required for Cycle 8. Section IV 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 3 Reload XN-1 will consist of 224 reload fuel assemblies fabricated by ENC and designated as type XN8D2.69-5. The core loading will consist of the following:

Number of Bundles	Fuel Type
72	GE 8x8-2.50%
228	GE 8x8-2.62%
200	GE P8x8R-2.65%
224	XN 1 8x8-2.69%

As shown in Figure 4.2 of Attachment 2, the reload fuel will be loaded in a 2 out of 4 scatter with the exception of the core axes and peripheral regions. This loading pattern is similar to that utilized in previous reloads.

A. Fuel Mechanical Design

The mechanical design of the 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.

A. Fuel Mechanical Design (Cont'd)

The XN-1 fuel design is an 8x8 array with 63 fuel rods (5 containing Gadolinia) and one water rod. Active fuel length is 145.24 inches which includes a 6 inch blanket of natural U at both top and bottom. Enriched fuel pellets are dished, natural pellets are not. Fuel rod pitch is maintained via seven Zircaloy-4 spacers with Inconel springs. Lower tie plates are drilled to improve reflood capability and employ a spring seal at the interface with the channel to limit coolant leakage to the bypass region as a result of channel side wall deformation (bulge) with exposure.

B. Thermal Hydraulic Design

Section 3 of Attachment 2 identifies the primary thermal hydraulic design criteria for XN-1 fuel. ENC has performed testing on full scale assemblies to determine the hydraulic resistances and pressure drops for XN-1 8x8 fuel and G.E. 8x8R fuel. The test results verify that XN-1 fuel is thermal hydraulically compatible with G.E. fuel and core thermal hydraulic response is expected to be similar to previous reloads.

The Fuel Cladding Integrity Safety Limit was calculated using a Monte Carlo technique to convolute uncertainties in the calculation of core power distribution and critical power ratio. The analysis demonstrated that a MCPR Safety Limit of 1.05 provides assurance that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition during steady state operation at the Safety Limit. Refer to Section 3.7 of Attachment 2, Attachment 3 and Reference 2 for further discussion of the methodology.

C. Fuel Centerline Melting during Overpower Conditions

One of the ENC's thermal hydraulic design criteria for D3 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 criteria, 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 (batch average discharge burnup of 30 GWD/MT). ENC has determined that the conditions

C. Fuel Centerline Melting During Overpower Conditions (Cont'd)

for minimum margin to centerline melt over the above mentioned fuel rod history occur at a fuel rod exposure of 21.2GWD/MT, which corresponds to a peak LHGR (steady state, 100% power) of 13.93 kw/ft. At these conditions, peak fuel centerline temperature was calculated to be 3909°F. To demonstrate margin to centerline melt during transients, power was escalated to 120% core power (LHGR=16.7 kw/ft) assuming heatflux in equilibrium with neutron flux. At the 120% overpower condition, centerline temperature was calculated to be 4607°F, demonstrating adequate margin to the UO_2 melting point at this exposure of about 4900°F. Since this calculation represents the most limiting point with respect to centerline melt throughout the life of the fuel, it is concluded that margin to centerline melt is assured for overpower conditions throughout the fuel lifetime.

Since the fuel rod history assumed in the above analysis is not enforced by the Technical Specification, an upper bound on the allowable peak LHGR (at full power) over the life of the fuel was determined by multiplying the proposed MAPLHGR limits for XN-1 fuel by a conservative value for the local peaking factor. Assuming this peak LHGR as an initial condition, power was again escalated to 120% to yield the maximum transient LHGR achievable from full power operation within the Technical Specification MAPLHGR limits. These maximum LHGR's were then compared to the LHGR corresponding to centerline melting (determined by ENC's RODEX2 code) for various exposures throughout the life of the fuel. This comparison showed that margin to centerline melting during overpower conditions initiated from full power is assured throughout the life of the fuel by Technical Specification MAPLHGR limits. Therefore, an LHGR Operating Limit has not been specified for ENC fuel.

For transients initiated from reduced power and flow, the peak transient LHGR could potentially exceed that of transients initiated from full power, if the power distribution is excessively peaked. Although existing restrictions on power distribution such as the reduced flow MCPR limits of Technical Specification 3.5.k and the RBM Rod Block will limit the peak LHGR and, therefore, the total core peaking at reduced flows.

C. Fuel Centerline Melting During Overpower Conditions (Cont'd)

supplemental protection will be provided by a daily surviellance on power distribution. This surveillance, performed in accordance with approved station procedures, will insure that the peak LHGR during reduced flow operation is limited so that fuel centerline melting would not occur during a transient terminating at the 120% flux scram. Performance of this surveillance as required by proposed Technical Specification 4.1.B.2, in conjuction with the proposed MAPLHGR limits for ENC fuel, will ensure that margin to centerline melt is maintained under all operating and transient conditions.

D. Nuclear Design

The XN-1 fuel design consists of 63 fuel rods and one water rod. The average assembly enrichment is 2.69% which includes a six inch natural U blanket at both top and bottom. The average enrichment of the central region (excluding blanket) is 2.87%. 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.

Attachment 2 also provides the results of the various routine cycle - specific analyses such as shutdown margin. core stability decay ratio and Standby Liquid Control System effectiveness.

<u>Fuel Storage Vault/Pool Criticality</u> - 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 $k_\infty < 1.31$ for the rack dimensions utilized in the Dresden spent fuel pool and < 1.30 for the rack dimensions utilized in the new fuel vault, where k_∞ is calculated in an infinite array of similar fuel in the core configuration (as opposed to the storage configuration). GE has calculated k_∞'s for their fuel designs and demonstrated that the criteria is satisfied for all GE fuel. ENC has calculated k_∞ for XN-1 reload fuel and for a comparable GE fuel design and shown that XN-1 fuel is slightly less reactive. Based on this comparison and the criteria from NEDE-24011, it is concluded that adequate margin to the Technical Specification keff limits exists for storage of XN-1 reload fuel in the vault and pool (for GE designed racks).

Fuel Storage Vault/Pool Criticality (Cont'd)

For the high density fuel storage racks designed by Nuclear Services Corporation, criticality analyses have been performed for ENC fabricated fuel which demonstrated that the 0.95 keff requirement is met. This assures that the XN-1 reload fuel will meet the .95 keff Technical Specification criteria when stored in the high density fuel racks. Refer to Kin W. Wong (NSC) testimony dated January 21, 1981 for the ASLB hearings on the Dresden Fuel Pool Modification.

II. TRANSIENTS AND ACCIDENTS

A. Anticipated Operational Occurrences (Transients)

In order to determine operating limits for D3C8, 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 D3C8.

- 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 addition, 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 Transients are provided in Attachments 2 and 3. The Generator Load Rejection without Bypass was determined to be the limiting event for D3C8, resulting in a ΔCPR of 0.25 which, when combined with the 1.05 Safety Limit, requires a MCPR operating limit of 1.30 for all fuel types.

Core-Wide Transients

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 LOFWH event was analyzed 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 for improved characterization of steam line dynamic behavior.

Core-Wide Transients (Cont'd)

Uncertainties in input parameters for the LOFWH and FWCF events were assumed to be at bounding values. For the limiting event, LRw/oB, uncertainties in the input variables were handled statistically as described in Attachment 3 and Reference 5. This results in a statistical distribution of Δ CPRs, arrived at by convoluting the uncertainty distributions of the input variables utilizing a Monte Carlo procedure. Using the mean value and standard deviations of the Δ CPR distribution, a Δ CPR of 0.25 was determined to bound 95% of the possible outcomes of the event.

Actual scram time data from previous cycles on Dresden 3 was used to generate the scram time distribution assumed in determining the CPR distribution for the LRw/oB event. In order to assure the applicability of the LRw/oB analysis to cycle 8 operation, compliance with the assumed scram time distribution must be verified throughout cycle 8 as required by proposed T.S. 4.3.C.3. Following each set of full or half-core (hot) scram testing, compliance with the assumed distribution must be demonstrated in accordance with the station procedures based upon information supplied by ENC. If the current cycle scram speeds deviate from the assumed distribution, an adjustment to the MCPR operating limit may be required. The ENC supplied methods for checking compliance and adjusting the MCPR operating limit (if necessary) will be incorporated in Station Procedures to ensure the proper MCPR operating limit is used throughout the cycle.

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 proposed Technical Specifications rod block monitor setpoint is 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 is 0.15. The CPR for the misloaded bundle event was 0.16 which was larger than the Δ CPR calculated for the misoriented bundle case (180° rotation). All of these Δ CPR's are less than the limiting value of 0.25 calculated for the LRw/oB event.

Reduced Flow Operation

ENC has reanalyzed the necessary adjustment in the MCPR operating limit for transients at reduced flow. Proposed Technical Specification 3.5.K incorporates ENC generated MCPR limits for reduced flow operation which protect the full flow MCPR Operating limit during Automatic Flow Control operation and the MCPR Safety Limit during all flow control modes. ENC's technical report describing the analyses for reduced flow operation will be submitted by (CECo) for your review in late January, 1982.

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 1346 psig or 108% of reactor vessel design pressure. The corresponding steam dome pressure was 1324 psig, for a vessel differential pressure of 22 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 1353 psig (1375-22). 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 criteria of 1375 psig peak vessel pressure while also maintaining consistency with the recently proposed Quad Cities Unit 2 pressure safety limit.

E. Postulated Accidents

In support of D3C8 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 3 is provided in Attachment 4 and summarized in Section 6 of Attachment 2. Operation within the MAPLHGR limits of Table 6.1 (Attachment 2) will ensure that the peak cladding temperature remains below 2200°F, local $Zr\text{-}H_2O$ 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 ENC 8x8 reload fuel and therefore provide MAPLHGR limits for ENC fuel only. 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 D3C8 and future cycles with GE/ENC mixed cores.

It should be noted that ENC MAPLHGR limits are provided as a function of bundle average exposure as opposed to nodal exposure for G.E. MAPLHGR limits. This is due to the different methodology employed by ENC for LOCA analyses. Axial exposure profiles which bound those expected during normal operation are input into the calculation of fuel rod stored energy, fission gas release and fuel rod heatup from which MAPLHGR limits are derived. Since conservative axial exposure profiles are inherent in the methodology, MAPLHGR limits as a function of assembly burnup will adequately protect the peak power plane. G.E. MAPLHGRs will remain as a function of nodal exposure.

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 calculates the fuel enthalpy rise during postulated RDA's over a wide range of reactor operating variables. Cycle specific parameters such as maximum control rod worth, Doppler coefficient, etc. are then applied to the parametric results to determine the fuel enthalpy rise. For D3C8, Section 6 of Attachment 2 shows a value of 151 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.

III. TECHNICAL SPECIFICATIONS

Attachment 6 provides proposed Technical Specification changes to support D3C8 operation with ENC 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 Operating Limit is specified for ENC fuel. Operation within the MAPLHGR limits and the power distribution assumptions of the Fuel Design Analyses will protect against fuel centerline melting during transients initiated from rated or less than rated conditions.

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 that margin to centerline melting is maintained during transients initiated from any allowable reactor conditions. For additional information regarding margin to centerline melt, refer to the previous discussion of Section I.C.

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 D3C8.

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 adjustmentfor 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.

*Also revised to allow adjustment of APRM gains in lieu of reducing trip settings.

C. MCPR

T.S. Section	Description
1.1.A 3.5.K	MCPR Safety Limit changed to 1.05 MCPR LCO changed to 1.30 for all fuel types. 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)

As discussed earlier, this will be changed from 1325 to 134_ psig. Previous value assumed a vessel pressure drop of 50 psi. The new value 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.

In addition, Specification 3.2.C.2 is being revised to clarify RBM operability requirements and to be consistent with Table 3.2.3 Note 1.

F. Section 3.5.D.3.a

This section, which allowed operation with only 4 ADS valves during Cycle 6 operation, has been deleted since it is no longer applicable.

G. MAPLHGR

Section	Description
3.5.I/4.5.I	Revised to distinguish that GE MAPLHGRs are functions of nodal exposure whereas ENC MAPLHGRs are dependent on bundle average exposure.

G. MAPLHGR (Cont'd)

Section	Description
Figure 3.5-1	Added MAPLHGR curve for ENC type XN8D2.69-5 fuel and deleted curves for 7x7 fuel. The labelling on the curve for fuel type 8DRB265-L has been revised to indicate that the curve applies to P8DRB265-L (Prepressurized) fuel also.

H. RPS Delay Time

Specification 3.1.A is being revised to change the allowable RPS delay time from 100 msec. to 50 msec to be consistent with the delay time used in ENC's deterministic transient analyses (refer to I.E. Circular 80-08). In the analysis of the Load Rejection without Bypass event, a conservative statistical distribution of the RPS delay time was used, rather than a single value as in the deterministic approach.

I. 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.

IV. REFERENCES

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

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

3. XN-NF-79-71 Revision 1 (Supplements 1 and 2), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors" dated November, 1981.

4. XN-NF-80-19(P), 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(P), October 1981
Generic Jet-Pump BWR3 LOCA Analysis Using the ENC EXEM Evaluation Model.

7. XN-NF-81-75, "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model-MAPLHGR Results" dated October, 1981
(Attachment 4).

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

9. XN-NF-80-19(P), 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(P), 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(P), 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

REFERENCES (Cont'd)

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

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

0168T

ATTACHMENT 5

AFFIDAVIT

STATE OF Washington)
ss.
COUNTY OF Benton)

I, James N. Morgan, 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 document XN-NF-81-75(P), entitled "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model - MAPLHGR Results," referred to as "Document". Information contained in this Document 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 Document contains 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 Document has 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 Document contains 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 Document is considered to be proprietary by ENC because it reveals certain distinguishing aspects of ECCS analytical 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 Document.

8. The disclosure of the proprietary information contained in the Document 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 reactor core operating characteristics, and would result in substantial harm to the competitive position of ENC.

9. The Document contains 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 Document 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. This Document provides information which reveals ECCS analytical methods developed by ENC over the past several years. ENC has invested hundreds of thousands of dollars and many man-years of effort in developing the analysis methods and calculating the results revealed in the Document. 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 Document.

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

FURTHER AFFIANT SAYETH NOT.

James F. Mallay

SWORN TO AND SUBSCRIBED
before me this 20 day of
November, 1981.

NOTARY PUBLIC