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August 25, 1983

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
Facility Operating License DPR-25
NRC Docket No. 50-249

References (a): T. J. Rausch letter to H. R. Denton dated December 21, 1982 (Dresden Unit 2 Proposed Amendment).

(b): D. M. Crutchfield letter to D. L. Farrar dated April 7, 1983.

(c): B. Rybak letter to H. R. Denton dated July 18, 1983 (ASEA-Atom Proposed Amendment).

Dear Mr. Denton:

Pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend Appendix A to the Technical Specifications of Facility Operating License DPR-25 to allow for the following:

(1) A revision of the MAPLHGR curves for Dresden Unit 3, Cycle 9 (D3C9).

(2) Replacement of the K_f curve with Exxon Nuclear Corporation's (ENC's) reduced flow MCPR limits.

(3) An administrative change to the bases of the reactor coolant safety limit specification, resulting from an oversight in the last (Cycle 8) submittal.

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Please note that the MAPLHGR limit curves proposed for both the XN-1 reload fuel (Cycle 8) and XN-2 reload (Cycle 9) were generated based on the current version of RODEX-2 which is now under NRC review. Assuming the NRC approves this version of RODEX-2, no additional confirmatory calculations will be required. Other parameters calculated with RODEX-2 (fuel temperature, internal rod pressure, corrosion and transient strain, etc.) will be confirmed upon completion of NRC's review of RODEX-2.

Appendix B to XN-NF-83-47, D3C9 Reload Analysis (Attachment 2) describes the licensing impact of the eight ASEA-Atom demonstration control blades. Use of these blades is contingent on NRC approval of the proposed Technical Specification amendment to allow use of Hafnium as a neutron absorber material (Reference (c)). Early completion of the Staff review is needed to allow blade management alternatives to be implemented in the unlikely event that the A-A blades cannot be utilized.

Exxon Nuclear Corporation (ENC) has performed transient analyses and a sensitivity study to confirm that the D3C8 CPR response surface is applicable to Cycle 9 as required by the NRC's Safety Evaluation Report for D3C8. As a result, the same MCPR operating limits will be retained. These operating limits include the 0.03 CPR penalty imposed for Cycle 8 to accomodate potential uncertainties in ENC's COTRANSA code. In addition, the MCPR Safety Limits in the Technical Specifications are also retained, which includes a 0.01 CPR adder for GE 8x8 retrofit fuel as required by the NRC for Cycle 8.

A summary of the reload package is contained in Attachment 1. The Dresden 3 Cycle 9 reload analysis, plant transient analysis, and LOCA analysis are contained in Attachments 2, 3, and 4, respectively. The proposed changes to the Technical Specifications are enclosed in Attachment 5 and have received both On-Site and Off-Site approval.

We have reviewed these proposed changes and find that no significant hazard consideration exists. Our review is documented in Attachment 6.

Pursuant to 10 CFR 170, Commonwealth Edison has determined that the proposed amendment change is Class III. As such, a fee remittance in the amount of $4,000 has been included.

Commonwealth Edison is notifying the State of Illinois of our request for this amendment by transmittal of a copy of this letter and its Attachments.

Please address any questions you may have on this matter to this office.

Three (3) signed originals and thirty-seven (37) copies of this transmittal are enclosed for your use.

Very truly yours,

B. Rybak

B. Rybak
Nuclear Licensing Administrator

lm

cc: R. Gilbert - NRR
NRC Resident Inspector - Dresden
G. N. Wright (Illinois)

Attachments (1): Discussion of Reload and Proposed Changes

(2): Dresden Unit 3 Cycle 9 Reload Analysis XN-NF-83-47

(3): Dresden Unit 3 Cycle 9 Plant Transient Analysis - XN-NF-83-58

(4): Dresden Unit 3 LOCA Analysis Using ENC EXEM Evaluation Model MAPLHGR Results; XN-NF-81-75, Supplement 1

(5): Proposed Amendment Change

(6): Significant Hazards Consideration.

SUBSCRIBED AND SWORN to
before me this 25th day
of August, 1983

Rosalie A. Punta
Notary Public

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ATTACHMENT 5

Proposed Changes to DPR-25

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ATTACHMENT 1

Dresden 3 Cycle 9 Reload XN-2

Dresden 3 Cycle 9 (D3C9) represents the second reload of Exxon Nuclear Company (ENC) fuel in Unit 3, and the first U.S. BWR to use control blades designed by ASEA-ATOM of Sweden (A-A). The reload fuel is identical to the NRC approved design that was loaded into Dresden 2 Cycle 9 with one minor change as described in Section I.A. of this attachment. This evaluation addresses fuel and core design, the A-A control blade design, reload transient analyses, and Technical Specification changes supporting D3C9 Reload XN-2. 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 3 Cycle 9 Reload Analyses, XN-NF-83-47 (Attachment 2), the Dresden 3 Cycle 9 Plant Transient Analysis Report XN-NF-83-58 (Attachment 3) and the Dresden 3 LOCA Analysis MAPLHGR Results XN-NF-81-75(P) (Attachment 4) and are submitted to the NRC in support of D3C9 operation. Section III provides proposed Technical Specification changes required for Cycle 9 operation. A summary follows in Section IV.

I. RELOAD FUEL AND CORE DESIGN

Dresden 3 Reload XN-2 will consist of 184 ENC 8x8 reload assemblies designated as type XN8D2.83-5, 224 once irradiated type XN-1 fuel assemblies, and 316 General Electric 8x8 assemblies. A summary of the major design features follows.

A. Fuel Mechanical Design

The mechanical design of the XN-2 8x8 reload fuel, described in Reference 1, is identical to the design approved for D2C9 except that the pellet L/D ratio is slightly smaller for 177 of the XN-2 reload assemblies. Of the 184 ENC reload bundles, seven were initially produced for use in D2C9 but were not used and are therefore being inserted in D3C9. These seven bundles are therefore identical to the D2C9 reload fuel. The following discussion of reduced pellet L/D refers to the remaining 177 reload bundles fabricated for D3C9.

The fuel pellet length-to-diameter (L/D) ratio has been reduced from 1.15 for the D2 XN-1 design to 1.0 for most of the D3 XN-2 reload fuel (some of the bundles will also contain pellets originally manufactured for D2 which have the 1.15 L/D). This change was made at the request of CECo to reduce cladding stresses

caused by pellet-clad interaction. ENC has determined that this change in pellet L/D has minimal impact on fuel design limits and does not change the conclusions in the Generic Fuel Design Report (Reference 1). Also note that the change in pellet length resulted in a small increase in the pellet dish volume. CECo considers the reduced L/D for D3 XN-2 fuel to be conservative and acceptable.

B. Thermal Hydraulic Design

Thermal hydraulic compatibility of ENC and GE fuel has been demonstrated previously for D3C8 and is also assured for Cycle 9 since the thermal hydraulic design of ENC reload fuel has not been changed for the XN-2 reload.

Analyses made during the calculation of the Fuel Cladding Integrity Safety Limit demonstrated that for all Dresden 3 fuel types, 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 transition boiling during steady state operation at the safety limit. For D3C8, the NRC required a 0.01 adder to the MCPR Safety Limit for the GE retrofit fuel (refer to NRC SER for D3C8). This adder will be retained for Cycle 9; therefore, the current Technical Specification MCPR Safety Limits of 1.05 for ENC 8 x 8, GE 8 x 8 and 1.06 for GE 8 x 8R remain applicable.

Revised hydraulic input parameters (particularly inlet orifice loss co-efficients) have been used in the D3C9 analyses. The change in these hydraulic parameters resulted from an ENC investigation into differences between General Electric pressure drop information available in the open literature and the original pressure drop data from ENC's hydraulic test facility. ENC determined that their original modeling of the lower tie plate (LTP) and inlet orifice was not sufficiently detailed and resulted in loss coefficients lower than those indicated by the open literature. ENC revised their model of the inlet orifice and LTP and performed additional pressure drop testing which confirmed the open literature values are applicable to ENC fuel. As a result, the revised (higher) loss coefficients are being used for D3C9. The primary effect of higher loss coefficients for the inlet orifice/LTP was to significantly decrease decay ratios from the core-wide stability analyses. Use of the revised hydraulic parameters as input data to the COTRAN code resulted in more stable behavior calculated for D3C9 than in the previous cycle. Refer to Section I.D. of this evaluation.

C. Fuel Centerline Melting

One of ENC's primary thermal hydraulic design criteria is that fuel centerline melting will not occur for anticipated transients throughout the life of the fuel. ENC has previously performed transient overpower analyses for their reload fuel design to demonstrate compliance with this design criterion. The results previously reported for 8x8 XN-1 fuel are applicable to the D3 XN-2 fuel. The results provided in Section 2 of the D3C9 Reload Analysis Report (Attachment 2) indicate a 293°F margin to centerline melting for the ENC reload fuel.

D. Nuclear Design

The 8x8 XN-2 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 the 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 neutronics design parameters and pin enrichment distribution are provided in Section 4 of Attachment 2. Note that the neutronic design of the D3 reload XN-2 fuel is identical to the D2C9 reload XN-1 fuel.

Core Reactivity

As reported in Attachment 2, the calculated BOC9 cold core k with all rods in is 0.949 and the value with the strongest rod out is 0.985. This results in a calculated shutdown margin of 1.46% K at the most reactive point in the cycle. The Standby Liquid Control System, with a boron concentration of 660 ppm in the reactor water, was calculated to provide a shutdown margin of 5.8% K for cold conditions with all rods in their full power positions.

Core Stability

The reactor decay ratio as a function of percent power is given in Figure 4.3 of the D3C9 Reload Analysis (Attachment 2). The D3C9 decay ratio at natural circulation and the 100% Flow Control Line is calculated to be 0.33.

The decay ratios depicted in Figure 4.3 of Attachment 2 are lower than the respective values reported for D3C8. Based on discussions with ENC technical personnel, the lower Cycle 9 decay ratios result from using revised hydraulic input data in the Cycle 9 COTRAN analysis of core stability.

E. ASEA-ATOM Control Blades

Dresden 3 Cycle 9 will be the first U.S. BWR to use ASEA-ATOM control blades. As part of an EPRI-sponsored demonstration, eight ASEA-ATOM control blades will be inserted in single rod sequencing locations during Cycle 9. The unique design should enhance blade lifetime and reduce the problems of blade cracking currently experienced with standard GE blades. The impact of the A-A blades on D3C9 Licensing calculations is described in Appendix B of Attachment 2.

Four of the A-A blades will contain a single control zone wherein the primary control material will be B_4C. The remaining four blades will contain two control zones: a zone of B_4C and a second zone containing hafnium as the primary control material. This second zone is located at the top six inches of the blade. To allow for the use of hafnium as a control blade absorber material, a Technical Specification change has been proposed independent of this licensing submittal.

CECo has reviewed ENC's evaluation of the impact of the A-A blades on D3C9 Licensing Analyses (Appendix B to Attachment 2) and concludes that the A-A blades have been adequately treated in the licensing analyses and will not impact Cycle 9 operating limits.

II. TRANSIENTS AND ACCIDENTS

A. Anticipated Operational Occurrences (Transients)

ENC considers eight categories of potential core-wide transients to determine limiting events for cycle specific analyses. The following three events were determined to be most limiting and were analyzed to determine the MCPR Operating Limit for D3C9:

1. generator load rejection without bypass (LRw/oB)
2. feedwater controller failure (FWCF)
3. loss of feedwater heating (LFWH).

In addition, two local events - Rod Withdrawal Error (RWE) and Fuel Loading Error - were analyzed and determined to be non-limiting.

The results of the core-wide and local transient analyses are provided in Attachment 2 (XN-NF-83-58, D3C9 Plant Transient Analysis). A discussion of the Cycle 9 transients analyses and resultant MCPR Operating Limit is provided below.

Core-Wide Transients

Table 5.1 of Attachment 2 summarizes the results of the core-wide transient analyses performed for D3C9. As in previous cycles, the LRw/oB event is the most limiting and establishes the Cycle 9 MCPR operating limits. This event was analyzed statistically for D3C8 to determine a CPR which bounds 95% of the possible outcomes of a LRw/oB event. In the statistical evaluation, the uncertainty distributions of four of the most important input variables to the transient analyses are statistically convoluted to determine a CPR response surface. This response surface, generated for D3C8, represents the statistical variation in CPR resulting from the combined uncertainties of the input parameters. The D3C8 MCPR operating limit was established by determining the CPR from the response surface which bounds 95% of the possible outcomes and adding it to the MCPR Safety Limit.

A substantial number of calculational runs using ENC's CONTRANSA code and a Monte Carlo procedure were performed to generate the D3C8 response surface. As noted in the NRC's Safety Evaluation Report for D3C8, this same response surface can be used for establishing D3C9 MCPR limits (in which case the Cycle 9 limits would be identical to Cycle 8) if it can be shown to bound Cycle 9 transient results. Otherwise a new response surface would need to be generated for D3C9. In Section 3.2.1 of Attachment 3, ENC describes the evaluation they performed to verify that the Cycle 8 response surface is applicable to D3C9.

The verification consisted of a review of D3C8 scram time data and scram valve opening times to check for consistency with the assumptions used in generating the D3C8 response surface. ENC determined that the average 90% insertion scram times for Cycle 8 had degraded slightly compared to the data used in generating the response surface as did the scram time delay. To determine the impact of the degraded scram performance on the transient analyses, ENC performed a LRw/oB analysis using D3C8 scram performance data with D3C9 neutronic and thermal hydraulic parameters*. The resultant CPR was compared to the results from a corresponding D3C8 transient run and found to be slightly lower. This indicates that the degraded scram performance observed in Cycle 8 is offset by the less negative void reactivity for D3C9; therefore, the Cycle 8 response surface will bound D3C9 transient results. ENC also reviewed the spread (distribution) of Cycle 8 scram times and scram valve opening times, and performed sensitivity studies to confirm that the standard deviation of the CPR distribution (response surface) was unchanged. Based on these evaluations, ENC has concluded that the D3C8 response surface is applicable for D3C9, hence the Cycle 8 CPR of 0.25 is used to establish the Cycle 9 MCPR Operating limits.

Using the CPR and the previously described MCPR Safety Limits, the Cycle 9 MCPR limits would be 1.30 for ENC 8 x 8 and GE 8 x 8 fuel; and 1.31 for GE 8 x 8R fuel. Note that the Cycle 9 limits identified in Attachments 2 and 3 do not include the NRC imposed 0.01 adder to the 8 x 8R Safety Limit, and therefore indicate a value of 1.30 for all fuel types. During the review of D3C8, the NRC imposed an additional 0.03 MCPR penalty to the CPRs for all fuel types due to COTRANSA uncertainties. Since this issue has not yet been resolved, the 0.03 penalty will be retained for Cycle 9, resulting in MCPR Operating Limits of 1.33 for ENC 8 x 8 and GE 8 x 8; and 1.34 for GE 8 x 8R fuel. These values are currently in the Dresden 3 Technical Specifications, therefore no changes to the full flow MCPR limit is required for Cycle 9 operation.

Since the Cycle 8 response surface is being applied to Cycle 9, the scram time conformance criteria (T.S. 3.5.K) remains the same as for Cycle 8. This criteria, 2.58 seconds, is provided in Appendix A to Attachment 2 along with the associated equation for adjusting the MCPR limit if the criteria is exceeded. Both the criteria and the equation are currently in the D3 Technical Specifications, therefore no change is required.

<u>Rod Withdrawal Error (RWE)</u> - The CPR for the RWE event with a 100% full flow RBM setpoint is 0.16, much less than the CPR for the LRw/oB event (0.25). Therefore, RWE is a non-limiting event for D3C9.

* This run assumed two relief valve setpoints at 1115 psig and three at 1135 psig. The proposed setpoints for the two valves is 1112 psig, so assuming 1115 psig for the ENC analysis is conservative.

Fuel Loading Error (FLE) - As shown in Attachment 2, the results for the FLE event are bounded by the LRw/oB event and therefore FLE is not limiting for D3C9. The CPR for the FLE event is 0.19.

Reduced Flow Operation - ENC has provided MCPR operating limits for reduced flow operation using manual and automatic flow control in Attachment 2. These limits are based on ENC's "Dresden Unit 3 Analyses for Reduced Flow Operation", (Reference 2) and were submitted to the NRC as proposed Technical Specifications for D3C8. However, the ENC reduced flow limits were not NRC approved for D3C8 due to inadequate time to complete their review of Reference 2.

As a result, the General Electric K_f curves were retained in the D3 Technical Specifications to provide reduced flow MCPR protection for Cycle 8. The NRC has since completed their review of Reference 2 and has approved the ENC reduced flow curves for Dresden 2. Therefore, we are now re-submitting the reduced flow MCPR limit curves for incorporation in the Dresden 3 Technical Specifications.

ASME Overpressurization Analysis - In order to demonstrate compliance with the ASME code overpressurization criteria of 110% of design pressure, the MSIV closure event with failure of the MSIV position scram was analyzed with ENC's COTRANSA code (Attachment 3). The maximum pressure observed in the analysis is 1347 psig or 108% of reactor vessel design pressure. The corresponding steam dome pressure is 1323 psig which is less than the Technical Specification limit of 1345 psig. This analysis includes the effect of the ATWS RPT which was assumed to initiate at a nominal pressure set point of 1240 psig.

B. Postulated Accidents

In support of D3C9 operation, ENC has reanalyzed the Loss of Coolant Accident (LOCA) to determine MAPLHGR limits for XN-2 fuel and has reanalyzed 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.

Loss of Coolant Accident - The D3C9 LOCA analysis is described 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^{o}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 8 x 8 reload fuel and therefore provides

MAPLHGR's for ENC fuel only. As discussed previously, ENC reload fuel is hydraulically and neutronically compatible with GE fuel. Therefore, the existing GE LOCA Analysis and MAPLHGR limits will remain applicable during D3C9 and future cycles with GE/ENC mixed cores.

A change is currently proposed to the Dresden 3 Technical Specifications to incorporate the MAPLHGR curves for the XN-2 reload fuel type XN8D2.83-5. The new MAPLHGR curves were generated using the RODEX2 code developed by ENC. The RODEX2 code is currently being reviewed by the NRC and approval is expected in September. Further information on this Technical Specification change can be found in Section III of this report.

Rod Drop Accident - For D3C9, Section 6 of Attachment 2 shows a value of 85 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

A. Proposed Changes

The purpose of the proposed D3C9 changes is to update the Technical Specifications to reflect:

a. a revision of the MAPLHGR curves for D3C9;

b. replacement of K_f curve with ENC's reduced flow MCPR operating limits for manual and automatic flow control.

c. an administrative change to the bases of the reactor coolant safety limit specification, resulting from an oversight in the D3C8 Technical Specification changes.

These changes are provided in Attachment 6 and are discussed in the following sections.

MAPLHGR Curves - The proposed change to Sheet 1 of Figure 3.5-1 provides a label change to include the XN-2 reload fuel type and eliminate the caveat concerning approval of the RODEX2 code. The MAPLHGR limit for the Dresden 3 XN-1 and XN-2 fuel types have been generated by a new ECCS analysis (see Attachment 4) based on the most recent revision of ENC's RODEX2 code. The version of RODEX2 used in the D3C9 analyses is currently under NRC review and is expected to receive approval in September. Assuming approval is forthcoming, the note concerning the validity of the MAPLHGR results for burnups exceeding 10,000 MWD/MT is irrelevant and may be removed.

MCPR Limits at Reduced Flow - The current K_f curve in the Dresden 3 Technical Specifications is being replaced with ENC supplied curves of MCPR operating limits for manual and automatic flow control reduced flow operation. The figures are taken directly from the D3C9 Reload Analysis (Attachment 2) and were calculated using the previously approved flow-iterative formulation of the critical power ratio.

Bases of Reactor Coolant Safety Limit Specification - The D3C8 reload licensing submittal processed a large number of Technical Specification changes. Due to an administrative error, appropriate changes to the Technical Specification bases for reactor coolant pressure safety limit (Bases 1.2) were omitted. These changes were approved by On-Site and Off-Site Reviews and should have been included in the D3C8 submittal to the NRC. They are now being re-submitted for consistency in specification limits and bases.

IV. SUMMARY

The preceeding discussion has addressed all major features of the XN-2 reload for D3C9. The mechanical, thermal hydraulic, and neutronic design of XN-2 fuel is compatible with GE and XN-1 fuel. CECo concludes that operation of D3C9 with XN-2 reload fuel is safe and acceptable provided the NRC approves the proposed Technical Specification changes of Attachment 6.

ATTACHMENT 6

Significant Hazards Consideration

Commonwealth Edison proposes to amend Provisional Operating License DPR-25 and reload the Dresden 3 reactor core in preparation for Cycle 9 operation. The proposed Technical Specifications changes do not represent significant changes in acceptance criteria or safety margins and all changes have been made based on methods that have been previously accepted by the NRC. The reload core involves no fuel assemblies significantly different from those found previously acceptable to the NRC for previous reloads at Dresden Units 2 and 3.

Under the provisions of 10CFR50.92 this means that the proposed amendment will not (1) involve a significant increase in the probability or occurrence of an accident previously evaluated; or (2) create the possibility for a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in the margin of safety.

The Technical Specification changes are proposed to incorporate new MAPLHGR and MCPR curves based on Exxon Nuclear Corporation's analysis of plant transient events, and to make changes to the technical bases which were inadvertently omitted in the Dresden 3 Cycle 8 licensing submittal to the NRC. The new MAPLHGR curves are based on calculations using ENC's approved ECCS methodology and the RODEX2 fuel rod performance code which is currently being reviewed by the NRC. The proposed MAPLHGR limits for the Cycle 9 reload are identical to the previously approved MAPLHGR values for the D3C8 reload.

The replacement of the existing Technical Specification K_f curves with ENC supplied MCPR curves is based on the results of ENC's previously approved analyses for reduced flow transient events. The change does not represent a significant alteration in operations at the plant; in fact, the proposed MCPR curves are identical to those previously approved and currently in use for Dresden Unit 2.

Administrative changes to the technical bases of the reactor coolant specification do not impact plant operation at all and were originally intended to be submitted to the NRC with the D3C8 license submittal. The changes are made only to the technical bases and are required to provide consistency between Technical Specification limits and bases.

Finally, the Commission itself has determined that reload fuel which is not significantly different from previously accepted designs conforms with the standards of 10CFR50.92 as indicated by the Examples in the Federal Register. As in the Examples, Dresden 3 Cycle 9 reload fuel assumes no significant change in the mechanical design from that used in Cycle 8; the thermal hydraulic design evaluation remains bounded by the FSAR and previous reloads; and the thermal performance of the core during accidents and transients for the Cycle 9 reload remains within the bounds of previously accepted analyses.

Based on the preceeding discussion, it is concluded that the consequences of previously evaluated accidents will not be increased and the margin of safety will not be decreased by the proposed D3C9 reload and associated Technical Specification changes. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10CFR50.92(e), the proposed changes do not constitute a significant hazards consideration.

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REFERENCES

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

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

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