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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-249

1.0 INTRODUCTION

By letter dated March 9, 1988 (Ref. 1), Commonwealth Edison Company (CECo) proposed to amend Appendix A and Section 3.E of Facility Operating License No. DPR-25 to support Cycle 11 operation of Dresden Unit 3 with an entire core of Advanced Nuclear Fuels (ANF) fuel. In a letter dated June 17, 1988, CECo submitted two Technical Specifications pages that were inadvertently deleted from the original submittal. These pages were related to reactor operation with relief valves out of service. The March 9 submittal addressed all aspects of the changes involved related to the relief valves except for the inadvertently deleted pages. All residual General Electric (GE) fuel is scheduled to be discharged during the end-of-cycle (EOC) 10 outage. The requested amendment furnished information to support (1) use of ANF 9x9 fuel with axially zoned burnable absorber (Gd_2O_3) rods, (2) modified limits for single loop operation (SLO) based on ANF analyses, (3) provisions for extended operation with a relief valve out-of-service, and (4) operation, including coastdown, with reduced feedwater heating.

In support of the Dresden 3 Cycle 11 (D3C11) reload CECo submitted topical reports which described the reload analysis (Ref. 2), the plant transient analysis (Ref. 3), analysis of operation with one relief valve out-of-service (Ref. 4), and the LOCA-ECCS analysis during SLO with ANF fuel (Ref. 5).

2.0 EVALUATION OF RELOAD

2.1 Reload Description

The D3C11 reload will include 72 fresh ANF 9x9 fuel assemblies designated XN-4H and 96 fresh ANF 9x9 fuel assemblies designated XN-4L. These assemblies have a central region enrichment of 3.35 weight percent U-235 and 6 inch natural uranium ends to yield an average assembly enrichment of 3.13 weight percent U-235. The remainder of the core is comprised of 556 previously irradiated ANF fuel assemblies designated XN-1 (8x8), XN-2 (8x8), XN-3 (9x9), and XN-3A (9x9). The core will be operated under the Single Rod Sequence (SRS) control strategy to assure that the control rod withdrawal error will not be limiting.

2.2 Fuel Design

The mechanical design of the XN-4H and XN-4L 9x9 reload fuel is described in References 6 and 7. The ANF fuel to be returned to the Dresden 3 core has been approved for operation in previous cycles. The XN-4H and XN-4L fuel assemblies are identical except for a difference in the Gd_2O_3 concentration in the central region of the gadolinia-bearing fuel rods. Both assembly types contain nine gadolinia-bearing fuel rods with 3.0% Gd_2O_3 in the top six inches and bottom 12 inches of the enriched region of these rods. The central region of the gadolinia-bearing rods in the XN-4H assemblies contains 4.5% Gd_2O_3 whereas the XN-4L assemblies contain 4.0% Gd_2O_3 in this region. Both fuel types contain 79 fuel rods (8 are tie rods) and two water rods. Based on the previous review of the generic submittal (Ref. 6) and the information submitted with the D3C11 reload application, the staff finds the mechanical design of the ANF fuel for the D3C11 reload is acceptable.

During the review of the Cycle 10 reload submittal, the staff placed an exposure cap on 8x8 and 9x9 fuel due to rod bow considerations. The limit was set at 30,000 MWD/MTU for 8x8 fuel and 23,000 MWD/MTU for 9x9 fuel (batch average exposure). The expected peak assembly exposures at the end of Cycle 11 are 34,400 MWD/MTU and 23,500 MWD/MTU for the 8x8 and 9x9 assemblies, respectively. Based on additional information on rod bowing (Ref. 8) which has been reviewed by the staff and on the staff's safety evaluation of XN-NF-82-06(P), Supplement 1, Revision 2 (Refs. 9 & 10), these expected Cycle 11 fuel exposures are acceptable.

2.3 Thermal-Hydraulic Design

Single phase flow tests of full scale assemblies have been performed in order to determine the component hydraulic resistances for the D3C11 fuel types. Based on the similar hydraulic performance illustrated in hydraulic demand curves for ANF 8x8 and ANF 9x9 fuel (Ref. 2), the staff concludes that the two fuel types are hydraulically compatible.

The XN-3 correlation used to develop the minimum critical power ratio (MCPR) safety limit has been approved for application to both the ANF 8x8 and the new 9x9 fuel type (Refs. 11 & 12). ANF has calculated the MCPR safety limit to be 1.05 for all fuel types in the D3C11 core. Since the calculations considered each of the constituent fuel types, conservative local power distributions for each type, the worst (bounding) radial power distribution at which each fuel type is expected to operate and used approved methodology (Ref. 13), the staff finds the safety limit acceptable for all Cycle 11 fuel types. The proposed operating limit MCPR for D3C11 is 1.39, which is the same value as for current (Cycle 10) operation, and bounds the delta-CPR results of the limiting plant transients as discussed in Section 2.5 of this safety evaluation.

The thermal-hydraulic stability of the Cycle 11 core was analyzed using the methods identified in XN-NF-80-19(P)(A), Volume 4, Revision 1 (Ref. 14). Reference 14 cites the use of the COTRAN and COTRANSA models for use in the analysis of core thermal-hydraulic stability. The resultant maximum decay ratios for natural recirculation flow determined analytically using the approved COTRAN code at various power and flow conditions are 0.35 (47.6% rated power and 31.5% rated flow) at the 100% flow control line (FCL) and 0.55 (58% rated power) at the average power range monitor (APRM) rod block intercept. Since both of these decay ratios are less than the surveillance criterion of 0.75 as calculated by COTRAN, no stability Technical Specification surveillance requirement is needed for Cycle 11 operation. A test comparing stability between dual loop operation (DLO) and single loop operation (SLO) was performed during Cycle 10 from which it was concluded that the operating region of concern exhibits adequate margin to power/flow instabilities in SLO and DLO. Although the staff concurs that present positions indicate stability monitoring Technical Specifications are not required during DLO, staff positions regarding calculated acceptable decay ratios and Technical Specifications requirements described in NRC Generic Letter 86-02 are under review due to the recent LaSalle instability event. Any new staff findings as a result of this review will likely be applied generically to all BWRs including Dresden 3. In addition, since ANF 9X9 fuel has not yet received generic approval by the staff, there is the possibility of additional stability testing as the amount of 9X9 fuel in the Dresden 3 core increases in future reloads. For SLO, the current stability surveillances required by Technical Specifications are adequate for detecting any core wide or local instabilities. Therefore, the thermal-hydraulic design of Cycle 11 is acceptable.

2.4 Nuclear Design

The nuclear design for D3C11 has been performed with ANF methodologies previously reviewed and approved (Ref. 15). The fuel loading pattern is given in Figure 4.4 of Reference 2. The beginning-of-cycle (BOC) shutdown margin is 1.24% delta-k and at minimum conditions is 1.11% delta-k, well in excess of the required 0.42% delta-k. The standby liquid control system (which is designed to inject a quantity of boron solution that produces a concentration of no less than 600 ppm of boron in the reactor core) was calculated to provide a shutdown margin of 6.19% delta-k for cold conditions with all control rods in their full power positions. This meets the shutdown margin requirement of 3.0% delta-k and is, therefore, acceptable. Since these results have been obtained by previously approved methods and meet the appropriate requirements, the staff concludes that the nuclear design of Cycle 11 is acceptable.

For D3C11, there will be 12 ASEA-ATOM (A-A) control blades designated CR-82B inserted which are similar to the CR-82 design previously generically reviewed and approved by the NRC (Ref. 16). These new control blades incorporate several enhancements compared to the CR-82 design and do not have any significant impact on the mechanical characteristics. The staff, therefore, finds them acceptable for use in Cycle 11.

2.5 Transient and Accident Analyses

Corewide transients were analyzed with the same methodology used to establish thermal margin requirements for Cycle 10 operation (Refs. 17 & 18). The XCOBRA-T hot channel model was used to calculate the delta-CPR values. The XCOBRA-T model has been reviewed by the staff and found to be acceptable (Ref. 19).

The licensee evaluated several categories of potential corewide transients for Cycle 11 and provided specific results for three transients, generator load rejection without bypass (LRWB), feedwater controller failure (FWCF), and loss of feedwater heating (LFWH). The limiting transient is identified as the LRWB. Since this limiting transient is a rapid pressurization event, the ANF methodology for including uncertainties in determining operating limits for rapid pressurization transients in BWRs (Refs. 20 & 21) was applied. This methodology uses a conservative deterministic multiplier of 110% on the calculated transient power to account for COTRANSA code uncertainties and treats the uncertainties in the important input variables (scram speed and scram delay) statistically. At rated power, the delta-CPR was 0.23 for ANF 8x8 fuel and 0.26 for ANF 9x9 fuel for the LRWB transient. Therefore, the delta-CPR results of the analyses for the limiting corewide transients are acceptably bounded by the proposed Cycle 11 MCPR limiting condition of operation (LCO) of 1.39.

The most limiting event for reactor vessel over-pressurization is the main steamline isolation valve (MSIV) closure without direct scram (single failure) on valve position. The maximum value of the sensed pressure in the steam dome was 1297 psig which corresponds to a maximum vessel pressure of 1324 psig at the lower plenum. These values are less than the Technical Specification limit of 1345 psig as measured by the steam dome pressure indicator and the 1375 psig ASME vessel pressure limit. This is acceptable.

The licensee has also evaluated the effect of a relief valve out-of-service (RVOOS) on the plant transients (Ref. 4). The results indicate that with one RVOOS there is no effect on delta-CPR calculated for the limiting transients and an insignificant effect on peak pressure for all fuel types in Cycle 11.

The licensee has determined the required reduced flow MCPR operating limit for off-rated conditions to complement the Cycle 11 MCPR full flow operating limits during the automatic flow control (AFC) condition and in manual flow control (MFC). The results are given in Tables 5.4, 5.5, and 5.7 of Reference 3 and are acceptable.

For the control rod withdrawal error (RWE) local transient, the licensee has determined that a rod block monitor (RBM) upper setting of 110% of full power results in a delta-CPR of 0.31 for 9x9 fuel and 0.30 for 8x8 fuel. The Technical Specification MCPR LCO of 1.39 for both fuel types in Cycle 11, therefore, bounds the RWE results.

Analyses with a feedwater heater out-of-service (FHOOS) were also performed to support coastdown operation for EOC 11. The results show that the delta-CPRs for the transients analyzed with a FHOOS are bounded by the delta-CPRs for transients at normal feedwater temperature (Ref. 3).

The licensee also evaluated the control rod drop accident (RDA) and the loss of coolant accident (LOCA) which are described as follows.

The RDA evaluation yields a value of 187 cal/gm for the maximum deposited fuel rod enthalpy. This is well below the NRC required limit of 280 cal/gm, and is, therefore, acceptable.

ANF has previously performed LOCA analyses for Dresden 3 using 8x8 fuel (Ref. 22) and 9x9 fuel (Ref. 23) which provided maximum average planar linear heat generation rate (MAPLHGR) limits. These limits remain applicable for the fuel in Cycle 11 during dual loop operation. ANF has also evaluated the effect of a RVOOS on the MAPLHGR limits (Ref. 4). The limiting postulated small break LOCA was analyzed since relief valves do not actuate in large breaks. Based on the results of this latter analysis, MAPLHGR multipliers of 0.89 and 0.76 were calculated for 8x8 and 9x9 fuel types, respectively. The results of a LOCA analysis for Dresden 3 during SLO (Ref. 5), which were performed by ANF using the generically approved EXEM/BWR Evaluation Model, established the multiplier to be applied to the MAPLHGRs of the ANF fuel during SLO. These results support a MAPLHGR multiplier of 0.91 for all fuel types in the Cycle 11 core during SLO. The LOCA analyses were performed with reviewed and accepted methods and the results are well within the limits of 10 CFR 50.46. Therefore, the staff concludes that the MAPLHGR limits proposed for Cycle 11 are acceptable.

2.6 Extended Load Line Limit Analysis (ELLLA)

The extended load line limit analysis (ELLLA) provides a basis to support plant normal operation in the region of the power/flow map above the 100% power/100% flow load line and bounded by the 108% APRM rod block line and the 100% rated power line. This added capability increases operating flexibility to permit flow compensation for xenon buildup following startups and for fuel depletion later in cycle, and to improve the efficiency of achieving and maintaining 100% power. The results of the previous ELLLA performed by ANF as part of the Cycle 10 reload analyses are also applicable to Cycle 11 since the cycle specific analyses for Cycle 11 have been performed consistently with respect to power/flow region assumptions. It is concluded that changes in core behavior caused by the extended operating range have been acceptably accounted for in D3C11.

2.7 Single Loop Operation

Current Technical Specifications for Dresden 3 permit plant operation with a single recirculation loop out-of-service for an extended period of time. GE analyses have demonstrated that transient events during single loop operation (SLO) are bounded by those at rated conditions. ANF analyses have confirmed the GE conclusions. Since the ANF fuel was designed to be compatible with the previous co-resident GE fuel in thermal-hydraulic, nuclear and mechanical design performance, and since the ANF methodology has given results which are consistent with those of GE for normal two-loop operation, the staff concludes that the GE analyses for SLO are also applicable to SLO with fuel and analyses provided by ANF.

For SLO, GE found that an increase of 0.01 in the MCPR safety limit was needed to account for increased flow measurement uncertainties and increased traveling incore probe (TIP) uncertainties associated with single pump operation. ANF has also evaluated these effects and found that the 0.01 increase in the allowed safety limit MCPR is applicable to ANF fuel during SLO. Therefore, the staff concludes that increasing the safety limit MCPR by 0.01 for SLO with ANF fuel during Cycle 11 is acceptable.

ANF has also performed LOCA analyses for SLO conditions, as discussed in Section 2.5, to determine an appropriate SLO MAPLHGR multiplier for ANF 8x8 and 9x9 fuels.

2.8 Technical Specification Changes

To support D3C11 operation with a mixed core of ANF 8x8 and ANF 9x9 fuel consistent with the safety analyses, the following Technical Specification changes have been requested:

(1) Specification 3.5.K: A new Section for transient LHGR limits is added. This provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events

beginning at any power and terminating at 120% of rated thermal power and is, therefore, acceptable.

(2) Specification 1.1A, 3.5.L.3, 3.6.H.3.f.iv, 3.6.H.3.f.v: The MCPR LCO adder during SLO is changed to 0.01 from 0.03. This results in an increase in the MCPR safety limit for SLO of 0.01 (relative to two loop operation). As discussed in Section 2.7, the increase of safety limit MCPR by 0.01 for SLO is to account for the increased uncertainties in the total core flow and TIP reading and is acceptable.

(3) Specification 3.5.L: The MCPR LCO is changed to 1.39 for both 8x8 and 9x9 fuel. This has been shown to bound the limiting transients and accidents in Cycle 11 and is, therefore, acceptable.

(4) Specification 3.5.L.1 and Figure 3.5-2: The Figure (Sheets 1, 2, and 3) is revised to incorporate changes in reduced flow MCPR values. This has been evaluated in Section 2.5 and found to be acceptable.

(5) Specification 1.0: The definitions of the Fraction of Limiting Power Density (FLPD) and the Maximum Fraction of Limiting Power Density (MFLPD) are deleted and replaced by the definitions of the Steady State Linear Heat Generation Rate (SLHGR), the Fuel Design Limiting Ratio for Exxon Fuel (FDLRX), the Transient Linear Heat Generation Rate (TLHGR) and the Fuel Design Limiting Ratio for Centerline Melt (FDLRC). These changes are administrative in nature and delete information no longer applicable or provide clarification to current specifications.

(6) Specification 2.1.A.1, 2.1.B, 3.1.A.2, 4.1.A.2.a, and Table 3.2.3: References to MFLPD, MFLPD/FRP, and FRP/MFLPD are changed to the indicated FDLRC or 1/FDLRC. These are also administrative changes and are acceptable.

(7) Specification 3.5.I: Figure 3.5-1 (Sheets 3, 4, and 5) are deleted. These administrative changes are acceptable.

(8) Specification 3.5.J, 4.5.J: The Section 3.5.J title is changed to "LOCAL STEADY STATE LHGR," references to FDLRX are added, GE LHGR design value of 13.4 KW/ft is deleted from Figure 3.5-1A, and "STEADY STATE" is added to title. These administrative changes are acceptable.

(9) Specification 3.K.5: A new Section on local transient LHGR and FDLRC is added as well as Figure 3.5-1B showing the Transient LHGR Limit curve. These administrative changes are acceptable.

(10) Specification 3.5.I, 3.6.H.3.f.vi: The SLO MAPLHGR multiplier is changed to 0.91 from 0.70. This has been justified by the results of the LOCA analyses for SLO discussed in Section 2.5 and is acceptable.

(11) Specification 3.5.L: MCPR Penalty based on scram time performance is deleted. This is acceptable since the MCPR LCO of 1.39 conservatively bounds the delta-CPR results of the plant transient analyses for Cycle 11.

(12) Specification 3.5.D, 4.5.D, 3.5.I, 3.6.H.3.f.vi: Wording is changed to allow extended operation with one RVOOS and limited (7 days) operation with two RVOOS provided HPCI is operable and MAPLHGR adjustment factors are applied. Operation is allowed with one RVOOS provided appropriate MAPLHGR reductions discussed in Section 2.5 are implemented. Analyses have shown that allowing two RVOOS (and assuming the HPCI is inoperable) may cause the 2200° F PCT limit to be exceeded. To allow a longer repair time, HPCI operability must be credited. Since HPCI must be tested upon finding two RVs inoperable, this change allows the same 7 day period recently approved for Quad Cities Unit 1 Cycle 10, provided HPCI is shown to be operable.

(13) Throughout the Technical Specifications and Bases, references to Exxon Nuclear Company (ENC) have been changed to Advanced Nuclear Fuels Corporation (ANF). In addition, various sections have been revised to

reflect the appropriate ANF methodologies and to delete GE methods and references where appropriate. These are acceptable administrative changes.

2.9 License Change

The following license restriction has been supported for Cycle 11 operation:
"Section 3.E Restriction

Operation in the coastdown mode is permitted to 40% power."

This restriction drops the portion of 3.E regarding off-normal feedwater heating which required a determination if the MCPR Operating Limit and calculated peak pressure for the worst case abnormal operating transient remain bounding. This has been evaluated in Section 2.5, Transient and Accident Analysis, of this Safety Evaluation and found to be acceptable. Thus modifying Section 3.E of the license to delete the requirement to prepare a safety evaluation for coastdown operation with off-normal feedwater temperature is acceptable.

3.0 SUMMARY

Based on the review of the fuel, nuclear, and thermal-hydraulic design as well as the transient and accident analysis presented by the licensee, the staff concludes that the proposed reload of Dresden 3 Cycle 11 and associated Technical Specification changes are acceptable.

4.0 REFERENCES

1. Letter from J. A. Silady (CECo) to T. E. Murley (NRC), Dresden Nuclear Power Station Unit 3 Proposed License Amendment and Analysis for Cycle 11 Reload, March 9, 1988.

2. ANF-87-097, "Dresden Unit 3 Cycle 11 Reload Analysis," September 1987.

3. ANF-87-096, "Dresden Unit 3 Cycle 11 Plant Transient Analysis," September 1987.

4. XN-NF-84-49, "Analysis of Dresden Units 2 and 3 Operation With One Relief Valve Out-of-Service," September 1984.

5. ANF-87-111, "LOCA-ECCS Analysis for Dresden Units During Single Loop Operation with ANF Fuel," September 1987.

6. XN-NF-85-067(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1986.

7. ANF-87-110(P), "Dresden Unit 3 Cycle 11 Nuclear Design Report," July 1987.

8. Letter from G. N. Ward (ANF) to G. C. Lainas (NRC), "Additional Information on Rod Bowing," GNW:021:87, March 11, 1987.

9. XN-NF-82-06(P), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Supplement 1, Extended Burnup Qualification of ENC 9x9 BWR Fuel," January 1987.

10. Letter from A. Thadani (NRC) to D. A. Adkisson (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-82-06(P), Supplement 1, Revision 2," May 3, 1988.

11. Letter from H. Bernard (NRC) to G. F. Owsley (ENC), "Acceptance for Referencing of Topical Report XN-NF-512, Revision 1," July 22, 1982.

12. Letter from C. O. Thomas (NRC) to J. C. Chandler (ENC), "Acceptance for Referencing of Licensing Topical Report XN-NF-734, Confirmation of the XN-3 Critical Power Correlation for 9x9 Fuel Assemblies," February 1, 1985.

13. XN-NF-524(P)(A), Revision 1, "Exxon Nuclear Critical Power Methodology for BWRs," November 1983.

14. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," September 1985.

15. XN-NF-80-19(P)(A), Volume 1 (and Supplements 1 & 2), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," March 15, 1983.

16. TR-VR-85-225-A, "Topical Report of ASEA-ATOM BWR Control Blades for US BWRs," October 1985.

17. XN-NF-79-71(P), Revision 2 (and Supplements), "ENC Plant Transient Methodology for Boiling Water Reactors," November 1981.

18. XN-NF-85-62, "Dresden Unit 3 Cycle 10 Plant Transient Analysis," September 1985.

19. Letter from G. C. Lainas (NRC) to G. N. Ward (ENC), "Acceptance for Referencing of Licensing Topical Report XN-NF-84-105(P), XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," October 27, 1986.

20. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors - THERMEX: Thermal Limit Methodology Summary Description," January 1987.

21. XN-NF-79-71(P), Revision 2, Supplement 3, "Revised Methodology for Including Uncertainties in Determining Operating Limits for Rapid Pressurization Transients in BWRs," March 1985.

22. CN-NF-81-75(P), "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model," November 1981.

23. XN-NF-85-63, "Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR Results for 9X9 Fuel," September 1985.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32 the Commission has determined that granting this amendment will have no significant impact on the environment (53 FR 18361).

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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