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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

I. OVERVIEW

By letter dated September 26, 1983 as supplemented by letters dated January 31 and February 8, 1984, the Omaha Public Power District submitted three reload core analysis methodology reports for staff review. The first report (OPPD-NA-8301-P) was entitled "Reload Core Analysis Methodology Overview." The second report (OPPD-NA-8302-P) was entitled "Neutronics Design Methods and Verification." The third report (OPPD-NA-8303-P) was entitled "Transient and Accident Methods and Verification." Our evaluation of the above specified reports follows.

II. EVALUATION OF OMAHA PUBLIC POWER DISTRICT NUCLEAR ANALYSIS RELOAD CORE ANALYSIS METHODOLOGY OVERVIEW (OPPD-NA-8301-P)

1.0 Summary of Report

This report provides an overview of the Omaha Public Power District (OPPD) reload core analysis methodology for application to the Fort

Calhoun Station Unit 1. Analyses performed either by OPPD or by one of its nuclear fuel vendors are briefly described. The areas of analysis are fuel system design, nuclear design, thermal hydraulic design, postulated accidents and transients, and setpoint generation.

2.0 Summary of Review

We have reviewed the information presented with regard to the calculational methods and assumptions used in reload core analyses for the Fort Calhoun Station Unit 1. In particular, we have reviewed only those portions of the analyses which are performed by OPPD. Analysis performed by one of the Fort Calhoun fuel vendors was merely reviewed to assure that previously approved methods were used and that the vendor has been previously qualified to perform the analysis.

2.1 Fuel System Design

The licensee has stated that the fuel system mechanical design and analyses are performed by the nuclear fuel vendor. These analyses are used to show that the fuel will meet all mechanical limits (e.g., stress, strain, fatigue, seismic and LOCA loads) previously identified in the FSAR and vendor's fuel design report. The fuel design report currently utilized for Fort Calhoun is "Generic Mechanical Design Report for Exxon Nuclear Fort Calhoun 14X14 Reload Fuel Assembly" (XN-NF-79-70), which has been reviewed and approved by the NRC.

2.2 Nuclear Design

The nuclear design methodology used by OPPD is presented in a companion topical report, OPPD-NA-8302-P, "Neutronics Design Methods and Verification." Our evaluation of this report is contained in section III.

The reload core fuel management is performed by OPPD and incorporates schemes to reduce the flux to the reactor pressure vessel welds. The approved CE method to measure power distributions is used by OPPD as described in CENPD-145-P, "INCA, Method of Analyzing Incore Detector Data in Pressurized Water Reactors, " dated April 1, 1975. Approved power distribution uncertainties derived in CENPD-153-P, Revision 1-A, "INCA/CECOR Power Peak Uncertainty," dated May 1980, are used by OPPD and applied to the total peaking factor and the integrated and planar radial peaking factors. The good agreement between OPPD calculated radial and axial power distributions using ROCS and measured distributions for Fort Calhoun justifies the continued use of these uncertainties by OPPD. In addition, the data for Fort Calhoun which was used by CE to derive these power distribution uncertainties was supplied by OPPD. The physics safety related data are produced using the methodology discussed in OPPD-NA-8302-P.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic design metholology used by OPPD for reload analysis was previously submitted to the NRC in the Cycle 8 reload application for Fort Calhoun and was approved for OPPD use.

This includes the steady state DNBR analysis using the TORC/CETOP/CE-1 methodology. In addition, the statistical combination of uncertainties associated with the thermal-hydraulic analysis is reviewed in a separate safety evaluation. In this method, the impact of component uncertainties on DNBR is assessed and the minimum DNBR is increased to include the effects of the uncertainties.

2.4 Postulated Accidents and Transients

The ability of OPPD to analyze postulated transients and accidents in Fort Calhoun reload cores as well as the methodology used is presented in OPPD-NA-8303, "Transient and Accident Analysis Methods and Verification", dated September 1983. This report is evaluated in section IV.

2.5 Setpoint Generation

Omaha Public Power District uses the methodology discussed in CENPD-199-P, Revision 1, "CE Setpoint Methodology," dated April 1982, to generate setpoints for Fort Calhoun. Although this report is still under NRC review, the described methodology has been used by CE to calculate limiting safety system settings (LSSS) for the local power density and thermal margin trip systems and limiting conditions of operation (LCO) for reactors such as Fort Calhoun which incorporate the CE analog reactor protection system.

The setpoint generation for the Cycle 8 reload of Fort Calhoun was performed by OPPD and accepted by the NRC. In addition, OPPD has demonstrated their competency by performing the setpoint generation calculations as part of their training under instruction by CE.

3.0 Evaluation Procedure

We have reviewed the report within the guidelines provided by Sections 4.2, 4.3, and 4.4 of the Standard Review Plan (NUREG-75-087). This report presents merely an overview of the methodology used by OPPD for reload analysis of Fort Calhoun. The actual methodology and qualification of its use by OPPD is presented in various other reports.

4.0 Regulatory Position

We have reviewed the OPPD reload core methodology overview for Fort Calhoun reload cores and we conclude that it presents an acceptable overview of the analysis which will be performed by either OPPD or by its fuel vendors CE or ENC to license reload cores for the Fort Calhoun Station. The fuel system mechanical design and analyses are performed by the nuclear fuel vendor as described in XN-NF-79-70 which has been reviewed and approved by the NRC.

The nuclear design methodology was presented in OPPD-NA-8302-P. The thermal-hydraulic methodology was approved by the NRC for use by OPPD in the review of the Fort Calhoun Cycle 8 reload application. The ability of OPPD to analyze postulated accidents and transients as well as the methodology used was presented in OPPD-NA-8303-P. The ability of OPPD to perform the required setpoint generation calculations for Fort Calhoun reload cores has also been demonstrated and approved.

III. EVALUATION OF OMAHA PUBLIC POWER DISTRICT RELOAD METHODOLOGY REPORT ON NEUTRONICS DESIGN METHODS AND VERIFICATION (OPPD-NA-8302-P)

1.0 Summary of Report

This report summarizes the steady-state nuclear analysis methods used by Omaha Public Power District (OPPD) in support of reload analysis for its Fort Calhoun Station Unit 1. Section 2.0 of the report describes the basic physics models which were supplied by Combustion Engineering (CE). Section 3.0 describes OPPD's application of models to the Fort Calhoun reactor. Section 4.0 presents the application of these physics models to the reload core analysis. Comparisons of OPPD calculated data to measured operating data from Fort Calhoun and data from independent calculations are discussed in Section 5.0.

The computer models used to obtain few group neutron cross sections and to perform few group diffusion theory calculations are the CE models. CEPAK and/or DIT are used for cross section generation and QUIX, PDQ-X, and ROCS for one, two or three dimensional static diffusion theory calculations. DIT-generated cross sections are presently generated for OPPD by CE. These codes are maintained on the CE computer system at Windsor, Connecticut and OPPD accesses them through a time sharing system.

These physics models are used by OPPD for the calculation of the various core physics parameters used in reload safety analyses. The main parameters considered are the radial peaking factors, the moderator temperature coefficient (MTC), the Doppler coefficient, the neutron kinetics parameters, control element assembly (CEA) drop data, CEA ejection data, CEA scram reactivity, and axial power distributions.

2.0 Summary of Review

We have reviewed the information presented with regard to calculational methods and comparisons of calculations and experiment. The computer programs used are previously accepted CE programs and, therefore, require no additional review. However, a licensee planning to perform his own safety analyses must demonstrate his ability to use a computer program by performing his own program verification. Therefore, during the staff's review of the Cycle 8 reload of the Fort Calhoun reactor, OPPD was asked to present verification of its reload technology. This topical report as well as the OPPD responses to NRC questions on Fort Calhoun Cycle 8 (CEN-242-(O)-P) dated February 18, 1983, comprise the OPPD response to the NRC request. In addition, a meeting was held with OPPD and CE on the use and verification performed by OPPD and an independent audit was performed by CE confirming that conservative results were obtained by OPPD for the Cycle 8 reload.

The DIT-generated neutron cross sections used in the verification of the neutronics models have been generated for OPPD by CE. New sets of cross sections may be required in future cycles and, if so, will be generated for OPPD by CE using the DIT code.

The verification of OPPD's ability to utilize the CE neutronics models for reload physics calculations is based on experimental data from various operating cycles of the Fort Calhoun reactor as well as on independent calculations performed by CE and Exxon Nuclear Company (ENC).

Core reactivity is benchmarked by comparing measured and calculated critical boron concentrations for the unrodded hot zero power (HZP) and full power Fort Calhoun core over the last several cycles of operation. The results, using ROCS with the DIT cross sections supplied by CE, are consistent with currently approved design methods as well as with previously submitted results using cross sections generated by OPPD with the CEPAK code and are within currently accepted calculational accuracies.

Comparisons between axially integrated assembly power as calculated by ROCS and that measured for instrumented assemblies show agreement which is consistent with that obtained from currently approved design methods. Comparisons of core average and assembly axial power distributions calculated by ROCS and measured in Cycles 5 through 8 of Fort Calhoun also show acceptable agreement as do comparisons of measured and QUIX code calculated axial shape indices (ASI) during an axial oscillation test.

The isothermal temperature coefficient (ITC) and the power coefficient (PC) have been benchmarked against physics tests from various Fort Calhoun operating cycles. The comparison of ITC's for zero power and near full power conditions are within the generally accepted calculational criterion. The comparison of calculated and measured power coefficients at various power levels is also consistent with that obtained from currently approved design methods.

The analytical methods described by the licensee require, as input, a number of fuel thermal design parameters. These include fuel centerline temperature (for fuel thermal limits), fuel volume-averaged temperature (for Doppler and stored energy), and fuel-to-cladding gap conductance (for cladding thermal limits and moderator feedback effects). The licensee has stated that these parameters are supplied by the fuel vendor on the basis of (previous and proposed) operational data from OPPD. In turn, the operating assumptions made in the vendor analysis are incorporated into the operating limits for the proposed core cycle. Because these calculations rely on reviewed and approved analytical methods, we find the Fort Calhoun thermal design procedure acceptable.

Comparisons of individual group CEA worths calculated by OPPD with both measured values and values calculated by CE and ENC are well within currently used acceptance criteria for both individual group and sum of individual group CEA worths. Both two-dimensional and three-dimensional ROCS calculations were performed by OPPD for these comparisons.

3.0 Evaluation Procedure

We have reviewed the report within the guidelines provided by Sections 4.2 and 4.3 of the Standard Review Plan (NUREG-75/087). Included in our review was the description of the experimental data base, the calculations performed, and the comparisons made between OPPD calculations and experiment and/or other vendor calculations. Information supplied by OPPD and CE during our previous review of the Fort Calhoun Cycle 8 reload was also utilized in this review. This information was reviewed in order to demonstrate the ability of OPPD to utilize the neutronics methods described to calculate the physics parameters used in reload safety analyses.

4.0 Regulatory Position

We have reviewed the CE PWR neutronic methods used by OPPD and verified by benchmarking against Fort Calhoun measurements over several cycles of operation. We find that OPPD has adequately demonstrated their ability to calculate the following parameters for PWR reload cores: (a) core reactivity, (b) power distribution, (c) control rod worth, (d) moderator temperature coefficient, (e) Doppler coefficient, and (f) neutron kinetics parameters.

Although the benchmarking has shown acceptable results, the data base in some instances is somewhat limited. Therefore, we strongly recommend continuation of the OPPD ongoing benchmarking program including startup physics testing predictions, reactor testing analysis and the core follow effort in order to provide continuing assurance of the model applicability and the calculational accuracy.

The DIT-generated neutron cross sections used in the verification of the neutronics models have been generated for OPPD by CE. Therefore, the ability of OPPD to generate few group neutron cross sections using DIT has not, as yet, been demonstrated. These will continue to be supplied to OPPD by CE.

IV. EVALUATION OF OMAHA PUBLIC POWER DISTRICT RELOAD ANALYSIS METHODOLOGY ON TRANSIENT AND ACCIDENT METHODS AND VERIFICATION (OPPD-NA-8303-P)

1.0 Introduction

Ft. Calhoun Cycle 8 reload analysis was performed by Omaha Public Power District (OPPD) using Combustion Engineering (CE) computer codes, methodology, and procedures except for the large break LOCA which was performed by the fuel vendor. A question was raised by the staff during its review of the Ft. Calhoun Cycle 8 reload with regard to the capabilities of the licensee to perform the reload safety analysis using CE computer codes. Consequently, an independent audit was performed by CE which confirmed that the results obtained by the licensee for the Cycle 8 reload were conservative. In response to the staff request, the licensee submitted its reload analysis methodology report by a letter dated September 26, 1983.

The purpose of this report was to document the methodology to be used by OPPD in future reload analyses and to demonstrate OPPD's capability to perform analyses using CE computer codes. The staff evaluation is addressed below.

2.0 Transient and Accident Models and Code Verification

The Omaha Public Power District (OPPD) utilized the CESEC III code to simulate the Ft. Calhoun plant response to non-LOCA initiating events. The CESEC III code has not been generically approved by the staff for all CE designed plants. The licensee should submit the topical report of the CESEC III code on the Ft. Calhoun docket for staff approval. This submittal should demonstrate the applicability of CESEC III to Ft. Calhoun. The OPPD used the CETOP and TORC computer codes for calculating the DNBR during transients and accidents. The CETOP code has been approved by the staff for Ft. Calhoun plant. The TORC code has been generically approved by the staff for all C-E designed plants.

In order to demonstrate OPPD's ability to correctly use the CESEC-III computer code, verification work has been performed by benchmarking both actual plant transient data and independent safety analyses previously performed by fuel vendors and accepted by the NRC.

For plant transient benchmarking, the CESEC-III code was set up to model Ft. Calhoun Cycle 1 in a best estimate mode to permit accurate comparisons to the actual measured plant responses for the turbine-reactor trip and total loss of RCS flow transients. The results show that the CESEC-III predicted parameters for the turbine-reactor trip and the 35% power total loss of coolant flow show very good agreement with those measured during Cycle startup testing.

For the transients and accidents analyzed by OPPD (using CE methodology) for which no plant data existed, OPPD compared its analysis for the Cycle 8 reload with the analyses for the Cycle 6 reload performed by the fuel vendors (CE and Exxon). The transients and accidents compared included Dropped CEA, MSLB at full power and zero power, and RCS depressurization. Although the comparison was performed for two different cycles, the important core physics parameters are essentially close enough so as to enable a direct comparison to be made between the two calculations. The primary system responses between the OPPD and CE/ENC analyses show good agreement with each other.

3.0 Transients and Accidents Considered in the Reload Core Analyses

The OPPD criteria for determining the events needing reanalysis are those adversely affected by the changes in various neutronic parameters with the reload core. If these parameters change such that the previously reported results for an event in the referenced safety analyses are no longer conservative, then this event must be reanalyzed. If these parameters are conservative with respect to the values assumed in the referenced safety analyses, this event is not reanalyzed.* Reanalyses will be performed if a change in the

*As with all plant changes, tests, or experiments, the criteria of 10 CFR 50.59 must be met.

Technical Specifications is necessary, and the reanalyses will be reported as part of the supporting documentation for a Facility License Change.

The methods of analyses for the transients and accidents considered in a reload core analysis are discussed below.

For an anticipated operational occurrence (AOO), the acceptance criteria are that the transient minimum DNBR must be greater than the 95/95 confidence interval limit for the CE-1 correlation, the peak linear heat generation rate (PLHGR) must not exceed 21 kt/ft, and the peak system pressure should not exceed 110% of the design pressure.

For a non-LOCA initiating accident, the acceptance criteria are that the site boundary doses must be within 10 CFR 100 guidelines and the reactor coolant pressure boundary must be maintained intact.

The licensee is not required to submit the results of an analysis for staff approval if the MDNBR is found to be greater and the PLHGR is found to be smaller than that reported in the latest reference analysis or if the required overpower margin calculated for the event is less than that being maintained by the current Technical Specifications.

3.1 CEA Withdrawal Event

The methodology used for the analysis of the CEA withdrawal event is the same as that described by CE in CEN-121(B)-P, "CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected Systems," dated November 1979, and approved by the NRC. This event is classified as an Anticipated Operational Occurrence (AOO).

CEA withdrawal occurring from both full power and hot zero power conditions will be analyzed in future cycles if necessary. The full power event will be used to determine the required overpower margin (ROPM) which must be factored into the setpoint analysis. If subsequent reload analyses for the full power or intermediate power level CEA withdrawal event show that the ROPM is less than the available overpower margin required by the Technical Specifications, the event is acceptable and the licensee is not required to submit the results of reanalysis for the staff approval. The hot zero power event will be used to demonstrate that the variable high power trip is initiated in time to assure that the above-mentioned AOO acceptance criteria are met.

The staff concludes that the described analytical methods contain sufficient conservatism, with respect to both assumptions and models, to assure that fuel damage will not result from such CEA withdrawal events.

3.2 Boron Dilution Event

An inadvertent boron dilution could occur as a result of a combination of operator error and a CVCS malfunction. No reactor protection system trips are assumed to terminate this event. This event is classified as an AOO.

The DNBR and PLHGR criteria are met by showing that sufficient time is available for the operator to take corrective action to terminate the event prior to exceeding the specified acceptable fuel design limits (SAFDLs). The acceptable time interval for the operator to take corrective actions before the shutdown margin is lost are 15 minutes for hot standby, hot shutdown and cold shutdown, and 30 minutes for refueling mode. The times calculated in OPPD analyses method were times from start of dilution to the time of loss of shutdown margin. However, the SRP 15.4.6 requires that the time available to the operator be measured from time of an audible alarm to the time of loss of shutdown margin. Also, the SRP requires positive redundant audible alarms. Ft. Calhoun has several passive alarms and indications but none of them could be considered as positive alarms for an inadvertent boron dilution event. In the staff's SER for Cycle 8 operation, the reliance on these alarms was found acceptable. Since that time generic analyses of unmitigated boron dilution events have been conducted. The preliminary results of these analyses indicate that the criteria for anticipated occurrences are not violated. We believe this is the case for Ft. Calhoun. However, in future reloads, the licensee should ensure that the criteria used and found acceptable for Cycle 8, as specified above, are met. Otherwise reanalysis and submittal to the staff for formal review are required.

Although the boron dilution event alarms reviewed in Cycle 8 were found acceptable, the staff believes the licensee should consider upgrading these alarms to provide a positive indication of a boron dilution event in progress.

3.3 CEA Drop Incident

The methodology used for the analysis of the CEA drop incident is the same as that described by CE in CENPD-199-P, Revision 1, "CE Setpoint Methodology," dated April 1982. Although this topical report is still under NRC review, the CE methods described therein have been accepted by the staff in previous CE PWR cores. The CEA drop incident is classified as an AOO, Sufficient initial steady state margin is built into the Technical Specification Limiting Conditions for Operation (LCO) to enable the core to withstand the event without requiring a reactor protection system trip. The analysis is used to determine the ROPM which must be built into the LCOs to assure that the DNBR and LHR acceptance criteria are met.

In order to demonstrate OPPD's ability to correctly use the CE methodology for calculating this transient, the Ft. Calhoun Cycle 8 dropped CEA analysis performed by OPPD was compared to the previous analysis performed by ENC and contained in the Updated Safety Analysis Report. Although the comparison was performed for two different cycles, the important core physics parameters are close enought to enable a direct comparison between the two calculations. The primary system responses between the ENC and OPPD analyses including core power, core heat flux, coolant temperature, and pressurizer pressure show good agreement.

The staff concludes that the analytical methods contain sufficient conservatism, in both input assumptions and models, to assure that fuel damage will not result from CEA drop incidents. In addition, good agreement was obtained between OPPD and ENC analysis which demonstrated OPPD's ability to correctly calculate this transient.

3.4 Four-Pump Loss of Flow Event

The four-pump loss of coolant flow event is initiated by the simultaneous loss of electrical power to all four reactor coolant pumps. A reactor trip would be initiated when the flow rate drops to 93% of full flow. The four-pump loss of flow event is classified as an AOO. The analysis determines the required overpower margin that must be build into the DNB LCOs such that in conjunction with the low flow trip the SAFDL is not exceeded.

The methodology used for the analysis of the four-pump loss of coolant flow event is the same as that described by CE in CENPD-199-P, Revision 1, "CE Setpoint Methodology," dated April 1982. Also CESEC III is used for transient analysis.

The staff finds that the OPPD's analysis method of the four pump loss of flow event is essentially the same as the methodology used by CE in the original FSAR and therefore it is acceptable.

3.5 Asymmetric Steam Generator Event

The asymmetric transients initiated by a secondary system malfunction in one steam generator result in changes in core power distribution. This event is analyzed to determine the initial steady state thermal margin which is built into and maintained by the Technical Specification LCO in term of overpower trip setpoint such that SAFDLs are not exceeded for these transients. The asymmetric steam generator transient protection trip function (ASGTPTF) has been reviewed and approved by the staff for Calvert Cliffs 1 and Ft. Calhoun Unit (See staff SER supporting Amendment No. 48 to Facility operating license No. DRP-53 for Calvert Cliffs Unit 1 and Staff SER on Ft. Calhoun Cycle 9 reload) and will be installed in the Ft. Calhoun plant RPS prior to operation of Cycle 9 to reduce the margin requirements associated with these asymmetric events. This event is classified as AOO. The analysis determines the required overpower margin that must be built into the LCO's such that in conjunction with the ASGTPTF the SADFL's is not exceeded.

The methodology used for the analysis of this event is the same as that described by CE in CENPD-199-P, Revision 1, "CE Setpoint Methodology," dated April 1982.

The staff concludes that the OPPD's analysis method contain sufficient conservatism, with respect to both assumptions and models, to assure that the SAFDL will not be exceeded from such event and therefore, it is acceptable.

3.6 <u>Excess Load Event</u>

A rapid increase in steam flow results in a power mismatch between the reactor core and the steam generator load demand. In excess load transients, there is a decrease in the reactor coolant temperature and pressure and the negative moderator temperature coefficient reactivity causes an increase in core power.

The OPPD will evaluate the following load increase events in the future reload analysis, (1) Rapid opening of the turbine control valves at power, (2) Opening of all dump and bypass valves at power due to steam dump control interlock failure, (3) Opening of the dump and bypass valves at hot standby conditions due to low reference temperature setting in the steam dump controller, and (4) Opening the dump and bypass valves at hot standby due to steam dump controller malfunction. Those events are classified as AOOs.

The methodology used for the analysis of this event is the same as that described by CE in CENPD-199-P, Revision 1, "CE Setpoint Methodology," dated April 1982. The licensee is not required to submit the results of reanalysis for the staff approval if the calculated pressure bias term is less than or equal to the value used in the current Thermal Margin/Low Pressure (TM/LP) trip equation. This indicates the current TM/LP setpoints would prevent DNB with larger thermal margin.

The staff finds that the OPPD's analysis method of the excess load event is essentially the same as the methodology used by CE in the original FSAR and therefore it is acceptable.

3.7 RCS Depressurization Event

A rapid decrease of the primary system pressure could be initiated by either the inadvertent opening of both power operated relief valves (PORVs) or the inadvertent opening of a single primary safety valve. The TM/LP trip is used to prevent exceeding the SAFDL from this transient. The RCS depressurization event is classified as an AOO.

The methodology used for the analysis of this event is the same as that described by CE in CENPD-199-P, Revision 1, "CE Setpoint Methodology," dated April 1982. The licensee is not required to submit the results of an analysis for the staff review if the calculated pressure bias term is less than or equal to the value used in the current Thermal Margin/Low Pressure (TM/LP) trip equation. Because this indicates the current TM/LP setpoints could prevent DNB with larger thermal margin.

The staff finds that the OPPD's analysis method of the RCS depressurization event is essentially the same as the methodology used by CE in the original FSAR and therefore it is acceptable.

3.8 Main Steam Line Break Accident

A large main steam line break (MSLB) causes a rapid depletion of steam generator inventory and an increased rate of heat removal from the reactor coolant. This accident will cause an increase in nuclear power and trip the reactor. Both full power and no-load (hot Standby) initial condition cases were considered in the previous reload analysis and will also be considered in the future reload analysis for two loop operation (Four RCPs). A MSLB during a single loop operation is not analyzed since a single loop operation is not permitted by the Ft. Calhoun Technical Specifications. The most probable trip signals resulting from an MSLB include low steam generator pressure, high power, low steam generator water level, TM/LP, and high rate-of-change of power. The MSLB event is classified as a postulated accident for which the site boundary doses must be within the 10 CFR 100 criteria.

The analysis of the MSLB accident will be performed using CESEC III code. The licensee is not required to submit the results of reanalysis for the staff approval if the calculated return-to-power is less than the return-to-power reported for the Cycle 1 analysis, using the current Technical Specification limit on shutdown margin and moderator temperature coefficient.

The staff finds that the OPPD's analysis method of the main steam line break accident is essentially the same as the staff generically approved MSLB methods documented in Appendix H to NUREG-0852 Supplement 2.

3.9 Seized Rotor Event

This event assumes one of the RCP seized instantaneously due to a mechanical failure. The rapid reduction in core flow will initiate a reactor trip on low flow within the first few seconds of the transient. A single RCP shaft seizure is classified as a postulated accident for which the dose rates must be within 10 CFR 100 guidelines. The reload analysis will calculate the amount of fuel pins failure during the event. The licensee is not required to submit the results of reanalysis for the staff approval if the number of pin failures is less than one percent. The OPPD uses TORC code in combination with CE-I correlation to calculate the number of failed fuel pins and utilizes the CESEC code to calculate the transient response for the seized rotor event. The CETOP code is used to determine the time of minimum DNBR.

The analysis of the seized rotor event in the original Ft. Calhoun FSAR does not assume a loss of offsite power following the plant trip and a limiting single active failure following the accident. The acceptance criteria was that the radioactive consequences must be within a small fraction of the 10 CFR 100 guidelines. The OPPD's method and assumptions used for analyzing this event is essentially the same as the ones used in the original analysis. We conclude that this is acceptable for the purpose of reload analyses.

3.10 CEA Ejection Accident

The methodology used for the analysis of the CEA ejection accident is the same as that described by ENC in XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," dated January 19, 1979, and approved by the NRC.

The limiting criteria are those specified in the NRC Regulatory Guide 1.77 "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

The objective of the analysis of a rod ejection from hot full power and hot zero power initial conditions is to demonstrate that the average enthalpy of the hottest fuel pellet is less than the 280 cal/gm NRC acceptance criterion and that the maximum reactor pressure during any portion of the transient is less than the value which will cause stresses to exceed the Service Limit C stress limit as defined in Section III of the ASME Boiler and Pressure Vessel Code. The transient is terminated by either the high power trip or the variable high power trip.

By using the generic ENC methodology, the total deposited energy in the hottest fuel pellet is determined from the cycle-specific physics parameters which include control rod worth, Doppler coefficient, power peaking factors, and delayed neutron fractions. For Cycle 8, OPPD has determined that the resultant enthalpy is less than the value calculated by ENC for Cycle 6, which is the reference cycle for this transient. This ensures that the resultant peak enthalpy is less than the 280 cal/gm criterion and also that the pressure surge will produce stresses less than the Service Limit C limit.

The staff concludes that the described analytical methods contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained in the event of a CEA ejection accident. Since the licensee does not have access to the ENC methodology for analyzing the CEA ejection accident, any required reanalysis will have to be done by ENC.

4.0 Regulatory Position

Based on the analysis methodology and the code verification discussed above, the staff concludes that the OPPD has demonstrated their ability to correctly analyze transients and accidents using CE codes and methods for future reload core analyses. Therefore, it is acceptable that OPPD perform the transient and accident analysis for future Ft. Calhoun reloads using the documented CE computer codes and methodology except the LOCA analysis which will be performed by the fuel vendor.