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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

I. INTRODUCTION

By application dated July 17, 1979, as supplemented October 30 and December 4, 1979, January 30, 1980, February 12 and 25, 1980, and March 12, 1980, Omaha Public Power District (OPPD or the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-40 to allow operation of Ft. Calhoun at 1500 MWt, power operation following Cycle 6 fuel reload and a change in the temperature-pressure requirements for heatup and cooldown. We have not completed our review of the OPPD application to increase Ft. Calhoun power to 1500 MWt and are therefore postponing our action on that request at this time.

II. CYCLE 6 FUEL RELOAD

DISCUSSION & EVALUATION

The Ft. Calhoun core for Cycle 6 operation will contain 40 fresh Exxon Nuclear Company, Inc. (ENC or Exxon) fuel assemblies and 93 burned Combustion Engineering Company (CE) assemblies. ENC performed all the safety analyses for Ft. Calhoun Cycle 6 with two exceptions. OPPD performed the calculations to determine the input to the CECOR program used to calculate the core power distribution and CE performed an evaluation of the validity of the reference CE Small Break

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Loss of Coolant Accident (LOCA) calculation for Ft. Calhoun with Exxon fuel in the core. These are discussed in later sections of the Safety Evaluation.

The licensee proposed to operate Ft. Calhoun at a power level of 1500 MWt (5.6% above the currently licensed power of 1420 MWt). Ft. Calhoun will not be authorized to operate at 1500 MWt at the beginning of Cycle 6, however, because we have not completed our review. We have reviewed the application and find that the licensee has justified operation at 1420 MWt. Ft. Calhoun will operate at 1420 MWt with some TSs which were originally intended for operation at 1500 MWt.

FUEL DESIGN

The Ft. Calhoun reactor consists of 133 fuel assemblies, each having a 14x14 fuel rod array. Each fuel assembly contains 176 fuel rods and five guide tubes. The fuel rods consist of slightly enriched UO_2 pellets inserted into Zircaloy cladding. The control element assembly (CEA) guide tubes and instrumentation tubes are also made of Zircaloy. Each ENC assembly contains nine Zircaloy spacers with Inconel springs; eight of the spacers are located within the active fuel region.

The projected Cycle 6 loading pattern is shown in Figure 3.1 of Reference 19. The initial enrichments of the various fuel batches are listed in Table 3.1 of Reference 19. The beginning of cycle (BOC) 6 exposures along with the batch IDs are shown in Figure 3.2 of Reference 19. The core consists of 40 fresh ENC assemblies at 3.5 w/o U-235 and 93 exposed CE assemblies scatter-loaded throughout the interior of the core. Pertinent fuel assembly parameters for the Cycle 6 fuel are given in Table 3.1 of Reference 19.

Demonstration Assembly

The licensee proposed to retain one CE fuel assembly (originally loaded in Cycle 2) during Cycle 6 to aid in obtaining data on high burnup fuel performance. The licensee stated (Reference 36) that discharge burnup of this assembly could be as high as 45,000 Megawatt days per Metric Ton Uranium (Mwd/MTU).

The licensee stated that the performance evaluation performed by the licensee for the demonstration assembly is based on a Cycle 5 length of 10,500 Mwd/MTU and a projected Cycle 6 length of 10,000 Mwd/MTU. It is applicable to any combination of cycle lengths no greater than a two-cycle length of 20,500 Mwd/MTU.

The licensee stated that this safety analysis is applicable to the operating conditions and TSs of Cycle 5 or the proposed conditions and TSs of Cycle 6 including the increase in power level to 1500 MWt.

Several aspects of fuel behavior become more important to safety at high burnup. Among these are fission gas release which increases the internal pressure of the fuel rod, fuel rod bowing, cladding collapse, corrosion and hydriding.

Increased fission gas release has its most significant effects on cladding collapse and LOCA performance. The licensee has demonstrated, as discussed below, that the response of the fuel rod to both these considerations is acceptable. The hot rod gas pressure was calculated to be below the minimum operating coolant pressure allowed by the TSs.

The licensee performed an analytical prediction of cladding creep-collapse time for the demonstration assembly. Using the computer code CEPAN (Reference 30), the licensee concludes that no creep-collapse will be experienced by this assembly during Cycle 6. Time to cladding creep-collapse for the demonstration assembly is predicted to be greater than 45,000 effective full power hours (EFPH), while the cumulative exposure expected at the end of Cycle 6 is less than 40,000 EFPH.

Fuel rod bowing effects on departure from nucleate boiling (DNB) margin for the high burnup demonstration assembly during Cycle 6 have been evaluated with the guidelines set forth in Reference 31. Since the demonstration assembly reached a burnup of less than 45,000 Mwd/MTU at end of Cycle 5, the fuel rod bowing penalty on DNB prescribed by Reference 31 would be less than 7%. However, the assembly never achieves radial peaks within 30% of the maximum radial peak in the core at any time during Cycle 6. Therefore, no penalty on core DNB margin is required.

The licensee evaluated the performance of this demonstration fuel assembly by performing an analysis according to the Acceptance Criteria for Light-Water-Cooled Reactors as presented in 10 CFR 50.46 (Reference 17).

The calculated peak cladding temperature was 1303°F and the maximum amount of local Zirconium/water reaction was 0.13%. These values are well below the criteria of Reference 17. The analysis was performed for a core power of 1420 MWt. The licensee stated that an increase to 1500 MWt would not significantly affect the results. We concur.

Sufficient data are available to provide reasonable assurance that this fuel assembly will not experience excessive corrosion or hydriding during the planned irradiation period as long as the usual operating coolant chemistry is maintained.

NUCLEAR DESIGN

The nuclear design of the Ft. Calhoun core for Cycle 6 was done with methods used by Exxon in the past (References 5 through 7 and 9). These have been approved by the NRC staff.

The peaking factors Fxy^T and F_R^T remain unchanged. The shutdown margin also remains unchanged as does the power dependent insertion limits.

Fuel rod bowing has an effect on the nuclear design since lateral movement of a fuel rod changes the local moderation. Exxon has submitted a topical report to the NRC staff describing the methods proposed for fuel compatible with CE reactors such as Ft. Calhoun. The conclusion of the report is that, at the maximum expected burnup of an Exxon fuel assembly, no additional multiplier on Fq need be applied to account for fuel rod bowing. While we do not yet agree that this conclusion is valid at the maximum expected burnup, we do agree that it is valid for the expected length of Cycle 6 at Ft. Calhoun. Before Cycle 7 the validity of the Exxon rod bowing model for Ft. Calhoun must be reassessed.

THERMAL HYDRAULIC DESIGN

Cycle 6 will contain both fresh Exxon fuel and CE fuel which has been burned for one or more cycles.

The Exxon fuel is designed to be compatible with the already burned CE fuel assemblies. The licensee has conducted hydraulic compatibility tests which serve two purposes. These tests demonstrate that the flow distribution in either fuel type is not perturbed excessively by the presence of the other. The tests also provide the input data required to perform the safety analyses. These tests were conducted similarly to those described in Reference 29.

The results of these flow tests indicate that the Exxon fuel has a higher flow resistance than the CE fuel. This difference is accounted for in the safety analyses.

Table 1 below gives a brief comparison of the Exxon and CE fuel designs for Ft. Calhoun Cycle 6.

TABLE 1

Design Factor	ENC Fuel	Combustion Engineering
Clad ID (in.)	0.378	0.384
Clad OD (in.)	0.442	0.440
Control rod OD (in.)	1.115	1.115
Bare rod flow area (in^2)	36.25	35.18
Wetted perimeter (in.)	261.9	260.8
Heated perimeter (in.)	244.4	243.3

The fuel temperatures for Cycle 6 were calculated with the GAPEX computer code. For the accident analyses the licensee used a low value of gap conductance (500 BTU/hr ft^2°F). Previous calculations performed by the NRC staff (unpublished) show that a low gap conductance is conservative for the loss of flow transient. For the CEA withdrawal at power, this would be conservative since the gap conductance would actually be increasing with power which would tend to increase the heat flux.

The limiting DNB transient for Ft. Calhoun Cycle 6 is the Rod Drop event. The licensee states that the minimum departure from nucleate boiling ratio (MDNBR) calculated for this transient occurs late in the transient at a steady-state. Hence, the resulting MDNBR, including peaking augmentation, is established via a steady-state calculation which assumes a one-to-one correspondence between the power level and the heat flux at the time of MDNBR and is virtually independent of the value of the gap conductance used in the analysis.

Based on the above considerations, we agree that gap conductance, and its effect on fuel rod transient heat flux, was adequately considered.

ACCIDENT ANALYSIS

All the accident analyses assumed a power level of 1500 MWt. The licensee has applied for authorization to operate Ft. Calhoun during Cycle 6 at this higher power but will begin Cycle 6 at 1420 MWt because we have not completed our review of 1500 MWt operation. The accident analysis is valid for both 1420 MWt and 1500 MWt.

In performing the safety analyses for Cycle 6, the licensee assumed a simultaneous occurrence of the important process variables at their most limiting values as defined by the proposed Cycle 6 TSs. Specifically, the initial reactor power is assumed to be 102% of rated power, system pressure is reduced by 47 psia to a value of 2053 psia, core inlet temperature is increased by 2°F to 547°F, and a minimum anticipated total core flow of 71.7×10^6 lb/hr is used. The limiting fuel rod power was assumed to be the maximum value allowed by the TSs. The simultaneous occurrence of all these parameters at their most limiting values is considered to be unlikely. The analyses show that protection is provided from an anticipated operational occurrence (AOO) or accident event for these unlikely conditions.

The impact upon the initial power distribution during normal operation due to xenon transients is considered to be an axial effect and, as such, is considered in establishing the limiting conditions for operation (LCOs). The axial power profile used in the transient analysis exceeds that allowed by the LCO and hence results in an initial MDNBR lower than actually anticipated. Thus, the initial MDNBR is set at a low value providing additional conservatism in the initial plant conditions to evaluate thermal margins.

Reference 2 reported the results of the postulated Uncontrolled CEA Withdrawal event for Ft. Calhoun Cycle 6. The results of the analysis of this event show that the Thermal Margin/Low Pressure (TM/LP) trip was the first Reactor Protection System response over the whole range of reactivity insertion rates (Figure 3.12 of Reference 2). Normally, protection from an uncontrolled CEA withdrawal is provided by several different trips (e.g., high pressure, high flux and TM/LP) rather than by only one. The licensee explained that this was because the TM/LP equation which was used was generated through the use of a single limiting axial power profile for values of power $>100\%$ which is calculated to be more limiting than would actually be allowed by the Axial Power Distribution (APD) Limiting Safety System Setting (LSSS); hence, a higher than required sensitivity of P_{var} with respect to power results. This increased sensitivity results in a large change in P_{var} with respect to a small change in power to the extent of initiating a reactor trip via TM/LP before the overpower or high pressure trip setpoints are reached.

The maximum reactivity insertion rate assumed by the licensee in the safety analyses report (Reference 2) was 1×10^{-4} $\Delta\rho$/sec. The maximum reactivity insertion rate calculated by the licensee for Cycle 6 is 1.725×10^{-4} $\Delta\rho$/sec. An additional CEA withdrawal analysis was performed with a withdrawal rate of 1.725×10^{-4} $\Delta\rho$/sec. The results are not significantly different from the 1.0×10^{-4} $\Delta\rho$/sec withdrawal rate and still above the safety limit.

In analyzing the Rod Drop event, which is the limiting AOO for Ft. Calhoun Cycle 6, the licensee's calculations show that at 70 seconds the reactor power tends to return to its initial value, the core inlet temperature decreases approximately 7°F and the system pressure decreases about 108 psia. The MDNBR value occurs at this time.

The licensee states (Reference 36) that sensitivity studies performed with XCOBRA-IIIC indicate that a conservative evaluation of the MDNBR for the Rod Drop event can be obtained by assuming initial core conditions plus the peaking augmentation anticipated for this event. In other words, no credit is taken for the decrease in core inlet temperature and no penalty is taken for the pressure decrease. The licensee stated that:

"Since the credit associated with the decrease in inlet temperature exceeds the penalty associated with the pressure decrease, the analysis of the MDNBR for the Rod Drop event is found to be more conservatively calculated using the initial conditions (7°F higher core inlet temperature, and 108 psia higher pressure) than the asymptotic conditions which exist in the transient."

We evaluated this assertion with sensitivity factors based on the W-3 correlation and agree with the licensee that his assumption for this combination of pressure drop and inlet temperature drop is conservative.

The steam line break analysis for Ft. Calhoun was done using several conservative assumptions. In the analysis, the location of the steam line break was assumed to be at the outlet nozzel of the steam dome. The fastest cooldown of the primary system is thus achieved. The discharge coefficient was assumed to be one so that maximum possible discharge rate could be realized. Break flow was calculated each time step based on a choke flow model proportional to the steam generator pressure. This gives the maximum flow rate which in turn gives the maximum cooldown. The steam was assumed saturated; computations during the transient indicated the quality was essentially unity. With the quality equal to unity the steam leaves the steam generator with the highest energy content.

Break flow was computed based on the ideal gas flow model and results in a greater flow than calculated using Moody's results. Thus, the above model is judged to result in a more rapid cooldown and hence an increased likelihood of return to power. These assumptions, although conservative, are consistent with the usual assumptions made in analyzing the steam line break.

We have reviewed the results of the safety analyses of postulated non-LOCA AOOs and accidents for Cycle 6 at Ft. Calhoun as presented in References 2 and 19. In addition, the licensee has provided a computer listing of the input to the PTSPWR computer code used in performing the Uncontrolled CEA Withdrawal event. We have reviewed both the references and the computer listing and conclude that the licensee has adequately calculated the consequences of these events.

The licensee analyzed the Design Basis Large Break LOCA using NRC approved methods. The results are reported in References 3 and 35. The results of the licensee's calculations are summarized in Table 2 of this Safety Evaluation. The most limiting peak cladding temperatures were calculated to occur at the end of life of both the CE and ENC fuel. This is due to the increased fission gas release, combined with the use by the licensee of an NRC model for flow blockage due to swelling and rupture of the fuel rods during a LOCA.

In the past, conditions close to beginning of life have been worse because of fuel densification effects.

The results of the calculations given in Table 2 show that the ENC fuel meets the emergency core cooling system (ECCS) criteria (Reference 17) with the design peaking factor F_Q^T of 2.53. For the CE fuel, the criteria are met with F_Q^T equal to 2.53 up to a peak pellet burnup of 32,600 Mwd/MTU. In order for the CE fuel to meet the ECCS criteria at greater burnups, an F_Q^T reduced by 3% was used. This reduction in F_Q^T is less than would occur at a burnup of 32,600 Mwd/MTU and is therefore conservative.

The Small Break LOCA was not calculated for Cycle 6. The licensee used the Cycle 3 Small Break LOCA analysis as the reference analysis (meaning that the input to the calculation is still valid and the results bound those expected for Cycle 6) and justified in Reference 36 that the analysis was still valid with Exxon fuel in the core. This was done by identifying the difference in fuel

TABLE 2

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Exposure Heatup Analyses Results for ENC and CE Fuel

	ENC FUEL		CE FUEL		
Exposure, PPBU (MWD/MTM	BOL	48,000 (EOL)	BOL	32,600	42,400 (EOL)
Total Peaking, F_Q	2.53	2.53	2.53	2.53	2.46
Peak Clad Temperature (PCT), ^oF	1980	2195	2012	2188	2190
Max. Local Zr/H_2O - Reaction, percent	4.6	9.1	6.2	9.5	10.1
Hot Rod Burst Time, sec	31.6	29.3	28.5	26.6	26.1
Hot Rod Burst Location, ft	7.47	7.47	7.47	7.47	7.47
Time of PCT, sec	208	252	229	235	254
PCT Location, ft	8.22	8.22	8.22	8.22	8.22
Max, Zr/H_2O Reaction Location, ft	8.22	8.22	8.22	8.22	8.22
Linear Heat Generation Rate, kw/ft at BOCREC	0.8218	0.8682	0.8206	0.8596	0.8338
Total H_2 Generation, % of total Zr reacted	<1%	<1%	<1%	<1%	<1%

and system parameters and justifying why these differences would have a negligible effect on the results of the analysis. The parameters considered are listed in Table 3 of this Safety Evaluation. We consider this list to be complete and agree that the reference analysis is valid.

TECHNICAL SPECIFICATIONS

1. Proposed Changes for Cycle 6

As discussed previously, the licensee proposed to operate Cycle 6 at a power of 1500 MWt rather than the currently licensed 1420 MWt. The TS changes to allow 1500 MWt operation were submitted to the NRC staff for review and approval. Since our review for 1500 MWt operation is not yet complete, the licensee is authorized to operate at 1420 MWt maximum. The licensee has chosen to use power distribution limits generated for 1500 MWt although they are more restrictive. These limits are the APD (Figure 1-2 of the TSs), TM/LP Safety Limits (Figure 1-1), TM/LP LSS (Figure 1-3), Allowable Peak Linear Heat Rate vs. Burnup (Figure 2-5), LCO for Excore Monitoring of Linear Heat Rates (Figure 2-6), LCO for DNB Monitoring (Figure 2-7) and Flux Peaking Augmentation Factor (Figure 2-8). The values of F_R^T, F_{xy}^T and axial tilt are unchanged from the previous cycle. We find this acceptable. The reasons are discussed for each item separately.

The APD gives a range of axial shapes which are allowable in the core at a given power level. The power level is given in percent of rated power. Therefore, if the curve was originally intended for operation at 1500 MWt, then 1500 MWt was intended to be 100% rated power. A point on the boundary of this tent is a combination of core power and Axial Shape Index (ASI) which gives a value of centerline temperature equal to the melting point of UO_2. But this is based on 1500 MWt. At 1420 MWt, the point would give a centerline temperature less than the melting point since the core power is 5.6% lower. Therefore, it is acceptable to use this curve for operation at 1420 MWt.

TABLE 3

General System Parameters

Quantity	Cycle 3	Cycle 6	
Reactor power level (102% of Nominal)	1448	1448	MWt
Average linear heat rate (102% of Nominal)	5.85	5.85	kw/ft
Fuel centerline temperature at peak linear heat rate	3945.8	<4120.8 *	°F
Hot rod gas pressure	1346.6	<1471.6 **	psia
Peak linear heat rate	15.5	15.5	kw/ft
Reactor vessel inlet temperature	540	547	°F
Reactor vessel outlet temperature	593	601.4	°F
Active core height	10.7	10.7	ft
Total core pressure drop	7.49	7.51	psi

* EXXON fuel only. The EXXON fuel temperature is conservatively calculated to be no more than 175°F higher than the CE fuel at 15.5 kw/ft.

**EXXON fuel only. The EXXON fuel pin internal pressure is conservatively calculated to have a pin pressure of no more than 125 psi higher than the CE fuel at 15.5 kw/ft.

The APD also gives the range of axial shapes used to calculate the TM/LP safety limit and LSSS discussed below.

The TM/LP Trip Safety Limits is given in Figure 1-1 as curves of coolant inlet temperature vs. core power for various isobars (constant pressure lines). The points along each isobar give values of DNBR equal to the safety limit (1.3 using the W-3 critical heat flux correlation)*. The core power was normalized so that 100% corresponds to 1500 MWt. Using this curve so that 100% corresponds to 1420 MWt means that a point along an isobar would not give a DNBR equal to the safety limit but a higher DNBR. This is conservative and the curve is therefore still acceptable at 1420 MWt.

Since the TM/LP LSSS is based on the TM/LP safety limit curve, it too is conservative with 100% rated power corresponding to 1420 MWt rather than 1500 MWt.

The allowable peak linear heat rate curve gives a value of linear heat rate as a function of burnup which corresponds to the peak linear heat rate assumed for the LOCA. This value is 15.22 Kw/ft for Cycle 6, which was determined from a LOCA analysis done at 1500 MWt. The linear heat rate of 15.22 Kw/ft will be conservative for 1420 MWt since the lower core power yields more favorable system response than at 1500 MWt. In addition, using the same linear heat rate at a lower core power increases the total peaking factor. This would imply that the core would have more localized peaking and therefore the total hydrogen generation might be less than at 1500 MWt.

* Actually a value of 1.35 was used by the licensee for conservatism.

The LCO for Excore Monitoring of the linear heat rate (Figure 2-6) is acceptable when 100% corresponds to 1420 MWt rather than 1500 MWt since it is based on the LOCA peak linear heat rate of 15.22 and therefore the previous argument applies.

The LCO for DNB monitoring is used for protection from those AOOs which do not cause a reactor trip. Even though this curve was also normalized to 100% power equal to 1500 MWt it is conservative for operation at 1420 MWt. This is because at 1500 MWt it is a locus of points from which an AOO which does not cause reactor trip can be initiated without exceeding the safety limit. Therefore, at 1420 MWt, there will be even more margin to the safety limit if the limiting AOO should occur.

For Ft. Calhoun Cycle 6, the limiting AOO is the Rod Drop event.

The flux peaking augmentation factor is a function of fuel bundle geometry and fuel density and is independent of the core power. Therefore, its value does not depend on whether the core operates at 1420 MWt or 1500 MWt.

Summarizing, the curves generated by the licensee for operation at 1500 MWt equal to 100% rated power are conservative when applied to the case of 1420 MWt equal to 100% rated power. Therefore, the licensee may use the same curves in the TSs.

The statement is used in several places in the documents supporting Cycle 6 operation that the safety analysis shows that the TM/LP provides adequate protection; and yet, in Reference 36 which was submitted at a later date after the TM/LP equation had been changed it is stated that TM/LP trip was not calculated to trip the reactor for any of the postulated AOOs (other reactor trips provide the necessary protection instead). Thus, the demonstration of the adequacy of this trip by transient analysis is not given for the final equation proposed by the licensee. However, as discussed later, Battelle Pacific Northwest Laboratories, at the request of the NRC staff, performed a detailed audit of this equation, including independent calculations, to verify that this equation is adequate.

As part of the review of the Ft. Calhoun Cycle 6 APD LSSS, an evaluation of the Exxon methods used to account for effects of rod shadowing in the APD setpoint was made by our consultants at Brookhaven National Laboratory.

The APD trip which restricts core power level in order to avoid fuel centerline melt is a function of core ASI. Since the response of the Excore detectors used to monitor the core ASI is dominated by the peripheral assemblies, a correlation between core and peripheral ASI must be determined. This correlation is generally factored into a shape annealing factor which is independent of rod insertion and a rod shadowing factor which accounts for control rod dependence.

The rod shadowing factor at a given rod insertion is a function of the ASI of a given assembly-i, ASI(i), and a weighting factor W(i) which is the fraction of Excore signal contributed by that assembly-i. The ASI(i) are determined from a 3-dimensional XTG power distribution as a function of rod insertion. Exxon determines W(i) for assemblies on the periphery using a cylindrical 123-group S_8-P_3 XSDRNPM transport model for tracking neutrons along the ray from assembly-i to the Excore detector. For assemblies not on the core periphery the weights are calculated with a 1-D PDQ slab model. Since the weights of these internal assemblies are a factor of 10 less than the peripheral assemblies, the effect of this approximation should be small. Implicit in this approach is the assumption that the weights W(i) are independent of rod insertion. Applying this method to Ft. Calhoun, Exxon finds that the Rod Shadowing Factor is approximately zero at rated power (low rod insertion) and increases to a maximum of approximately 0.02 ASI units at 65% rated power (high rod insertion.)

Exxon incorporates this rod shadowing correction in the APD LSSS barn by conservatively shifting all (power, ASI) data points by .02 ASI towards the center of the barn. In order to account for the uncertainties in the Exxon calculation of Rod Shadowing Factor, a more appropriate prescription is to (1) adjust each data point by the ASI rod shadowing correction calculated for that point and (2) as an uncertainty allowance, displace all points by an additional .02 ASI in the conservative direction. This procedure has been discussed in detail with Exxon and is being adopted as a revision to the Exxon setpoint methodology for CE reactors (Reference 37). After applying this new procedure to Ft. Calhoun (Cycle 6), the licensee stated that all data points remain outside the APD barn and the proposed APD LSSS remains conservative.

We consider that operation of Ft. Calhoun with the Rod Shadowing Factor calculated by the revised method as described above is acceptable.

Reference 19 discusses an ECCS axial profile analysis in which the licensee determined the LOCA limits for reactor operation by performing LOCA calculations with the peak power assumed to be located at various axial positions. A curve was generated (Fig. A.2 of Reference 19) which shows that it is necessary to limit the local power above 70% core height. However, the licensee states (Reference 27, page B-3) that additional studies verify that as long as the ASI is maintained within the bounds of Figures 2-6 and 2-7 of the TSs, the use of a single value of linear heat rate (as given in Figure 2-5 of the TSs) still provides assurance that the LOCA criteria (Reference 17) will be satisfied.

The licensee also proposed a change to Section 2.10.3 "In-Core Instrumentation" to use language closer to that of the CE Standard TSs. The in-core detector system will be considered operable if it consists of (a) at least 75% of all in-core detector strings, and (b) a minimum of two in-core detector strings per quadrant.

The licensee omitted the Standard TS requirement from (b) that the minimum of two detector strings be quadrant symmetric.
We agree that this is acceptable since Ft. Calhoun uses the CECOR monitoring system which is a full core system and does not assume symmetry as does the older version, INCA.

The Standard TSs require that if the in-core detector system is not operable (does not meet conditions (a) and (b) above) then since the in-core detectors are used to calibrate the excore detectors, after the excore calibration period has expired (a month), the reactor must go to MODE 3 (hot standby).

The proposed Ft. Calhoun TS includes the existing requirement that if the recalibration of the excore detectors has not been accomplished within the previous 30 equivalent full power days, the APD monitoring limits and trip setpoints, be reduced by 0.03 ASI units. If the recalibration of the excore detectors has not been accomplished within the previous 200 equivalent full power days, the power is limited to less than that corresponding to 75% of the peak linear heat rate.

This restriction is based on a study that showed a .03 drift in the excore measurement of the ASI in 200 days for Ft. Calhoun.

While this restriction is not as severe as the requirement of the Standard TSs, it is an existing requirement based on data from Ft. Calhoun. We consider the combination of the Standard TSs and the existing Ft. Calhoun TSs to be acceptable.

2. Independent Auditing of Cycle 6 Set Points

As part of the review of proposed changes to the TSs for Cycle 6, we performed two independent analyses to audit the numbers being proposed by the licensee.

These two audit calculations are: 1) a calculation of the power at which incipient fuel melting will occur to compare with the licensee's value of 21 kw/ft and 2) an independent calculation of the TM/LP safety limits and trip equation.

(1) Power to Fuel Melting

The calculation of the power to fuel melting was done using the GAPCON THERMAL II computer code. Several different cases were considered.

1. A best estimate calculation at Beginning of Life (BOL)
2. A conservative calculation at BOL
3. A best estimate calculation at 1000 Mwd/MTU
4. A conservative calculation at 1000 Mwd/MTU
5. Case 1 with a fuel temperature uncertainty of 256°C added

(The 256°C is a standard error of estimate obtained from comparing the FRAPS 3 computer program with a wide variety of fuel temperature data.)

The difference between the conservative and best estimate calculations is the assumptions made in densification and fuel pellet relocation models.

All cases gave a power to fuel melting greater than 21 kw/ft. Therefore, we consider the licensee's calculation acceptable.

(2) TM/LP Trip Equation

We contracted with Battelle Pacific Northwest Laboratories

(BPNL) to independently calculate a TM/LP trip equation for Ft. Calhoun Cycle 6 and to review the methods used by the licensee to derive the TM/LP trip equation. This work was part of the review of Reference 10 which is still continuing. BPNL's conclusion after completing their work is that although the modeling and calculational techniques are questionable in some areas, the assumptions and forced conditions are so conservative that the TM/LP trip function as derived by ENC is adequate.

A detailed list of our findings from these audit calculations is given in summary form in Table 4. The complete BPNL analysis is given in Reference 34.

We followed the work of BPNL, reviewed their methods and discussed their findings with them.

We agree with the BPNL conclusion and agree that the specific equation derived for Cycle 6 is sufficiently conservative. However, we find that the general method requires additional justification and will continue the review of the ENC setpoint methods as described in Reference 10.

PHYSICS STARTUP TESTS

The revised Startup Physics Testing summary as submitted in Reference 28 has been reviewed. This program includes hot functional, low power physics and power ascension tests. During the hot functional phase, surveillance tests are performed to check CEA position indication and all other interlock and control features of the rod drive system. The low power physics tests include initial criticality, critical boron concentration, isothermal temperature coefficient, CEA group worth and CEA

TABLE 4 : SUMMARY OF TM/LP REVIEW

1. More flux shapes should be used for the TM/LP analysis
2. An actual quarter core symmetric model rather than an approximation to symmetry should be used.
3. The ENC subchannel model should be normalized to unity.
4. There was some disagreement in calculated values between those reported by the licensee and those calculated by BPNL.
5. The ENC methodology for calculating the TM/LP trip function should be changed to not average α and β, but determine the most restrictive values in a more physically meaningful or mechanistic manner.
6. The PF(B) function appears to have no physical meaning and its derivation is not clear.
7. The general methodology should be changed to conduct transient analysis of potential pressure, power, and temperature changing incidents. This would allow the establishment of a documented buffer zone between the calculated TM/LP trip function and the 1.30 MDNBR lines. This buffer zone would exist to preclude the exceeding of thermal hydraulic limits during transients.
8. The TM/LP equation proposed for Cycle 6 will provide sufficient protection to the reactor for Cycle 6. However, we are not assured that the method used to derive this equation, because of the concerns listed above, will provide a suitably conservative equation for future cycles.

symmetry check tests. The power ascension tests include power distribution tests at 50%, 70% and 100% power as well as isothermal temperature coefficient, power coefficient and critical boron concentration tests at 100% power.

For each test, the acceptance and review criteria are stated as well as the remedial actions if these criteria are not met. The results of the startup testing will be reported to NRC and differences between measured and predicted values greater than acceptance and/or review criteria will be discussed.

This entire program meets our startup physics test guidelines of our November 28, 1978 memorandum. This program is acceptable for the Cycle 6 startup of Ft. Calhoun. This program is also acceptable for all future startup following reloads unless a reload involves major changes which require additional confirmatory tests.

SUMMARY - CYCLE 6 OPERATION

For reasons discussed above, it is our conclusion that operation of Ft. Calhoun for Cycle 6 at 1420 MWt is acceptable with safety analysis and core limits based on 1500 MWt.

CONCLUSION - CYCLE 6 OPERATION

We have 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 action will not be inimical to the common defense and security or to the health and safety of the public.

III. HEATUP AND COOLDOWN TEMPERATURE-PRESSURE LIMITS

DISCUSSION

10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including AOOs.

The pressure-temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel belt-line region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principle effect of the neutron irradiation is that it causes the vessel material nil-ductility temperature (RT_{NDT}) to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation induced increase in RT_{NDT} by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure-temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel are maintained within acceptable limits.

EVALUATION

The licensee submitted revised pressure-temperature operating limits for Ft. Calhoun for the operating period from 4.3 to 5.2 effective full power years (EFPY). The proposed operating limits were calculated in accordance with Appendix G, 10 CFR Part 50. They also requested that Specification

2.1.1(6) be deleted. This specification requires that reactor coolant system leak tests be conducted at normal operating pressure and less than 400°F.

We have performed independent calculations to verify compliance with Appendix G, 10 CFR Part 50. Our calculations show that at the end of service life the fluence on the vessel wall ID will be 4.4×10^{19} n/cm^2. Based on this value the fluence will be 4.3×10^{18}n/cm^2 at the 1/4T location in the vessel wall at 5.2 EFPY. From the test results on material surveillance specimens and the damage predictions (changes in RT_{NDT}) in Regulatory Guide 1.99, Revision 1, we calculate the increase in RT_{NDT} of the vessel weld metal (the limiting vessel material) to be 215°F at 5.2 EFPY. This agrees with the value calculated by the licensee. From our review, we conclude that the proposed operating limits are acceptable and are in accordance with Appendix G, 10 CFR Part 50.

We reviewed the deletion of Specification 2.1.1(6): "Reactor coolant system leak tests shall be conducted at normal operating pressure and less than 400°F". We have determined that instead of deleting that specification the following should be substituted: "Reactor coolant system leak and hydrostatic tests shall be conducted within the limitations of Figs. 2-1A and 2-1B". This change has been discussed with OPPD and has been incorporated into the TSs.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because this action does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, this action does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this action will not be inimical to the common defense and security or to the health and safety of the public.

V. ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR § 51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Dated: APRIL [illegible] 1980

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