GA-C19661

SAFETY ANALYSIS REPORT FOR CYCLE 4
EXTENDED OPERATION

FORT ST. VRAIN NUCLEAR
GENERATING STATION

GENERAL ATOMICS PROJECT 1900

MAY 1989

GENERAL ATOMICS

CONTENTS

FIGURES

TABLES

1. INTRODUCTION AND SUMMARY

This Safety Analysis Report (SAR) is prepared to assess extended operation of the Fort St. Vrain Nuclear Generating Station (FSV) in the current reload cycle (Cycle 4). Operation of the plant during Cycle 4 up to a total of 300 effective full power days (EFPD) has been evaluated in Ref. 1. Public Service Company of Colorado (PSC) has decided to cease nuclear operations at FSV no later than June 30, 1990. Cycle 4 will be the last reload cycle at FSV, and extension of the cycle beyond 300 EFPD may be necessary while PSC prepares for permanent shutdown, defueling, and decommissioning. The total duration of Cycle 4 has been analyzed up to 520 EFPD.

This report contains sections describing the operating history of the reactor through December 31, 1988, evaluations of nuclear, thermal-hydraulic, and mechanical performance of the core, and the safety aspects of the core during extended operation of Cycle 4 up to 520 EFPD.

A safety evaluation for extended operation is presented in this report. It is concluded that extended operation of Cycle 4 up to 520 EFPD presents no unreviewed safety questions, as defined in 10CFR50.59 and requires no changes to the Technical Specifications.

2. REACTOR OPERATING HISTORY

Initial criticality of the FSV reactor was achieved on January 31, 1974, with initial generation of electricity on December 11, 1976. Prior to February 1, 1979, when the plant was shutdown for refueling, the initial core had operated a total of 174 EFPD. Cycle 2 operation began on May 26, 1979 and was completed on May 13, 1981, having accumulated a total of 189 EFPD. Cycle 3 operation began on July 15, 1981, and was completed on January 20, 1984, having accumulated a total of 294.5 EFPD. Cycle 4 operation began on May 16, 1984, and as of December 31, 1988 had accumulated a total of about 154.5 EFPD.

The nuclear performance of the FSV core has been, in general, as predicted. Good agreement between measurements and calculations has been obtained for shutdown margins, temperature coefficient, xenon worth, and control rod worth (i.e., measurements are well within the acceptance criteria specified for the tests). Initial cold criticalities in Cycles 1 through 4 were predicted within 0.003 Δk. Analyses have overpredicted the end-of-cycle (EOC) reactivities of the core at operating temperatures by a few tenths of a percent; however, the difference between observed and expected reactivity has remained within the 0.01 Δk limit of Technical Specification LCO 4.1.8 throughout operation.

Fission product release to date has been very low. Measured circulating activity has been approximately a factor of 30 less than the limit provided in Technical Specification LCO 4.2.8. Measurements of plateout activity obtained after removal of the first plateout probe in November 1981 indicate that these activity levels are also substantially below Technical Specification limits (Ref. 2).

The most unusual occurrence, to date, was the detection, initially in October 1977, of temperature fluctuations. These fluctuations affected the nuclear channels, the region exit temperatures, and the steam generator module temperatures. During fluctuations, however, the total core coolant flow and core thermal power remained essentially constant. In addition, the temperature swings during fluctuations stayed within plant operations and technical specification limits.

A comprehensive program to evaluate and resolve the fluctuation issue was begun in late 1977. This program led to installation, in November 1979, of core region constraint devices (RCDs) (Ref. 3). These devices limit the small (approximately 0.10 in.) lateral movements of fuel columns to which the fluctuations were attributed. Fluctuation testing of the core up to 100% power with RCDs installed was completed in November 1981. No fluctuations have been detected since installation of the RCDs. The results of these tests were formally submitted to NRC in July 1982 (Ref. 4).

A second major issue with regard to reactor operation has been the existence of discrepancies between measured and calculated region outlet helium temperatures. The largest discrepancies have been limited to regions in the northwest boundary of the core (Regions 20 and 32-37), with measured temperatures being consistently less than calculated temperatures. These discrepancies are caused by a transverse flow of relatively cool helium from the core-reflector interface along the inside of the region outlet thermocouple sleeves (Type II flow). This flow passes over the region outlet thermocouple assemblies of these regions and depresses the indicated region outlet temperature.

To compensate for these discrepancies, special operating procedures were provided which insure compliance with the original core design intent. Technical Specification LCO 4.1.7/SR 5.1.7 governs operation with these measurement errors.

On October 5, 1982 NRC issued Amendment No. 28 to the FSV Operating License. In this amendment, the NRC concluded, based upon a review of Ref. 4, that the fluctuation issue is resolved. The technical specifications proposed in Ref. 5 were incorporated in the operating license, and all previously imposed restrictions on reactor power level were removed.

As a result of the visual examinations conducted on fuel elements removed from the reactor during the second refueling, two Segment 2 fuel elements were each found to have one or two cracked graphite webs. The presence of these cracks did not affect the cooling geometry of the fuel or the ability of the fuel handling machine to safely remove the fuel elements from the core. A DOE-funded program was carried out at General Atomics (GA) to investigate this issue, while a similar NRC-funded program was conducted at Los Alamos Laboratory. Results of the cracked web program were submitted to NRC by PSC in Refs. 6 through 8, which showed that the cracks were caused by high localized in-plane tensile stresses resulting from high fluence and large gap coolant flows, and that the cracked webs have no effect on the ability of the elements to perform their functions safely. The NRC concurred with these results and closed the issue of fuel element web cracking on December 30, 1986 (Ref. 9). The examination of Segment 3 fuel elements, removed during the third refueling, indicated no cracked graphite webs.

The replacement fuel elements for Cycle 4 featured one design change (use of H-451 graphite) relative to the fuel design described in the FSAR. This design change was the subject of a lengthy generic review and approval by NRC in 1978 and 1979. The safety evaluation for the change as it affected Cycle 4 core reload was presented in Ref. 1, and the NRC approved the use of H-451 graphite in FSV via Amendment No. 40 to the FSV Operating License.

A significant operating event occurred on June 23, 1984. As reported in Ref. 10, following an automatic scram action, six control rod pairs failed to automatically insert and had to be driven into the core. Investigations determined that shim motor bearing wear and debris buildup were the primary contributors. Corrective action included a comprehensive control rod drive (CRD) refurbishment program and implementation of interim Technical Specifications for reactivity control. In Ref. 11, PSC committed to operate FSV in accordance with procedures based upon the interim Technical Specifications until formal specifications are approved and implemented.

Cycle 4 operation has been limited since it began in May 1984. The plant was shutdown for most of 1984, 1985, 1986, and 1987 for engineering modifications to the control rod drives, environmental qualification (EQ) of safety-related electrical equipment, helium circulator removal and replacement, and for recovery from a fire in the turbine building. The plant was shutdown throughout the second half of 1988 for circulator refurbishment and removal of moisture from the reactor vessel.

Based on economic considerations associated with the ongoing operating costs of FSV, PSC has decided to cease nuclear operations on or before June 30, 1990 (Ref. 12). The final duration of Cycle 4 will be based upon the time to complete planning and preparations for the initiation of defueling and upon the reliability of FSV.

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3. NUCLEAR PERFORMANCE

In this section the effects of extended operation of Cycle 4 up to a total of 520 EFPD on the nuclear performance of the core are presented. Nuclear analyses were carried out using the same methods applied to the analyses presented in the FSAR, previous reload SARs, and the semiannual fuel accountability reports. Extended operation of Cycle 4 introduces no new aspects to high-temperature gas-cooled reactor (HTGR) core design or analysis techniques; consequently, there was no need to develop or adapt any new methods or procedures for the nuclear performance analysis.

The depletion analyses described in this chapter were performed by simulating the actual core power history for Cycles 1, 2, and 3 and the first 154.5 EFPD of Cycle 4. Continuous operation at 100% power for the balance of Cycle 4 was assumed. Cycle 4 was assumed to continue to a total length of 520 EFPD.

During the extended operation of Cycle 4, all control rod groups except the regulating rod, which remains partially withdrawn from the central region, will gradually be fully withdrawn. Therefore, the only means of introducing positive reactivity in compensation for the reactivity decrease due to the burnup of fuel will be to decrease the fuel temperature and the reactor power as shown in Table 3-1. Although rated power cannot be maintained throughout extended operation, it was conservatively assumed in the depletion analyses that operation from 154.5 EFPD to 520 EFPD would take place at 100% of rated power.

3.1 HEAVY METAL AND BURNABLE POISON LOADINGS

It is anticipated that the reactor will have operated for up to 957.5 EFPD, generating a total of up to 1.9×10^7 MWhr of energy at the nominal end of Cycle 4 (300 EFPD). The projected heavy metal loadings in the core segments at the EOC 4 (957.5 EFPD) are given in Table 3-2. The maximum burnup in fissile particles is projected to be about 16.0% FIMA and in fertile particles about 3.4% FIMA. These burnups are lower than the limiting values given in the FSAR, Appendix A, Table A.2-2. The maximum projected fast flux ($E > 0.18$ MeV) exposure in Segment 4 is about 4.2×10^{21} nvt. This exposure is also lower than the limiting values given in the FSAR.

It is anticipated that at the end of Cycle 4 extended operation, the reactor will have operated for up to 1177.5 EFPD, generating up to 2.34×10^7 MWhr of energy. The maximum fuel element irradiation allowed by Technical Specification LCO 4.1.1 is 1800 EFPD. The projected heavy metal loadings in the core segments at the extended EOC 4 (1177.5 EFPD) are given in Table 3-3. The maximum burnup in fissile particles is projected to be about 17.4% FIMA, and in fertile particles about 4.4% FIMA. These burnups are also lower than the limiting values given in the FSAR. The maximum projected fast flux exposure in the core at 1177.5 EFPD is about 5×10^{21} nvt, also lower than the limiting values given in the FSAR.

The burnable poison loaded into Segment 9 fuel elements at BOC 4 will be essentially totally depleted by the nominal EOC 4. No burnable poison adjustments will be required for Cycle 4 extended operation.

3.2 CONTROL ROD SEQUENCE

[redacted] LCO 4.1.3 states that a control rod sequence will be specified for each fuel cycle and that the sequence will always be followed, except for rod insertion resulting from a scram or rod runback or during low-power physics testing. Interim Technical Specification LCO 3.1.5 has similar requirements. The control rod sequence for use during

Cycle 4 is given in Table 3-4. The identification of the control rod groups is shown in Figure 3-1. Extended operation of Cycle 4 up to an additional 220 EFPD will require no changes to the control rod sequence.

The regulating rod is located in the central refueling region (rod group 1). This group is partially withdrawn before criticality is achieved and then maintained in its most reactive control rod position for the remainder of the operation. In this manner, minor reactivity adjustments can be made most rapidly with the minimum amount of control rod motion. This is consistent with the method of operation utilized for the control rods in previous cycles, including Cycle 4.

A summary of the calculated power peaking factors obtained during a typical rise-to-power at the nominal EOC4 using the control rod sequence is given in Table 3-5. It was conservatively assumed that the reactor was shutdown for 90 days prior to the startup. The results in Table 3-5 show the correlation between control rod group insertion and the core power level. Power peaking factors during Cycle 4 extended depletion are discussed in Section 3.3.

~~The basis of~~ Technical Specification LCO 4.1.3 contains maximum values of region peaking factors. At power less than 20% (core outlet gas temperature $\leq$950°F), the maximum RPF is 3.0 and the maximum tilt is 1.61. At powers between 20 and 60% (outlet temperature from 950°F to 1250°F) the applicable limits are an RPF of 2.15 and a tilt of 1.34 to 1.46. At higher powers, above 60% (outlet temperature >1250°F), the limits are an RPF of 1.83 and a tilt of 1.34 to 1.46.

From the data given in Table 3-5, it can be seen that at the nominal EOC4 the calculated power peaking factors for the various power levels do not exceed those given in the basis of Technical Specification LCO 4.1.3. This is true for both the radial region peaking factors and the intra-region peaking (column tilt) factors. The analyses indicate that at powers between 0 and 2%, the maximum RPF may slightly exceed the LCO 4.1.3 basis maximum value of 3.0. A two-foot insertion of group 2B until group 4E is fully withdrawn, as allowed by Interim Technical Specification LCO 3.1.5, will correct this situation, if necessary.

For rise-to-power during extended operation of Cycle 4 to 520 EFPD, criticality and subseque t power levels are reached with fewer control rods inserted in the core. He ce, power peaking factors are reduced relative to those shown in Table 3-5.

3.3 PROJECTED CYCLE 4 EXTENDED OPERATION

This section presents the results of Cycle 4 extended depletion analyses using design methods discussed in Section 3.5 of the FSAR. Fuel and burnable poison loadings discussed previously were used as input (see Section 3.1). The total Cycle 4 burnup of 520 EFPD was carried out using actual power history up to 154.5 EFPD, and assuming operation at rated power for the remainder of the cycle.

Figures 3-2a and 3-2b present envelopes encompassing projected RPFs and column tilts during Cycle 4 depletion to 520 EFPD. The results indicate that RPFs and tilts during the extended Cycle 4 will be well within the maximum values contained in the basis of LCO 4.1.3. Envelopes are not presented for fully rodded regions because there are no fully inserted control rods during extended depletion at high powers.

Axial zoning of the fuel and burnable poison is provided (1) to produce a power distribution which tends to reduce axial fuel temperature peaking, and (2) to maintain the desired axial power distribution with depletion. The calculated axial power factors in the bottom layer of each fuel region during Cycle 4 are shown in Table 3-6. The calculations were carried out with the GATT code, the three-dimensional whole core model which has been used in the semi-annual fuel accountability analyses for the past three cycles and for the early portion of Cycle 4. It can be seen that the limits on peaking factor in the lower fuel layer assumed in the basis of [redacted] are not exceeded. These calculations were carried out to 520 EFPD.

The basis of Technical Specification LCO 4.1.3 also states that an acceptable flux distribution shall be maintained at lower power levels by keeping the flux level in the center of the core at least as high as the average level. Table 3-7 shows the ratio of the flux in the inner core regions (Regions 1 through 19) to the core average flux for control rod configurations in Table 3-5 which can result in operation between 0% and about 20% power at the nominal EOC4. The flux ratio is above 1.0 for all cases, consistent with the basis of LCO 4.1.3. With further burnup during extended operation, control rods will be removed from the core until all control rods except the regulating rod in the central refueling region will be removed from the core. The flux ratio with this control rod configuration was calculated to be between 1.12 and 1.13 throughout extended operation.

3.4 MAXIMUM CONTROL ROD WORTH

The basis of Technical Specification LCO 4.1.3 states that the accidental removal of the maximum worth single rod pair shall result in a transient with consequences no more severe than the withdrawal of 0.012 ΔK, at rated (i.e., 100%) power, from a core which has a temperature defect between 220°F and 1500°F of 0.028 ΔK. In addition, the calculated worth of any rod pair in any configuration with the reactor critical must be less than 0.047 ΔK. The same requirements are contained in Interim Technical Specification LCO 3.1.5. The rod withdrawal accident (RWA) at full power evaluated in Section 14.2 of the FSAR assumes withdrawal of a control rod worth of 0.012 ΔK at equilibrium EOC with an equilibrium EOC temperature defect of 0.028 ΔK. Because the consequences of an RWA are a function of rod worth, steady-state core temperature (i.e., initial power level), and temperature coefficient (which varies with burnup during the cycle), it is necessary to evaluate control rod worth as a function of control rod insertion.

The control rod withdrawal sequence for the extended Cycle 4 is described in Section 3.2. For this sequence the maximum control rod worths at nominal EOC4 are shown in Table 3-8. The results in Table 3-8 indicate that the maximum worth rod pair in any source power critical configuration during Cycle 4 is 0.018 ΔK, which is less than the 0.047 ΔK limit ~~of LCO 4.1.3 and~~ Interim LCO 3.1.5.

As was previously discussed, under most circumstances all control rods with the exception of the regulating rod are fully withdrawn during the extended depletion of Cycle 4. Under these circumstances the RWA is by default limited to the regulating rod. The worth of this rod (from 115 inches to fully withdrawn) is only 0.002 ΔK, i.e., much less than the 0.012 ΔK used for the FSAR analysis. The consequences of RWA of the regulating rod are negligible compared with those discussed in the FSAR. Furthermore, the power level during Cycle 4 extension is less than 100%, so a larger temperature defect is available to mitigate the RWA.

However, as shown in Table 3-5, at the beginning of extended operation (i.e., at 300 EFPD in Cycle 4), especially after a prolonged shutdown, there is sufficient excess reactivity (due to decay of xenon and conversion of Pa-233 into U-233) to allow core operation for a brief time at the rated power. Consequently, the RWA must be evaluated for these conditions. It was conservatively assumed that the reactor is shutdown for 90 days prior to beginning extended operation, resulting in reactivity buildup due to xenon and Pa-233 decay.

As indicated in Table 3-5 operation at rated power can be achieved at 300 EFPD with control rod group 3B inserted 30% into the core. As shown in Table 3-7, the maximum worth fully inserted rod in this configuration is only 0.008 ΔK. The maximum worth rod with group 3B fully inserted is 0.014 ΔK, but with group 3B inserted only 30% into the core the worth of any control rod in that group is less than 0.008 ΔK. As is shown in Section 3.6, the temperature defect between 220°F and 1500°F is larger than 0.028 ΔK at 300 EFPD. Therefore, the consequences of an RWA at the nominal EOC4 at rated power are less than those of the RWA described in Technical Specifications.

Table 3-8 indicates that with two rod groups fully inserted, a maximum control rod worth of 0.014 ΔK is obtained. However, Table 3-5 indicates that the maximum power level that can be achieved in this configuration is from 40% to 60% of rated power. The difference in average fuel temperature between these conditions and operation at rated power make an additional 0.006 ΔK in temperature defect available to mitigate the consequences of an RWA. Therefore, the consequences of an RWA of an 0.014 ΔK rod at 60% power at 300 EFPD are the same as those of an 0.008 ΔK rod at 100% power, which are in turn, as discussed above, less than those of the RWA described in the Technical Specifications.

Similar evaluations of the RWA for all configurations shown in Table 3-8 indicate that in all cases the RWA consequences at nominal EOC are bounded by those of the RWA described in the Technical Specifications. As core operation is extended to 520 EFPD, the power level associated with the maximum control rod worth decreases, thereby providing more temperature defect available to mitigate the consequences of an RWA. The total temperature defect, as discussed in Section 3.6 remains above 0.028 ΔK. It is, therefore, concluded that the consequences of an RWA during Cycle 4 extended operation are bounded by those of the RWA discussed in Section 14.2 of the FSAR and in the Technical Specifications.

3.5 CORE SHUTDOWN MARGINS

3.5.1 Control Rod System SDM

Interim Technical Specifications LCO 3.1.4/SR 4.1.4 state that a shutdown margin (SDM) of 0.01 ΔK shall be achieved under the following conditions:

1. The highest worth control rod pair is fully withdrawn and not insertable, all inoperable rod pairs are at their pre-scram positions, the core average temperature is at 220°F, and Xe-135, Sm-149, and Pa-233 levels are equal to those at the time of shutdown. These SDM calculations should assume shutdown from operation at 100% power with equilibrated xenon, samarium, and protactinium.

2. The highest worth control rod pair is fully withdrawn, inoperable rod pairs are in their known position or assumed fully withdrawn, the core average temperature is at 80°F, xenon is fully decayed, samarium is fully built up, and protactinium converts into U-233 as a function of time after shutdown

In assessing SDM for the extended Cycle 4, it is assumed that the two highest worth control rod pairs are fully withdrawn. This assumption is consistent with Interim Technical Specification LCO 3.1.1 and the single failure criterion. These SDM calculations provide an estimate of the time after shutdown available to reinsert the second highest worth control rod, if necessary to maintain an 0.01 Δk SDM. It also must be shown that a SDM of at least 0.01 ΔK can be maintained indefinitely with full protactinium decay after insertion of the second highest worth control rod or insertion of reserve shutdown system (RSS) material in one or both of the regions with withdrawn control rods.

The SDM as a function of burnup in the extended Cycle 4 is given in Table 3-9. The shutdown time is defined as the time for which the core is subcritical following a scram in which the maximum worth rod(s) failed to insert. The first row in Table 3-9 indicates the SDM under normal scram conditions where all rods are inserted. The results in the other rows of Table 3-9 indicate that, after a core shutdown, there is no limit on time available to repair and to reinsert at least one of the inoperable control rods. Additional SDM could be achieved by activating the RSS or reinserting the second highest worth control rod. The SDMs increase systematically with burnup during the Cycle 4 extended operation. The results in Table 3-9 are consistent with the results calculated for the Segment 9-Cycle 4 SAR (see Table 5-10 of Ref. 1), which show that the minimum SDM occurs at the middle of Cycle 4, and then starts to increase systematically to the nominal EOC4.

On the basis of the above presentation, it may be concluded that the control rod system is adequate to provide an adequate SDM for the FSV core under all normal and postulated accident conditions during extended operation of Cycle 4. General requirements for operability of control rod drives during the cycle are provided in Interim Technical Specification LCO 3.1.1.

3.5.2 Reserve Shutdown System SDM

As noted above, Interim Technical Specifications LCO 3.1.4/SR 4.1.4 state that the reactor SDM shall be greater than or equal to 0.01 ΔK. Specifically, for reactivity control with the reserve shutdown system (RSS) Interim Technical Specifications LCO 3.1.8/SR 4.1.8 state that with any one RSS unit inoperable core operation may continue provided that the unit is capable of being made operable within 14 days following a reactor shutdown. Furthermore, in the basis of Interim LCO 3.1.8/SR 4.1.8 it is stated that the RSS must be capable of achieving reactor shutdown in the event that the control rod system fails to insert.

On the basis of above considerations, the SDM calculations for the RSS were carried out with the following conservative assumptions:

1. The core, prior to shutdown, was operated at rated power long enough to equilibrate xenon and Pa-233.

2. The scram signal fails to insert any withdrawn control rods.

3. The inoperable RSS unit is the maximum worth one.

4. The core is at room temperature (80°F).

5. The worth of RSS in rodded regions is neglected.

The calculated SDMs as a function of burnup in the extended Cycle 4 are given in Table 3-10. These results indicate there is no time limit for the repair and insertion into the core of inoperable control rods and/or RSS units before the SDM becomes inadequate.

In the basis of Interim LCO 3.1.8/SR 4.1.8 it is stated that a worth of RSS of 0.100 ΔK is sufficient to ensure SDM during the first 14 days of Pa-233 decay. Calculations indicate that the worth of RSS in the extended Cycle 4, without any control rods present in the core (as may be the case at the end of the extended Cycle 4) is 0.111 ΔK. On the basis of SDMs and the total worth of RSS, it may be concluded that the RSS during the extended Cycle 4 meets or exceeds the reactivity control requirements.

3.6 KINETICS PARAMETERS

The kinetics parameters for the extended Cycle 4, as well as for the initial and equilibrium cycles (taken from the FSAR), are given in Table 3-11. The data in this table indicate that the equilibrium cycle kinetics parameters are in close agreement with the extended Cycle 4 kinetics.

Interim Technical Specification LCO 3.1.7 requires that the reactivity change due to an average core temperature increase between 220°F and 1500°F (refueling temperature to rated power conditions) be at least as negative as -0.031 ΔK and no more negative than -0.065 ΔK during Cycle 4. This requirement is imposed because FSAR accident analyses assumed a temperature defect of -0.028 ΔK, and the uncertainty in measured temperature defect is about ±10% or 0.003 ΔK. The calculated temperature defect decreases with burnup during each cycle due to thorium depletion and U-233 buildup.

The calculated temperature defect during the extended Cycle 4 is shown in Fig. 3-3. The results indicate that the temperature defect between average core temperatures of 220°F and 1500°F is -0.037 ΔK at 390 EFPD and

-0.034 ΔK at 520 EFPD. Both of these calculated values meet the requirements of Interim LCO 3.1.7 for measured temperature defect.

3.7 NUCLEAR DETECTOR DECALIBRATION

The power range nuclear detector signals, used by the control system to initiate plant protective system (PPS) action, exhibit significant decalibration due to control rod motion. This detector decalibration is accommodated by a reduction in the fixed PPS setpoints of Technical Specifications SL 3.3 and LCO 4.4.1 and frequent recalibration of the detectors. The reduced setpoints have been reevaluated each cycle because of the different control rod withdrawal sequence and the different fuel loading distribution. A rigorous analysis was done to determine these reduced setpoints for Cycle 1. For subsequent cycles including Cycle 4 the reduced setpoints have been determined as follows:

- o Calculate the "worst-case" detector decalibration factor (DF) for each control rod group (i.e., the case which would most delay the PPS trip).

- o If the "worst-case" DFs indicate less delay in the PPS trip than the previous cycle, use the reduced setpoints from the previous cycle.

- o If these "worst-case" DFs indicate more delay in the PPS trip than the previous cycle, then the reduced setpoints and/or the detector recalibration schedule for the previous cycle must be reevaluated.

The control rod sequence for the extended operation of Cycle 4 is the same as that for the reference cycle. Furthermore, the extended operation of Cycle 4 is characterized by a deficiency of excess reactivity, i.e., all shim banks are expected, except for a few special cases discussed in Section 3.4, to be fully withdrawn. Consequently, during the expected mode of operation there will be very little control rod motion, and the decalibration of the detectors will be minimal.

However, during a rise-to-power operation early in the extended cycle, as was shown in Table 3-5, it is possible to have one or more rod groups fully or partially inserted to achieve core reactivity control. Consequently, decalibration of detectors may occur in this brief period (relative to the total extended burnup duration). The DF calculations for Cycle 4 indicated that the burnup effect is not significant, i.e., only one set of setpoints is needed to cover the burnup from 0 to 300 EFPD. To retain these setpoints during the extension of Cycle 4 it is necessary to show that the worst possible DFs during extended operation are about the same as those for the first 300 EFPD. The built-in conservatism of setpoints can accommodate small changes between the DFs of the reference cycle and the extended cycle.

The worst DFs calculated for the reference and extended Cycle 4 are given in Table 3-12. When the extended Cycle 4 DFs stay the same or increase, the reference setpoints are more conservative to use during the extended cycle. The extended cycle DFs for rod groups 4A, 3A, and 3B are slightly lower, i.e., less conservative than those of the reference Cycle 4. However, the reference setpoints remain conservative. As shown in Table 3-5, group 4A is involved at power operations between 0% and 2% of rated. The reference setpoint for the high power reactor scram at these powers is currently 64% power, i.e., far below 0.68 x 140 = 95%. Full insertion of group 3A could occur at power operations between 18% and 28% of rated. The high power scram setpoint for these powers is also currently at 64% power, well below 0.90 x 140 = 126%. Group 3B may be involved in core operations at higher powers including 100% (under the special core conditions discussed in Section 3.4). In this case the high power scram setpoint at rated power should be 0.79 x 140 = 110%, which is less than the current setpoint of 115%. However, at the current limit on core power of 82%, the high power scram setpoint is currently set at 105% power, which is less than the necessary setpoint of 110% Furthermore, the detector calibration schedule for the reference Cycle 4 specifies that the Group 3B should be calibrated at 85 to 105 inches of withdrawal. Of course the burnup of Cycle 4 during extended operation will result in a further loss in the core reactivity. This in turn will result in the use of all rod groups, including Group 3B, occurring at ever lower powers, thus increasing the margin between the setpoint and the actual core power.

Therefore, it is concluded that the reference setpoints for Cycle 4 are equally applicable for the extended operation of Cycle 4.

TABLE 3-1

PROJECTED POWER HISTORY
FOR CYCLE 4 EXTENDED OPERATION

CALENDAR DAYS INTO EXTENSION	CUMULATIVE CYCLE 4 EFPD	REACTOR POWER %	AVERAGE FUEL TEMP DEGREES F
0	300	80	1373
75	360	80	1373
118	390	70	1332
176	425	60	1242
236	455	50	1182
324	490	40	1116
424	520	30	1026

TABLE 3-2

PROJECTED CORE LOADINGS AT THE NOMINAL END OF CYCLE 4(a)

Nuclide	Nuclide Weight (kg)/Segment						Total Nuclide Weight
	4	5	6	7	8	9	
Th232	2,309.2	2,773.9	2,306.9	2,196.9	2,215.5	2,246.4	14,048.8
Pa233 + U233	50.0	59.8	50.0	44.2	38.4	24.2	266.7
U234	6.4	7.5	6.4	5.1	3.0	2.4	30.8
U235	33.9	37.6	33.9	53.6	74.3	122.6	355.9
U236	18.1	20.0	18.1	20.5	18.8	12.9	108.3
U238	7.5	8.3	7.5	9.4	10.1	11.4	54.1
Plutonium	0.8	0.9	0.8	0.8	0.6	0.5	4.4
Total	2,425.9	2,908.0	2,425.9	2,330.6	2,360.7	2,420.4	14,871.5

(a)957.5 EFPD

TABLE 3-3

PROJECTED CORE LOADINGS AT THE END OF EXTENDED CYCLE 4[a]

Nuclide	Nuclide Weight (kg)/Segment 4	5	6	7	8	9	Total Nuclide Weight
Th232	2,289.1	2,744.3	2,287.0	2,169.8	2,199.4	2,214.9	13,904.5
Pa233 + U233	49.6	60.3	49.5	47.9	45.4	40.0	292.7
U234	8.0	9.9	7.9	7.2	5.8	4.6	43.4
U235	24.9	26.3	25.1	38.9	54.9	84.3	254.4
U236	19.2	21.5	19.2	22.9	22.1	19.9	124.8
U238	7.1	7.9	7.2	9.0	9.6	10.5	51.3
Plutonium	1.2	1.3	1.1	1.3	1.0	0.7	6.6
Total	2,399.1	2,871.5	2,397.0	2,297.0	2,338.2	2,374.9	14,677.7

[a] 1177.5 EFPD

TABLE 3-4

CONTROL ROD SEQUENCE FOR CYCLE 4

Sequence	Group Withdrawn	Regions
1	2A(a)	2,4,6
2	4F(a)	25,31,37
3	4D	23,29,35
4	1(115" out)	1
5	4B	21,27,33
6	2B	3,5,7
7	4E	24,30,36
8	4A	20,26,32
9	4C	22,28,34
10	3C	10,14,18
11	3A	8,12,16
12	3B	9,13,17
13	3D	11,15,19
14	1(fully out)	1

(a) Rod groups used for rod runback.

TABLE 3-5

TYPICAL RISE-TO-POWER AT NOMINAL EOC4 (300 EFPD)

Power, %	Group Insertion Fraction	Max RPF	Max Tilt Rodded	Max Tilt Unrodded
0	4E @ 0.5	N/A	N/A	N/A
2	4C @ 0.5	2.13	1.44	1.29
8	4C @ 0.07	1.94	1.35	1.25
18	3C @ 0.19	1.93	1.29	1.28
28	3A @ 0.5	1.78	1.28	1.26
40	3A @ 0.05	1.74	1.21	1.23
60	3B @ 0.71	1.73	1.21	1.24
70	3B @ 0.56	1.72	1.22	1.24
85	3B @ 0.46	1.72	1.22	1.24
100	3B @ 0.30	1.71	1.23	1.25

TABLE 3-6

SUMMARY OF CONTROL ROD INSERTIONS AND AXIAL POWER FACTORS IN BOTTOM FUEL LAYER

CYCLE 4 REG/EFPD	*300.0		350.0		390.0		435.0		470.0		520.0	
1	2	.704	2	.760	2	.758	2	.768	2	.773	2	.789
2	0	.615	0	.672	0	.671	0	.680	0	.685	0	.700
3	0	.575	0	.628	0	.628	0	.640	0	.646	0	.663
4	0	.564	0	.619	0	.619	0	.629	0	.635	0	.652
5	0	.559	0	.615	0	.613	0	.623	0	.628	0	.644
6	0	.611	0	.671	0	.669	0	.680	0	.685	0	.700
7	0	.616	0	.675	0	.673	0	.683	0	.688	0	.703
8	0	.564	0	.622	0	.622	0	.634	0	.640	0	.656
9	0	.611	0	.665	0	.665	0	.676	0	.682	0	.698
10	0	.550	0	.607	0	.607	0	.618	0	.624	0	.641
11	0	.560	0	.628	0	.630	0	.643	0	.649	0	.666
12	0	.603	0	.664	0	.664	0	.674	0	.680	0	.697
13	0	.558	0	.608	0	.607	0	.619	0	.625	0	.643
14	0	.586	0	.645	0	.644	0	.656	0	.662	0	.679
15	0	.521	0	.585	0	.586	0	.598	0	.605	0	.622
16	0	.592	0	.652	0	.651	0	.664	0	.670	0	.687
17	0	.561	0	.615	0	.614	0	.626	0	.632	0	.648
18	0	.556	0	.614	0	.614	0	.628	0	.635	0	.653
19	0	.574	0	.644	0	.644	0	.657	0	.664	0	.681
20	0	.642	0	.700	0	.699	0	.710	0	.716	0	.732
21	0	.589	0	.640	0	.641	0	.653	0	.660	0	.678
22	0	.535	0	.583	0	.587	0	.603	0	.612	0	.634
23	0	.615	0	.671	0	.673	0	.685	0	.692	0	.711
24	0	.620	0	.687	0	.688	0	.700	0	.707	0	.724
25	0	.536	0	.599	0	.601	0	.614	0	.621	0	.639
26	0	.627	0	.686	0	.686	0	.696	0	.702	0	.718
27	0	.647	0	.701	0	.699	0	.710	0	.716	0	.733
28	0	.575	0	.628	0	.627	0	.639	0	.647	0	.665
29	0	.511	0	.563	0	.567	0	.582	0	.592	0	.614
30	0	.605	0	.669	0	.669	0	.681	0	.688	0	.706
31	0	.621	0	.688	0	.688	0	.701	0	.707	0	.725
32	0	.538	0	.593	0	.595	0	.611	0	.620	0	.640
33	0	.533	0	.583	0	.587	0	.604	0	.614	0	.637
34	0	.624	0	.680	0	.679	0	.693	0	.700	0	.719
35	0	.590	0	.650	0	.650	0	.663	0	.671	0	.689
36	0	.545	0	.603	0	.609	0	.623	0	.631	0	.649
37	0	.621	0	.687	0	.686	0	.698	0	.704	0	.720
AVERAGE		.585		.644		.643		.656		.662		.680

*CONTROL ROD INSERTION DEPTH BY FUEL LAYER.

LCO 4.1.3 BASIS LIMITS: FULLY INSERTED OR FULLY WITHDRAWN: 0.90
PARTIALLY INSERTED: 1.23

TABLE 3-7

FLUX RATIOS AT LOWER POWER
CYCLE 4 @ 300 EFPD

POWER, %	R
0	1.36
2	1.12
8	1.03
18	1.01

R = Average of flux (Regs. 1 through 19)/average flux (core)

TABLE 3-8

WORTH OF CONTROL ROD GROUPS AND MAXIMUM ROD AT NOMINAL EOC4

Groups In	Group Worth, Δk	Cumulative Worth, Δk	Max Rod Worth, Δk	RWA Region
RR(1)	0.002	0.002	0.002	1
+3D	0.016	0.018	0.008	15
+3B	0.023	0.041	0.014	17
+3A	0.012	0.053	0.014	17
+3C	0.022	0.075	0.016	18
+4C	0.014	0.089	0.014	15
+4A	0.007	0.096	0.017	11
+4E(2)	0.007	0.103	0.018	13
(3)	0.112	0.215	N/A	N/A

(1)Regulating rod 115" withdrawn.

(2)Source power criticality at 300 EFPD.

(3)Groups 2B, 4B, 4D, 4F, 2A and the regulating rod fully inserted to assure subcriticality.

TABLE 3-9

CONTROL ROD SHUTDOWN MARGINS (ΔK) IN THE EXTENDED CYCLE 4

Shutdown Time, Days	Core Avg Temp, °F	CR out	RSS in	EFPD 390	EFPD 520
0	220	0	0	0.175	0.200
0	220	22	0	0.116	0.143
0	220	21 + 22	0	0.086	0.114
0	80	21 + 22	0	0.080	0.108
3	80	21 + 22	0	0.050	0.079
14	80	21 + 22	0	0.044	0.073
28	80	21 + 22	0	0.039	0.067
56	80	21 + 22	0	0.032	0.060
224	80	21 + 22	0	0.025	0.052

TABLE 3-10

RSS SHUTDOWN MARGINS (ΔK) IN THE EXTENDED CYCLE 4

Shutdown Time, Days	Inoperable RSS Hopper	EFPD 390	EFPD 520
0	0	0.102	0.130
0	22	0.080	0.107
3	22	0.050	0.078
14	22	0.043	0.071
28	22	0.037	0.065
56	22	0.030	0.057
224	22	0.023	0.049

TABLE 3-11

KINETICS PARAMETERS

	Initial Core		Extended Cycle 4		Equilibrium Cycle	
	BOC, with Xe	EOC	300 EFPD	520 EFPD	BOC, with Xe	EOC
Fractional productions						
From U-233	0.0	0.19	0.48	0.58	0.38	0.48
From U-235	1.0	0.81	0.52	0.42	0.62	0.52
Prompt neutron lifetime, s						
Hot	$2.69x10^{-4}$	$3.17x10^{-4}$	$3.41x10^{-4}$	$3.41x10^{-4}$	$2.85x10^{-4}$	$3.41x10^{-4}$
Cold	$2.43x10^{-4}$	$2.81x10^{-4}$	$3.09x10^{-4}$	$3.09x10^{-4}$	$2.64x10^{-4}$	$3.09x10^{-4}$
Effective delayed neutron fraction	0.00650	0.00577	0.00465	0.00428	0.00505	0.00451
Delayed neutron decay constant, λ, s^{-1}						
Precursor 1	0.01240	0.01250	0.01250	0.01250	0.01250	0.01251
2	0.03050	0.03090	0.03160	0.03190	0.03126	0.03164
3	0.11140	0.11360	0.12000	0.12300	0.00700	0.11990
4	0.30130	0.30250	0.30600	0.30800	0.30470	0.30680
5	1.13600	1.13600	1.13500	1.13500	1.13500	1.13500
6	3.01300	2.98100	2.89200	2.86000	2.91300	2.85900
Delayed neutron fraction, β						
Precursor 1	0.000214	0.00020	0.000222	0.000223	0.000220	0.000222
2	0.001424	0.00100	0.001121	0.001061	0.001186	0.001099
3	0.001274	0.00100	0.000983	0.000925	0.001046	0.000961
4	0.002568	0.00200	0.001685	0.001509	0.001874	0.001619
5	0.000748	0.00050	0.000452	0.000393	0.000516	0.000430
6	0.000273	0.00020	0.000185	0.000167	0.000204	0.000179

TABLE 3-12

WORST DETECTOR DECALIBRATION FACTORS

Description	Reference Cycle 4	Extended Cycle 4
Insert 4E + RWA CR15	0.62	0.65
Insert 4A + RWA CR18	0.77	0.68
Insert 4C + RWA CR10	0.92	0.94
Pull 3C + RWA CR9	0.86	0.91
Insert 3A + RWA CR9	0.95	0.90
Pull 3B + RWA CR15	0.83	0.79
Pull 3D + RWA CR1	0.90	0.91

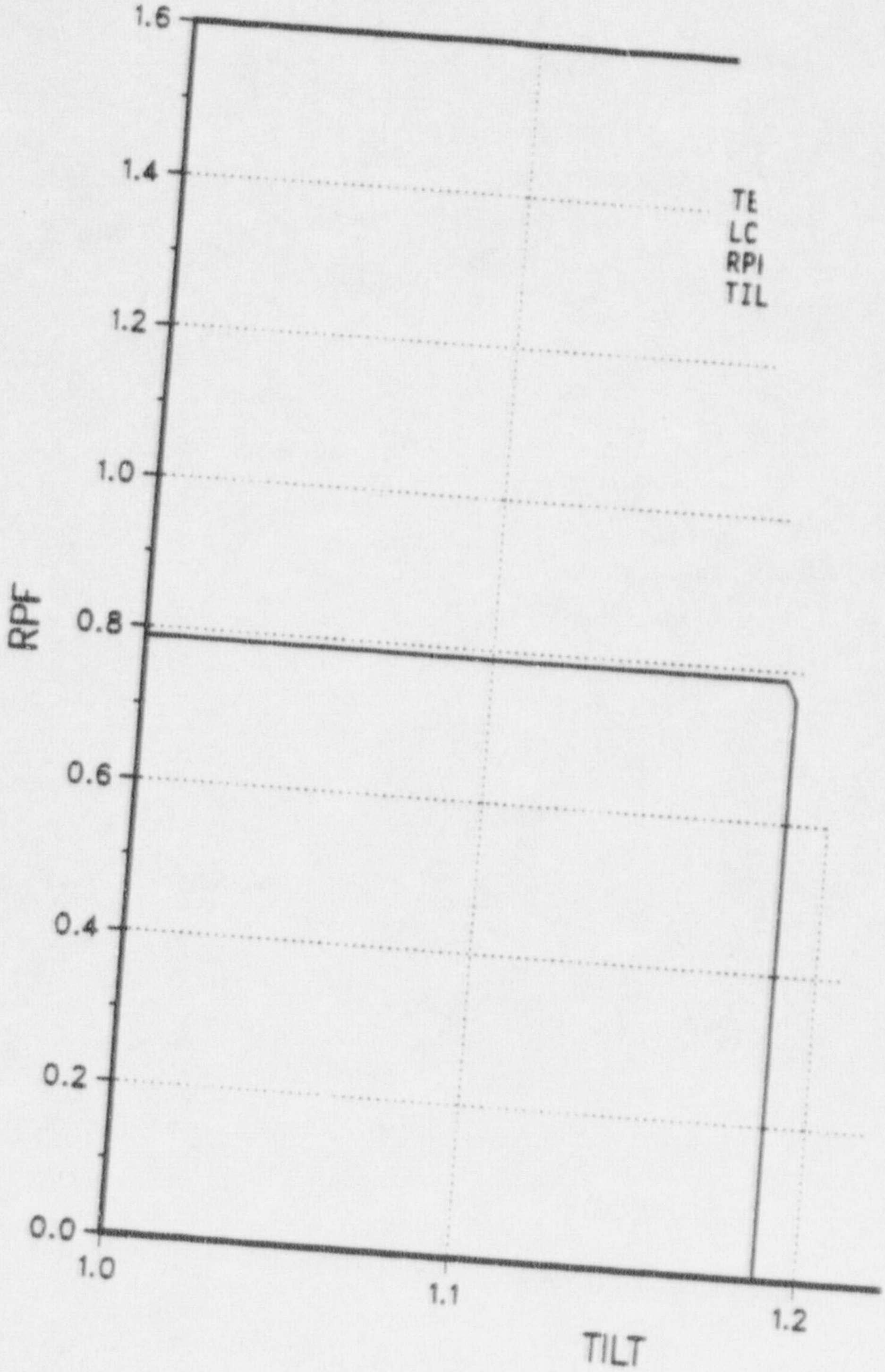

Figure 3-2 Tilt envelope for extended Cycle 4: (b) partially roc

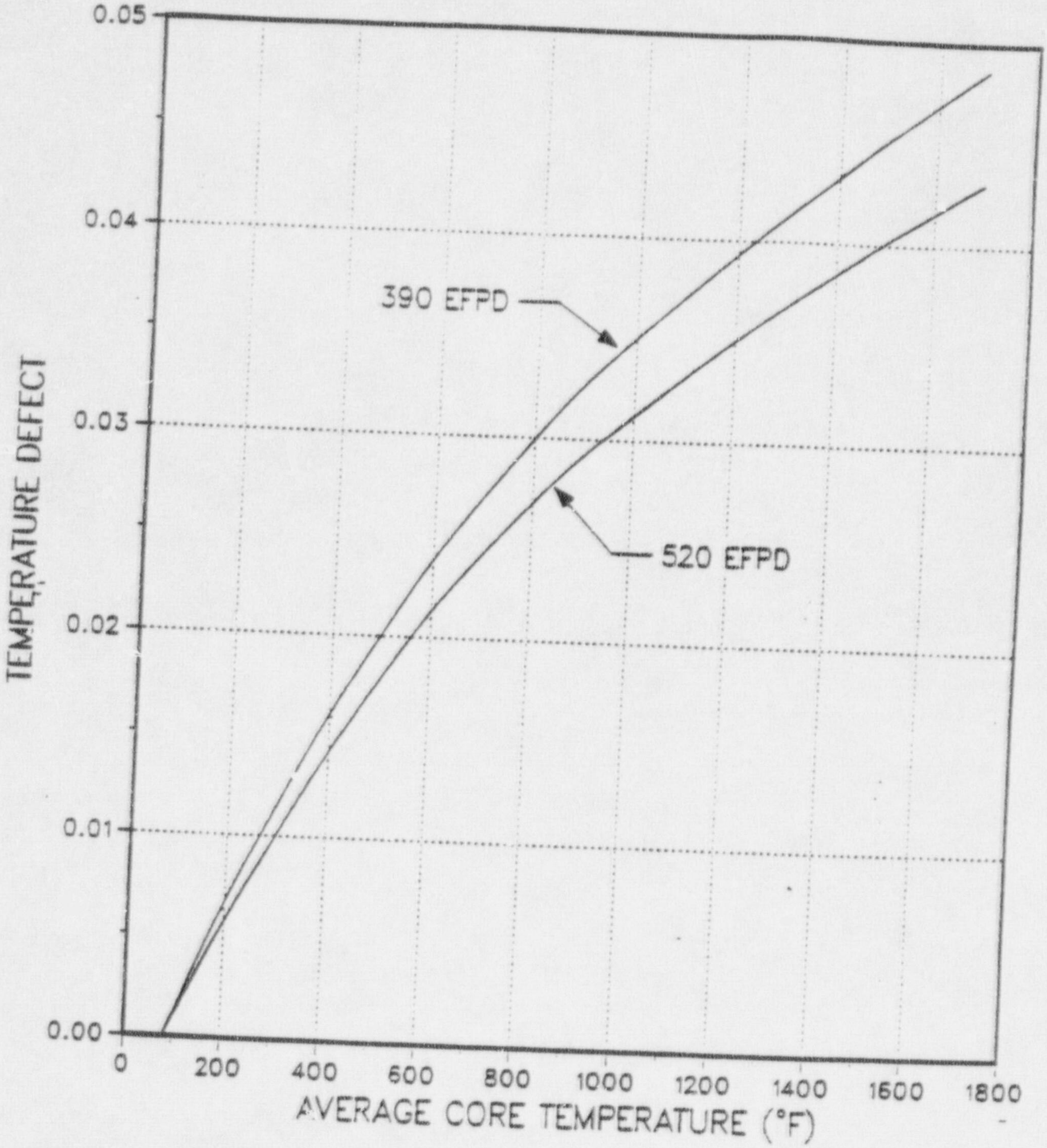

Figure 3-3 Temperature Defect Vs. Average Core Temperature

4. THERMAL-HYDRAULIC AND MECHANICAL PERFORMANCE

4.1 THERMAL PERFORMANCE

The nuclear performance analyses discussed in Section 3 indicate that the power distribution during Cycle 4 extended operation falls within the limits described in the Technical Specifications and the FSAR. No changes are planned for the operation of the core cooling during the extension of Cycle 4 (i.e., helium temperature at the core inlet and average outlet temperature will be enveloped by the FSAR reported values).

Accordingly, the temperature limits presented in the FSAR will not be exceeded during extended operation of Cycle 4. This conclusion is supported by analyses using the COPE code (Ref. 13), which is discussed in the FSAR. The results of these analyses are shown in Table 4-1.

4.2 HYDRAULIC PERFORMANCE

As noted in Section 4.1, the thermal performance of the core during Cycle 4 extended operation is essentially the same as that of previous cycles. No changes will be made in fuel element geometry. The power distributions expected during the extension of Cycle 4 are within the envelopes defined by the basis of Technical Specification LCO 4.1.3. Hence, except for the opening of cross-flow gaps, as discussed and accounted for in Section 3.6.2.2 of the original and updated FSARs, core coolant flow characteristics are also unaffected. Accordingly, there are no changes in the hydraulic performance of the core from that of the initial core or the equilibrium core.

Maximum core pressure drop during extension of Cycle 4 is not expected to exceed about 5.7 psid, which is less than the design equilibrium core value of 8.4 psid.

4.3 FISSION PRODUCT RELEASE

During extension of Cycle 4, the FSV core is expected to be operated within the limits presented in the FSAR and contained in the Technical Specifications. Accordingly, the coated fuel particle failure and the fission product release characteristics of the fuel are expected to be within design limits, and the design radionuclide inventories presented in Section 3.7 of the FSAR will not be exceeded. These conclusions are consistent with operating experience gained during Cycle 1-3 and during the first 154.7 EFPD of Cycle 4.

4.4 MECHANICAL PERFORMANCE

Table 4-1 provides a summary of the fuel element stress, strain and bowing analyses described in this section. These analyses were performed using the methods discussed in the FSAR. Operating and shutdown strain and stress distributions were calculated for the axial and radial orientations throughout the period of extended operation. During core operation the fuel elements will be exposed to fast neutron irradiation, which will induce dimensional changes in the graphite. An analysis was performed to calculate the expected dimensional changes of the fuel elements as a result of extended operation of Cycle 4.

As shown in Table 4-1, all fuel element mechanical performance parameters will be less than the maximum values given in the FSAR for the initial core fuel elements except for the fuel element bowing. The maximum calculated fuel element bow is 0.129 in. The FSAR maximum value, given for the initial core, is 0.090 in. A review has been performed to assess the consequences of this bowing. The possible safety consequences were evaluated, and the potential effect on the operation of the fuel handling machine (FHM) was also considered. It was determined that the safety consequences are within the bounds previously considered in the FSAR and that the fuel element with the maximum bow can, even in the worst case, be readily handled by the FHM.

The evaluation considered the following: control rod insertion with a misaligned core, seismic events, reactivity effects due to gaps, coolant channel misalignment, effect on fuel temperatures, effect on fuel element stresses, and flow induced vibration. This evaluation assumed a bow of 0.150 in., which is larger than the maximum calculated fuel element bow. The fuel handling evaluation determined that the maximum allowable bow for an element using conservative assumptions for the effect of the bow on the fuel handling geometry is 0.136 in., which is larger than the maximum calculated bow.

Control rod insertion will not be affected by the bow. The maximum bow occurs in the top layer of the active core. The maximum radial displacement of the control rod channel the top of the core is limited by the region constraint devices. Even with the angular misalignment introduced by the bow, the total misalignment of the control rod channels is much less than that of the core misalignment tests reported in Section 3.8 of the FSAR.

The effect of the fuel element bow on fuel temperatures can be due to two conditions: crossflow and coolant channel alignment. The dowels will maintain block alignment during all normal operating conditions. The bow will not cause any relative block motion which could result in partial coolant channel blockage. The larger bow will result in increased crossflow in the region. However, the bow will be larger in older, lower power regions. These regions have the orifice valves relatively closed so the direction of crossflow will be into the regions, resulting in lower fuel temperatures.

The seismic event discussed in Section 14.1.1.2 of the FSAR, which results in a single dowel engagement, will not be affected by the increased maximum fuel element bow. Single dowel engagement is possible only for six standard fuel columns and only for the gap between the second and third layers of the active core in those columns. The calculations performed determined that bows larger than the FSAR value of 0.090 in. will not occur for those locations.

There will be no effect on core reactivity due to the fuel element bowing. While the bow will cause some gaps to increase in size, other will decrease correspondingly. The net change in gap size will be due only to radial shrinkage of the fuel elements with irradiation in a manner consistent with the original design assumptions.

There will be no fuel element stress increase resulting directly from the bow. The gaps surrounding each column, coupled with the radial shrinkage of the fuel elements with irradiation, are large enough to preclude interference. The increased crossflow could possibly result in higher thermal stresses but this will be offset by the fact that stresses decrease rapidly after the first two or three years of operation. Accordingly, the peak stress conditions will occur prior to reaching the maximum bow condition.

The fuel columns will be more stable with respect to flow induced vibration or deflection since the orientation of bow defines a preferred direction for the column to lean.

The FHM grapple can engage elements with a top face about 2° off from the horizontal. This corresponds to a difference in elevation of 0.500 in. across the flats of the element. This difference would result from a cross-block differential in axial strain of 0.0160 in/in (neglecting block average axial shrinkage which would allow a slightly higher differential strain). The calculated bow due to this axial strain differential is 0.136 in. This result conservatively assumes that all of the gap due to the bow is at the top of the element.

The maximum bow calculated at end of Cycle 4 extended operation is 0.129 in. This occurs for a fuel element in the top layer of a Segment 5 region (Region 23). An additional 45 elements from Segments 4, 5, and 6 were identified as having projected cross-block axial strain differentials large enough to result in a bow greater than 0.090 in. The top layer of Segment 3 was also analyzed, and one element was identified with calculated end of life bow greater than 0.090 in. It should be noted that no problems were encountered removing this element.

Based upon this evaluation, it is concluded that the maximum bow calculated for extended operation of Cycle 4 is acceptable and presents no fuel handling problems or safety consequences beyond those previously evaluated in the FSAR.

TABLE 4-1

EXTENDED CYCLE 4 CALCULATED PEAK CONDITIONS VERSUS FSAR INITIAL CORE PEAK VALUES

Parameter	FSAR Peak Value	Extended Cycle 4 Peak Value(a)
Axial stress (psi)	450	301
Radial stress (psi)	200	54.3
Axial strain (%) (contraction)	3.0	2.0
Radial strain (%) (contraction)	0.8	0.8
Fuel element bowing (in.)	0.09	0.129
Fuel temperature (°F)	2300	2109(b)

(a)Values calculated using FSAR methods.

(b)Peak fuel temperature in core during Extended Cycle 4.

5. SAFETY ANALYSIS

5.1 INTRODUCTION

In this section, the Safety Analysis presented in Chapter XIV of the Fort St. Vrain FSAR is reviewed to determine potential effects of extension of Cycle 4 on accidents and events discussed in the FSAR. The purpose of such a review is to assure that the worst case conditions previously defined for accident analyses, and found to be acceptable during the FSAR review, are not exceeded during extension of Cycle 4, and that no unreviewed safety questions are presented.

As a first step in this review process, Chapter XIV of the FSAR has been examined to identify analyses potentially affected by the extension of Cycle 4. The results of this review are presented in Table 5-1. Seven accident conditions (some of which envelope other less severe events) have been identified as requiring more detailed review for potential effects. These are:

1. Earthquake.

2. Rod withdrawal accidents (RWAs).

3. Column deflection and misalignment.

4. Fuel element malfunctions.

5. Loss of normal shutdown cooling (limiting case: cooldown on one firewater-driven circulator).

6. Permanent loss of forced circulation [Design Basis Accident No. 1 (DBA-1)].

7. Rapid depressurization/blowdown (DBA-2).

As indicated in Table 5-1, RWAs are discussed in Section 3.4 of this document, and earthquake, column deflection and misalignment, and fuel element malfunctions are discussed in Section 4.4 of this document. It is concluded in Section 3.4 that the neutronic parameters of a RWA will be unchanged, and that RWA consequences are no more severe than those of the postulated RWA described in the FSAR. The likelihood of a RWA will decrease significantly during the extension of Cycle 4, because most of the operation will take place with only the regulating rod pair inserted in the core. Thus, accidental withdrawal is precluded of any of the shim control rod pairs that will already have been withdrawn.

For fuel element malfunction, all stresses and axial and radial dimensional changes are calculated in Section 4.4 to be less than or equal to those predicted in the FSAR for the initial core fuel elements. The maximum fuel element bowing, induced by fast neutron irradiation, is calculated in Section 4.4 to exceed that predicted in the FSAR. The possible safety consequences of this bowing were evaluated in Section 4.4. The safety evaluation considered control rod insertion with a misaligned core, reactivity effects due to gaps, seismic events, coolant channel misalignment, effect on fuel temperatures, effects on fuel element stresses, and flow induced vibration. It was determined that the safety consequences are within the bounds previously considered in the FSAR. The worst fuel element deflections, which could occur during an earthquake, are concluded in Section 4.4 to have consequences no worse than those described in the FSAR. The probabilities of occurrence of fuel element malfunction and column misalignment have not increased. The remaining three areas of safety analysis are discussed below.

5.2 LOSS OF NORMAL SHUTDOWN COOLING, PERMANENT LOSS OF FORCED CIRCULATION, AND RAPID DEPRESSURIZATION/BLOWDOWN

The core thermal conditions resulting from firewater cooldown (Safe Shutdown Cooling in the event of loss of normal shutdown cooling), permanent loss of forced circulation, and rapid depressurization/ blowdown are known from past studies to be sensitive to specific core parameters which could be affected by extended operation of Cycle 4. These parameters are radial region power peaking factor (RPF) and core outlet region temperature dispersion (mismatch), which, in turn, are limited by the FSV Technical Specification LCOs 4.1.3 and 4.1.7, respectively. FSAR analyses of these three events include RPF and temperature dispersion values up to the LCO-allowable values. The maximum RPF expected during extension of Cycle 4 varies with power, but remains within the maximum values contained in the basis of LCO 4.1.3, as discussed in Section 3. The maximum temperature dispersion will be controlled not to exceed the LCO 4.1.7 allowable value by using the variable-orifice flow-control assembly located at the inlet to each refueling region. Hence, extension of Cycle 4 does not result in core thermal conditions more severe than those already analyzed. As discussed in Section 4.3, no additional fuel failure occurs beyond end-of-equilibrium cycle conditions described in the FSAR. Hence, extension of Cycle 4 does not result in core radiological conditions more severe than those already analyzed in the FSAR. Operation during extension of Cycle 4 is therefore bounded by the FSAR accident analyses.

5.3 CONCLUSIONS

A review of Chapter XIV of the FSAR identified seven postulated accident conditions that required more detailed examination for potential impact from extension of Cycle 4. No requirements for additional analysis have been identified; the FSAR analysis is found to remain valid in all cases. It is concluded that the worst-case conditions previously defined for accident analyses, and found to be acceptable during the FSAR review, are not exceeded during extension of Cycle 4, and that the extension of Cycle 4 presents no unreviewed safety questions, as defined in 10CFR50.59.

TABLE 5-1

POTENTIAL EFFECTS OF EXTENSION OF CYCLE 4 ON FSV FSAR ACCIDENT PREDICTIONS

FSAR Chapter XIV Event	Potential Effects on Event Analysis Due to Extension of Cycle 4
14.1 Environmental Disturbances	
- Earthquake	Evaluation required, see Section 4.4 of this document
- Wind effects - Flood - Fire - Landslides - Snow and ice	None - The core is not affected by these events
14.2 Reactivity Accidents and Transient Response	
- Summary of reactivity sources	
Excessive removal of control poison Loss of fission product poisons Rearrangement of core components Introduction of steam into the core Sudden decrease in reactor temperature	Reactivity insertions in these events are bounded by rod withdrawal events
- Rod withdrawal accidents	Evaluation required, see Sec. 3.4 of this document
14.3 Incidents	
- Incidents Involving the Reactor Core	
Column deflection & misalignment	Evaluation required, see Sec. 4.4 of this document
Fuel element malfunctions	No change from Sec. 3.5.4.5 of FSAR
Misplaced fuel element	No change from Sec. 3.6.5.2 of FSAR
Blocking of coolant channel	No change from Sec. 3.8 of FSAR
Control rod malfunctions	No change from Sec. 3.6.5.1 of FSAR
Orifice malfunctions	No change from Sec. 3.3.2.2 of FSAR
Core support floor loss of cooling	
- Incidents involving the primary coolant system	No change from Section 4.2.2 of FSAR

Table 5-1 (continued)

FSAR Chapter XIV Event	Potential Effects on Event Analysis Due to Extension of Cycle 4
- Incidents involving the control and instrumentation system	No change from Section 6.4.2 of FSAR
- Incidents involving the PCRV	No change from Sections 5.9.2 and 9.7 of FSAR
- Incidents involving the secondary coolant and power conversion system	No change from Section 10.3 of FSAR
- Incidents involving the electrical system	No change from Section 10.3 of FSAR
- Malfunctions of the helium purification system	None - The core is not affected by these events
- Malfunctions of the helium storage system	
- Malfunctions of the nitrogen system	
- Malfunctions involving handling of heavy loads	
14.4 Loss of Normal Shutdown Cooling	Evaluation required, see Section 5.2 of this document
14.5 Secondary Coolant System Leakage	
- Steam leaks outside the primary coolant system	No change from Section 14.5.1 of FSAR
- Leaks inside the primary coolant system/steam generator leakage (moisture ingress)	None - FSAR analysis encompasses core thermal conditions allowable under Tech. Specs.; extended Cycle 4 will not exceed same.

Table 5-1 (continued)

FSAR Chapter XIV Event	Potential Effects on Event Analysis Due to Extension of Cycle 4
14.6 Auxiliary System Leakage	
- Failures involving the helium purification system	
Loss of both purification trains	Possible effects would be bounded by Design Basis Accident No. 2, FSAR Section 14.11
Failure of regeneration line with simultaneous valve failure and operator error	
- Accidents involving the gas waste system	No change from Section 14.6.2 of FSAR
- Fuel handling and storage accidents	
Fuel handling accidents Fuel storage accidents	None - Analysis in Section 14.6.3 of FSAR is bounding
14.7 Primary Coolant Leakage	Possible effects would be bounded by Design Basis Accident No. 2, FSAR Section 14.11
14.8 Maximum Credible Accident	
14.9 Maximum Hypothetical Accident	Same as FSAR Section 14.11
14.10 Design Basis Accident No. 1, "Permanent Loss of Forced Circulation (LOFC)"	Evaluation required, see Section 5.2 of this document
14.11 Design Basis Accident No. 2, "Rapid Depressurization/ Blowdown"	Evaluation required, see Section 5.2 of this document

6. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

No changes to the plant Technical Specifications are necessitated by the extension of Cycle 4 up to a total of 520 EFPD.

As noted in Section 2, the plant will continue to be operated in accordance with procedures based upon interim Technical Specifications for reactivity control.

7. SURVEILLANCE TESTS

The following reactor core physics surveillance tests are performed at the beginning of each new fuel cycle.

SR 5.1.5-RX, Control Rod Reactivity Worth, is performed to demonstrate that the measured worth for each control rod group withdrawn during power operation compares with the calculated worth within some specified uncertainty. This test provides assurance that the calculated values used in the safety analyses are acceptable and that the reactivity discrepancy as monitored continuously can be accurately determined.

SR 4.1.7-RX, Temperature Coefficient of Reactivity, is performed over the fuel temperature range from 220 through 1500°F. This measurement is done at the beginning of cycle (BOC), or as close to that as feasible, to demonstrate that the temperature defect with a new fuel segment added is within the limits specified in Interim Technical Specification LCO 3.1.7. The reactivity temperature defect must be more negative that the minimum limit (-0.31 Δk) to ensure that the temperature coefficients used in the accident analysis are adequate, and less negative than the maximum limit (-0.65 Δk) to assure that the calculated worth of the Reserve Shutdown System is adequate. Since the temperature coefficients decrease with burnup throughout the cycle, their minimum value is at the end of cycle (EOC). Therefore, the measurements are made to the BOC when the new fuel has been added, and are extrapolated to obtain an EOC minimum value.

A special test, Detector Decalibration, is performed for the control rod configurations (each control rod group fully withdrawn) that exist for the specified control rod sequence from startup to full power. The measured decalibration factors are compared with the calculated values, and the high-power level scram setpoint schedule for decalibration given in Technical Specification LSSS 3.3 is justified using the measured data.

In addition, the following surveillance tests are required at specified frequencies throughout the cycle operation.

SR 4.1.1 and SR 4.1.3, Control Rod Operability and Position Indication, and SR 4.1.8 and SR 4.1.9, Reserve Shutdown System Operability, are conducted to demonstrate that the control rods and reserve shutdown system will perform per the design requirements.

SR 4.1.4 A-W and SR 4.1.4 B-P-X, Reactivity Status Check, are performed weekly during operation and prior to each startup (approach to critical), respectively. These surveillances ensure that the reactivity discrepancy, the difference between the expected and actual control rod configuration, does not exceed 0.01 Δk.

SR 5.1.7 a-X and b-X, Calculated Region Peaking Factors and RPF Discrepancies, are performed monthly or at regular burnup intervals during power operation to ensure that the measured RPF distribution is in agreement with the calculated RPF distribution and that Regions 20 and 32-37 are being orificed appropriately.

As a consequence of these continuous surveillances, both the core reactivity and the power distribution are continuously compared with the previously calculated data. Any trend in discrepancy can be monitored and evaluated, and appropriate action can be taken.

All of the physics tests required to be performed at the BOC for Cycle 4 have been acceptably completed. Since early operation in Cycle 4 was limited to 35% power, this testing was not completed until about 80 EFPD of burnup had been achieved.

All of the measured control rod group worths, group 2B through 3A, met the associated acceptance criteria. The measured reactivity temperature defect met the acceptance criteria that the measured value be less negative than -0.065 Δk and more negative than -0.031 Δk. The measured value at the BOC was -0.0538 Δk, and the value extrapolated to the nominal EOC was -0.0434 Δk. Both of these are well within the required limits. A comparison of the Cycle 4 measured and calculated detector decalibration data indicated agreement consistent with that for previous cycles and supportive of the analytical methods. An evaluation to confirm the adequacy of the high power level scram setpoint schedule using combined measured/calculated decalibration data demonstrated the conservatism of this setpoint schedule.

Since the physics testing has demonstrated the adequacy of the calculated physics data for Cycle 4, and since required surveillance tests monitor the core reactivity and power distribution on a continuous basis, additional physics tests are not planned for extended operation of Cycle 4. However, if during the core monitoring any trend is observed which indicates a discrepancy of the calculated data, additional physics tests will be performed as required.

In addition, before the previously planned end of Cycle 4 core burnup (300 EFPD) is extended, the measured value for the reactivity temperature defect will be extrapolated to the planned end of extended operation. This will ensure that the extended end of cycle value is more negative than that required in Interim Technical Specification LCO 3.1.7, -0.031 Δk. Based upon burnups achieved through April 30, 1989, the extended operation of Cycle 4 is not expected to exceed 460 EFPD by the time nuclear operations are discontinued on or before June 30, 1990.

8. REFERENCES

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