Entergy
Operations

Entergy Operations, Inc.
Route 3 Box 137G
Russellville, AR 72801
Tel 501-964-8888

Neil S. "Buzz" Carns
Vice President
Operations ANO

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U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, DC 20555

SUBJECT: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Technical Specification Change Request
Pressurizer Pressure

Gentlemen:

Attached for your review and approval are proposed Technical Specification (TS) changes which revise 2.1.1 Bases, Section 3.2.8 and Bases, and Tables 2.2-1 and 3.3-4. This change increases the allowable pressurizer pressure range. A lower low pressurizer pressure setpoint for reactor trip, safety injection, and containment cooling is also being proposed by this change.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that these changes involve no significant hazards considerations. The bases for these determinations are included in the enclosed submittal.

Although the circumstances of this proposed change are not considered emergency or exigent, your prompt review and approval is requested. The purpose of this proposed Technical Specification change is to improve the reliability of the pressurizer safety valves by reducing valve simmering and subsequent leakage.

We request that the effective date of this change be upon issuance of the amendment.

Very truly yours,

Neil S. Carns

NSC/sjf
Attachment

A001
1/1

cc: Mr. James L. Milhoan
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

NRC Senior Resident Inspector
Arkansas Nuclear One - ANO-1 & 2
Number 1, Nuclear Plant Road
Russellville, AR 72801

Mr. Thomas W. Alexion
NRR Project Manager, Region IV/ANO-1
U. S. Nuclear Regulatory Commission
NRR Mail Stop 11-D-23
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

Ms. Sheri R. Peterson
NRR Project Manager, Region IV/ANO-2
U. S. Nuclear Regulatory Commission
NRR Mail Stop 11-D-23
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

STATE OF ARKANSAS)
) SS
COUNTY OF LOGAN)

Affidavit

I, N. S. Carns, being duly sworn, subscribe to and say that I am Vice President, Operations ANO for Entergy Operations, that I have full authority to execute this affidavit; that I have read the document numbered 2CAN079202 and know the contents thereof; and that to the best of my knowledge, information and belief the statements in it are true.

N S Carns
N. S. Carns

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for the County and State above named, this 22nd day of July, 1992.

Sandy Siebenmorgen
Notary Public

My Commission Expires:

May 11, 2000

ATTACHMENT

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT TWO

DOCKET NO. 50-368

PROPOSED CHANGES

The proposed amendment would change Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specification (TS) 3.2.8 and associated bases to allow plant operation with pressurizer pressure between 2025 and 2275 psia.

A clarification of TS 2.1.1 Bases with regards to the application of the peak linear heat rate (PLHR) limit to anticipated operational occurrences analysis results is also being proposed.

Additionally, this proposed amendment would lower the ANO-2 TS Table 2.2-1 and associated bases reactor protection low pressurizer pressure trip setpoint and allowable values to 1717.4 and 1686.3 psia respectively. The safety injection and containment cooling actuation trip setpoint and allowable values given in ANO-2 Technical Specification Table 3.3-4 would also be lowered to 1717.4 and 1686.3 psia by this proposed amendment.

BACKGROUND

Small amounts of pressurizer safety valve leakage have historically occurred on ANO-2 but not until recently has this problem alone resulted in plant shutdowns for valve replacement or repair. In the past, small amounts of safety valve leakage below the allowable TS limit for Reactor Coolant System (RCS) leakage were not considered significant. Also, continuous plant operation runs were sufficiently shorter. Plant shutdowns and forced outages due to other reasons allowed the safety valves to be replaced or repaired if necessary.

Recently, leakage out of the pressurizer has been monitored more closely. Modifications have been made to the safety valves (Flexi-Disk), and to the valves' discharge piping (nozzle load reduction) in an attempt to minimize or eliminate valve simmering/leakage. However, these changes have not eliminated the problem. Initially, valve simmering is characterized by low volume, high velocity, saturated steam leakage across the valve seats. This is condensed immediately without causing a temperature increase at the discharge piping temperature sensors (located approximately 6 feet from the valve), nor a measurable volume increase in the quench tank. Prolonged simmering tends to allow an increased steam volume to the point of detection. Continued exposure at the higher volume and velocity typically will cause seat damage. Once seat damage occurs, the valve cannot be reseated into a leak tight condition. After seat damage, the leakage rate tends to increase with time, so the longer the leakage duration, the higher the leakage rate. As the performance of ANO-2 improves, as indicated by longer continuous plant operating times, the total safety valve leakage increases in an exponential manner eventually causing plant shutdown. In October, 1991, ANO-2 was shut down to replace the safety valves due to leakage concerns. This type of forced outage has both economic impacts and associated safety concerns.

The economic impacts of the high maintenance costs and as low as reasonably achievable (ALARA) considerations to repair/replace the valves, in addition to the power loss associated with a forced outage are significant. There are also costs associated with the treatment of the radwaste generated by the leaking valves in addition to the requirement for makeup water. Valve leakage within the range currently administratively allowed due to pressurizer heater capacity (1 gpm versus 10 gpm allowed by Technical Specifications) does not affect the capability of the valve to perform its safety function. Additionally, there is an exposure to risks associated with perturbating the RCS for plant shutdown and cooldown for valve replacement.

Reducing operating pressurizer pressure, for a short time period when the valve starts to simmer, is expected to substantially curtail or eliminate the safety valve simmering problem and subsequent valve leakage. By operating the RCS at reduced pressures, the safety valves are given a chance to reach a thermal equilibrium point at a pressure with sufficient margin to the valve lift setpoint (2500 +1,-3% psia) to avoid simmering. Presently, plant operation with RCS pressure within the bounds specified by Technical Specification 3.2.8, maintains only an approximate 10% margin to the safety valve lift setpoint. Small perturbations in the valve thermal equilibrium point can initiate valve simmering. It is postulated that when the perturbations occur, if the RCS pressure is reduced for a sufficient time period to allow the valve to reestablish an equilibrium point, simmering and valve leakage can be terminated. To reestablish the equilibrium point involves a significant pressure reduction to approximately 2025 psia for short durations. To ensure the valve(s) remain leaktight, it may be appropriate to maintain continuous operation at a smaller pressure reduction (approximately 2150 psia). By avoiding the simmering, valve seat damage is also precluded which enhances the valve reliability and lengthens the life of the valve.

DISCUSSION

The proposed change revises Technical Specification 3.2.8, Pressurizer Pressure Limiting Condition for Operation (LCO) to allow plant operation in Mode 1 with pressurizer pressure between 2025 and 2275 psia. These lower pressure limits are consistent with other CE plants. Additionally, this proposed change would reduce the low pressurizer pressure Reactor Protection System (RPS), Safety Injection Actuation System (SIAS), and Containment Cooling Actuation System (CCAS) trip setpoint and allowable values to 1717.4 and 1686.3 psia, respectively.

The following discussion is divided into three parts. First, the pressurizer pressure reduction justification is provided based on the Safety Analysis Report (SAR) Chapter 15 evaluations and plant safety system Core Protection Calculator (CPC) and Core Operating Limit Supervisory System (COLSS) range verification. Next, the clarification to the PLHR Technical Specification Basis is discussed. Finally, the proposed low pressurizer pressure RPS, SIAS, and CCAS trip setpoint and allowable value changes are presented.

1) Pressurizer Pressure Reduction

Safety analyses supporting Chapter 15 of the ANO-2 SAR presently bound plant operation with actual pressurizer pressure (i.e., including pressure instrumentation measurement uncertainty) between 2200 and 2300 psia. These analyses were reviewed to identify which would be adversely affected by plant operation at a lower pressurizer pressure. Table 1 presents those events affected by lower pressurizer pressure.

The affected analyses identified in Table 1 have been reperformed assuming an initial pressurizer pressure of 2000 psia. The plant response to these events was simulated with the NRC approved CESEC-III computer code. Departure from nucleate boiling ratio (DNBR) analyses were performed based on the TORC computer code, the CE-1 critical heat flux correlation, and the CETOP code.

Calculation factors were combined statistically with other uncertainty factors at the 95/95 confidence/probability level for Cycle 2 to define a specified acceptable fuel design limit (SAFDL) on the CE-1 minimum DNBR. The DNBR limit was again revised in Cycle 5 to directly incorporate NRC penalties and credit a reduced rod bow penalty. The Cycle 5 DNBR limit of 1.25 remains applicable for current analyses. Results were verified to be within acceptance criteria, including the SAFDLs. No fuel cladding damage is predicted for any event, therefore, no changes to the radiological doses were calculated.

Over the past 9 cycles, changes have been made in some of the conservative assumptions utilized in the accident analysis that is presented in Tables 2, 5, 7, 10, and 12. These changes are a result of changing plant conditions, cycle specific parameters, conservative enhancements, crediting more limiting TS, and plant modifications. This proposed TS change only relates to the reduced RCS pressure; therefore, unrelated assumptions will not be discussed in this submittal. ASEA Brown Boveri - Combustion Engineering Nuclear Power maintains all the recorded calculations for the reanalyzed events. These calculations document the conservative input assumption bases and are available for review.

The Loss of External Load/Turbine Trip bounding analysis was performed using an initial RCS pressure of 2000 psia. A lower initial pressurizer pressure delays the reactor trip on high pressurizer pressure, allowing the pressurization rate to increase prior to trip, and thereby forcing a greater pressure increase following the trip. This event was reanalyzed with the conservative assumptions listed in Table 2. The results are presented in Table 3 and compared to Cycle 2 results in Table 4. A peak RCS pressure of 2744 psia and steam generator pressure of 1160 psia were calculated. These values are within 110 percent of the design limit pressures, 2750 psia and 1210 psia for the RCS and steam generator, respectively.

An uncontrolled CEA bank withdrawal event from both subcritical and 1% power was evaluated. The 100% power case is not adversely affected by the lower RCS pressure due to credit taken for the CPC low DNBR trip, which as indicated below is valid over the proposed pressurizer pressure range. The subcritical CEA bank withdrawal evaluation, which is terminated by the High Logarithmic Power Level Trip, produced acceptable results. Table 5 lists the conservative assumptions used for the Cycle 10 analysis with results presented in Table 6. The analysis determined that at the time of minimum DNBR, a three dimensional power peaking factor (Fq) of 5.78 corresponds to a DNBR of 1.25. The analysis also determined that for Fq < 10, the fuel centerline temperature is less than 3450°F. Since the maximum Fq calculated for this event is 4.5, the specified acceptable fuel design limits (DNBR ≥ 1.25 and fuel centerline temperature below 4900°F) are met.

CEA withdrawal from 1% power event was also performed at an initial RCS pressure of 2000 psia. The High Pressurizer Pressure Trip was originally credited in the Final Safety Analysis Report (FSAR analysis) to terminate this event. Since the original analysis, a Variable Over Power Trip (VOPT) has been added to the CPCs as part of the CPC Improvement Program implemented during Cycle 5. A list of the conservative assumptions used in this analysis are given in Table 7. The sequence of events is presented in Table 8 and the results in Table 9. As indicated, the VOPT is the first trip encountered and terminates the reactor power excursion at a lower level than previously calculated. As a result, the minimum DNBR remains above 1.8, and the maximum linear heat rate remains below 17 kw/ft, each within the acceptance criteria of 1.25 and 21 kw/ft respectively. Crediting the VOPT more than offsets the small adverse effects of the lower RCS pressure; hence, acceptable results were obtained when the CEA bank withdrawal from 1% power was evaluated.

The VOPT trip condition for a CEA withdrawal event for 1% power will occur prior to 40% power based on the current CPC addressable constants. These constants are determined for each cycle consistent with the software and methodology established in the CPC Improvement Program (CEN-304-P, CEN-308-P, CEN-305-P, and CEN-310-P) and the current cycle design, performance, and safety analyses.

The CEA ejection events from Hot Full Power (HFP) and Hot Zero Power (HZP) were both reevaluated utilizing a new lower limit of 2000 psia. The STRIKIN-II computer program was used in this evaluation to simulate the heat conduction within a reactor fuel rod and its associated surface heat transfer. Conservative assumptions used in the CEA ejection analysis are given in Table 10 and results are presented in Table 11. The maximum centerline enthalpy decreased for both cases; however, a slight increase in the number of fuel pins having incipient centerline melting for the HFP was noted (a total of 0.32%); but no fuel pins were calculated as having clad damage or fully molten centerline. Hence, the results from this evaluation were considered acceptable.

A single part length CEA (PLCEA) drop incident was reevaluated to determine the effects of reduced RCS pressure. Only positive reactivity insertions resulting from a PLCEA drop are of concern. With the PLCEA insertion limits imposed by Section 3.1.3.7 of the ANO-2 Technical Specifications, positive reactivity insertions can only be postulated for PLCEA drops below 50% power. It should be noted that the FSAR analysis was performed prior to the addition of TS section 3.1.3.7 in which 100% power was originally assumed to be conservative. The positive reactivity increases core power, and consequently, pressurizer pressure. The pressure and power increases are constrained by the action of the High Pressurizer Pressure Trip. A reduction in the initial pressure can delay the high pressurizer pressure trip, thereby allowing a greater power increase, and a correspondingly larger decrease in the fuel thermal margin. However, sufficient initial thermal margin will be preserved by the COLSS, which is verified every cycle in the reload analyses, to assure that the DNBR SAFDL is met throughout the PLCEA drop event. The conservative assumptions for the PLCEA drop, which produce the maximum power increase that avoids the high pressurizer pressure trip are listed in Table 12. The corresponding plant response is presented in Table 13.

The data and algorithms of the CPCs were verified to be valid for a range of pressurizer pressures which cover the proposed operating pressure range. Additionally, the CPCs generate a reactor trip signal if the RCS pressure exceeds 2375 psia or drops below 1860 psia.

COLSS monitors and initiates alarms if the LCOs on DNBR, peak linear heat rate, core power, axial shape index, or core azimuthal tilt are exceeded. Only the DNBR LCO is dependent on pressurizer pressure. An uncertainty factor is applied in the COLSS calculation for DNBR to account for instrument uncertainty on the measured parameters (RCS temperature, pressure, flow, etc.) used as inputs to the COLSS calculations. Additionally, the COLSS calculations are benchmarked against more detailed calculations over the allowed operating ranges. Differences between the calculation results were also included in the uncertainty factors. The uncertainty factors have been reviewed and verified to be conservative over the proposed expanded pressurizer pressure range down to 2000 psia.

2) PLHR Clarification

The uncontrolled CEA bank withdrawal from subcritical conditions analysis results indicated a transient peak linear heat rate in excess of the 21 kw/ft limit given in TS 2.1.1.2 (A PLHR less than 28 kw/ft and exceeding 21 kw/ft for less than one second was calculated). The intent of this TS is to prevent the fuel from melting during normal operation and following anticipated operational occurrences. A limit of 21 kw/ft is specified based on steady state operation fuel centerline melting temperatures; hence, higher linear heat rates can occur under transient conditions without resulting in fuel melting. As indicated above, the fuel centerline melt temperature was acceptable for the subcritical CEA bank withdrawal. A clarification to the Bases of TS 2.1.1.2 is proposed in this amendment to ensure the appropriate application of the peak linear heat rate limit.

CPCs monitor linear heat rate and analysis results are typically given in kw/ft, thereby making the peak linear heat rate limit of 21 kw/ft appropriate in most situations. But, for anticipated operational occurrences with transient peak linear heat rates, the more appropriate limit is the specified acceptable fuel design limit centerline melting temperature, which is the basis for the peak linear heat rate.

This clarification is consistent with other CE plant interpretations of this Technical Specification. Additionally, this is consistent with the interpretation utilized for the CPCs as documented in the methodology and software manuals.

3) Lower Low Pressurizer Pressure Setpoint

The ANO-2 SAR briefly addresses an inadvertent operation of Emergency Core Cooling System (ECCS) during power operation. In this scenario, an SIAS is inadvertently actuated during power operation. Operating with the pressurizer pressure below approximately 2150 psia may result in an undesirable SIAS following a reactor trip from significant power levels. The probability of an inadvertent actuation of an SIAS during power operation is not increased, but the likelihood of receiving an SIAS following a reactor trip is increased during reduced pressure operation. It should be noted that if an undesirable SIAS is received following a reactor trip, the consequences of that event would be bounded by the inadvertent actuation during power operation.

Post-trip pressure response from reduced initial pressurizer pressure is expected to be comparable to higher pressure trips, with the minimum post-trip pressure being correspondingly lower. These lower post-trip responses will come closer to the low pressurizer pressure (i.e. SIAS) setpoint. To minimize the impacts of reduced pressure operation post-trip, previous plant data was reviewed to determine the minimum acceptable pressurizer pressure for avoiding an SIAS post-trip. As a result, a lower low pressurizer pressure trip setpoint is being proposed.

All of the safety analyses identified above as being adversely impacted by the reduced pressurizer pressure were reevaluated down to 2000 psia. CPCs and COLSS were also verified to be applicable down to 2000 psia. However, due to the potential for an undesirable SIAS actuation following a reactor trip when operating with the pressurizer pressure below 2150 psia, a minimum continuous operating pressure of 2150 psia will be administratively controlled. Operation down to the proposed minimum pressurizer pressure limit 2025 psia will also be administratively controlled to short durations, thereby allowing operator flexibility when attempting to reseat simmering pressurizer code safeties; yet minimize the exposure to an undesirable SIAS actuation. Operation above 2150 psia will not increase the probability of having an undesirable SIAS (post-trip). The duration of steady state operation below 2150 psia will be administratively controlled to a value (in this case 24 hours) that will ensure that the probability of inadvertent SIAS actuation will not be significantly increased. The 2025 psia limit is based on the analysis assumption of 2000 psia plus 25 psi which bounds pressure measurement uncertainties.

Reductions in the low pressurizer pressure RPS and Engineered Safety Feature Actuation System (ESFAS) trip setpoint and allowable values are also being proposed. These reductions will help prevent an undesirable SIAS following a reactor trip when operating at reduced pressures. The new low pressurizer pressure setpoints and allowable values are based on new instrument error calculations; the safety analysis setpoint assumptions have not changed. A reduction in the calculated instrument error is due primarily from the use of more realistic environmental assumptions at the time of actuation. A new instrument error calculation has been generated using ANO instrument error procedures. The statistical method of the square root of the sum of squares was used to determine the random error on a component level and for the loop. Non-random errors were combined algebraically with the random error term to establish total error. Although ANO has not committed to strict compliance with the requirements of ISA-S67.04-1988 "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," these guidelines were considered in calculating the loop errors, periodic test errors (PTE), and allowable values associated with the low pressurizer pressure setpoints. This calculation supports the proposed low pressurizer pressure setpoint of 1717.4 psia and allowable value of 1686.3 psia for the RPS, SIAS, and CCAS trip functions.

In the new instrument error calculation, several non-realistic assumptions are removed with regard to the containment conditions; specifically, the conditions at which the low pressurizer pressure instruments reach the trip setpoint. The original instrument error calculation conservatively assumed worst case long-term harsh environment inside containment based on a large break loss of coolant accident (LOCA). In reality these instruments would perform their function prior to the worst case conditions occurring. Based on this, small break LOCAs, large break LOCAs, and steam line breaks (SLBs) were evaluated separately. Seismic error terms were removed in all cases as a concurrent initiating event since this is beyond the ANO design basis. Radiation terms were also removed due to the quick response of the pressurizer pressure following LOCAs. No fuel damage is predicted following a SLB and the primary system remains intact; hence, no radiation terms are assumed for a SLB. The maximum containment temperature assumed for the LOCA instrument error calculations was based on specific analyses and the availability of the redundant high containment pressure trip instruments. Crediting these more realistic containment conditions results in a better defined and more realistic low pressurizer pressure setpoint.

DETERMINATION OF SIGNIFICANT HAZARDS

An analysis of the proposed changes has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards consideration using the standards in 10CFR50.92(c).

A discussion of these standards as they relate to this amendment request follows:

<u>Criterion 1</u> - Does not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

Previously analyzed accidents and anticipated operational occurrences that are affected by this change have been reviewed. This change has no impact on probability of occurrence of these accidents. Pressurizer pressure and low pressurizer pressure setpoints are only used as input parameters to the accident analyses which do not affect probability. No physical changes to the plant are being proposed; therefore there is no impact on the probability of accidents.

One item, "Inadvertent Operation of ECCS During Power Operation" is considered to be potentially impacted since operation at RCS pressures below 2150 psia may result in undesirable ECCS (SIAS) actuation <u>post-trip</u> (not actually at power). This susceptibility is not considered a significant increase in probability since the reduction in pressure does not impact the probability of ECCS initiation signal generation, rather, if a plant trip occurred for any reason and the plant was operating at reduced pressures, the statistical combination of uncertainties shows that the SIAS initiation setpoint may be reached. Since administrative controls will ensure that operation below 2150 psia is limited to very short durations (i.e. less than 24 hours steady state operation) any postulated increase in probability of an inadvertent operation of ECCS is not considered significant.

Previously evaluated accidents and anticipated operational occurrences which were determined to be adversely impacted by the reduced pressurizer pressure have been evaluated with no significant increase in the consequences. As indicated in the discussion, the SAFDLs and acceptance criteria were verified to be maintained. Additionally, no fuel cladding damage is predicted for any event and no changes to the radiological doses were calculated. Therefore, no increases in the consequences of any accident are predicted.

Changing the low pressurizer pressure setpoint and allowable values is based on the refinement of the instrument error calculations. No change to the analyzed events is proposed due to the new setpoint and allowable values. These new limits still ensure the analysis assumptions are valid; hence, there is no increase in the consequences of the accidents previously evaluated.

<u>Criterion 2</u> - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes do not involve any physical modifications (i.e. new systems, new components, etc.) to the plant. The normal operating value at which RCS pressure is held remains within the ranges typical of pressurized water reactors and more specifically, CE designed nuclear steam supply systems. The results of the accident re-analyses suggest no different phenomena or plant behavior than previously considered. The change to the

bases for Technical Specification 2.1.1 are for clarification and result in consistency with other CE plants' approved Technical Specifications and bases. The low pressure setpoint change does not create any new or different system actuations or interactions than evaluated previously. The slightly increased potential for SIAS initiation post-trip does not suggest new or different type accidents since inadvertent SIAS (at power) is already assessed in SAR Chapter 15 and is bounding for any post-trip SIAS. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in a Margin of Safety

Any accident or anticipated operational occurrence which was determined to be adversely impacted by the reduced pressurizer pressure was evaluated to ensure acceptable results are maintained. The refined instrument error calculations supporting the lower low pressurizer pressure setpoint and allowable values were verified to ensure the present accident analysis assumption are still maintained. Based on these evaluations, the proposed changes do not involve any significant reduction in a margin of safety. Rather, an overall increase in the margin of safety is anticipated, as discussed above, by increasing the safety valve reliability and decreasing the exposure to risks of plant shutdown and cooldown for valve replacement or repair.

CONCLUSION

Based on the above safety evaluation, it is concluded that the proposed change does not constitute a significant hazards consideration as defined by 10CFR50.92.

TABLE 1

EVENTS AFFECTED BY
LOWER PRESSURIZER PRESSURE

1. Loss of External Load/Turbine Trip

2. Uncontrolled Control Element Assembly (CEA) Withdrawal

3. Single Part Length CEA Drop

4. CEA Ejection

TABLE 2

CONSERVATIVE ASSUMPTIONS FOR
LOSS OF EXTERNAL LOAD/TURBINE TRIP

Parameter	Cycle 2 Value	Cycle 10 Value
Core Power, Mwt	2900	2910
Core Inlet, Temperature, °F	540	540
RCS Pressure, psia	2200	2000
Steam Generator Pressure, psia	810	796
CEA Worth at Trip, %ΔK/K	-5.4	-5.4
Moderator Temperature Coefficient, 10^{-4} ΔK/K/°F	0.5	0.0
Number U-Tubes Assumed Plugged per Steam Generator	0	841
Doppler Coefficient Multiplier	0.85	0.85
Pressurizer Safety Valves Opening Pressure, psia	2500	2525

TABLE 3

SEQUENCE OF EVENT FOR THE
LOSS OF EXTERNAL LOAD/TURBINE TRIP

Time, sec	Event	Setpoint or Value
0.	Loss of Condenser Vacuum, Turbine Stop Valves Close, and Main Feedwater Valves Close	- - -
7.5	High Pressurizer Pressure Trip Condition Occurs	2422 psia
8.4	Trip Breakers Open, Pressurizer Safety Valves Open	- - - 2525 psia
8.7	Main Steam Safety Valves Open	1093 psia
9.0	CEAs Begin to Drop	- - -
10.6	Maximum RCS Pressure Occurs	<2750 psia
14.8	Pressurizer Safety Valves Close	2424 psia
15.0	Peak Secondary Pressure Occurs	<1210 psia

TABLE 4

LOSS OF EXTERNAL LOAD/TURBINE TRIP
COMPARISON OF RESULTS

Parameter	Cycle 2 Value	Cycle 10 Value
Maximum RCS Pressure, psia	2671	2744
Maximum Steam Generator Pressure, psia	1144	1160

TABLE 5

CONSERVATIVE ASSUMPTIONS FOR
SUBCRITICAL CEA WITHDRAWAL

Parameter	Cycle 2 Value	Cycle 10 Value
Core Power, Mwt	2.9×10^{-7}	1.52×10^{-7}
Core Inlet Temperature, °F	544.6	545
RCS Pressure, psia	2200	2000
RCS Flow, gpm	322,000	321,200
Steam Generator (SG) Pressure, psia	990	1003
CEA Worth at Trip, %ΔK/K	-5.0	-1.44†
Doppler Coefficient Multiplier	0.85	0.85
Moderator Temperature Coefficient, $\times 10^{-4}$ ΔK/K/°F	0.5	0.5
CEA Withdrawal Worth, ΔK/K/second	0.00025	0.00025

† Only the reactivity that was withdrawn was credited on reactor trip.

TABLE 6

SUBCRITICAL CEA WITHDRAWAL
COMPARISON OF RESULTS

Parameter	Cycle 2 Value	Cycle 10 Value
Minimum DNBR	1.28	>1.25*
Peak Fuel Centerline Temperature, °F	3800	<3450**

*Based on a Cycle 10 Fq < 5.78.
**Based on a Fq < 10.

TABLE 7

CONSERVATIVE ASSUMPTIONS FOR
1% POWER CEA WITHDRAWAL

Parameter	Cycle 2 Value	Cycle 10 Value
Core Power, Mwt	29	29
Core Inlet Temperature, °F	544.6	545
RCS Pressure, psia	2200	2000
RCS Flow , gpm	322,000	321,200
Steam Generator Pressure, psia	978	994
CEA Worth at Trip, %ΔK/K	-5.0	-5.5
Moderator Temperature Coefficient, $\times 10^{-4}$ ΔK/K/°F	0.5	0.5
Reactivity Addition, $\times 10^{-4}$ ΔK/K/second	1.5	1.5
Doppler Coefficient Multiplier	0.85	0.85

TABLE 8

SEQUENCE OF EVENT FOR THE
CEA WITHDRAWAL (BANK) FROM 1% POWER EVENT

Time, sec	Event	Setpoint or Value
0.0	CEA Bank Begins to Withdraw from the Core	- - -
20.8	CPC VOPT Condition Occurs	40.0%
21.4	Trip Breakers Open, CEA Bank Withdrawal is Terminated	- - -
21.7	Maximum Peak Core Power,	50.4% of 2815 Mwt
	Maximum PLHR Occurs	≤21 kw/ft
22.0	CEAs Begin to Drop	- - -
22.3	Maximum Core Heat Flux,	34.5% of 2815 Mwt
	Minimum DNBR Occurs	≥1.25
25.1	Maximum RCS Pressure Occurs	<<2750 psia

TABLE 9

CEA WITHDRAWAL FROM 1% POWER
COMPARISON OF RESULTS

Parameter	Cycle 2 Value	Cycle 10 Value
Maximum Core Power % of 2815 Mwt	91	50.4
Maximum Core Heat Flux % of 2815 Mwt	76	34.5
Maximum RCS Pressure, psia	2662	2303
Maximum LHR, kw/ft	*	16.5
Minimum DNBR	*	1.89

*These results from the Cycle 2 reload analyses were not originally transmitted and are not readily available.

TABLE 10

CONSERVATIVE ASSUMPTIONS FOR THE
CEA EJECTION EVENTS

Parameter	Cycle 2 Value	Cycle 10 Value
Core Power, Mwt		
HFP	2900	2910
HZP	1	1
Core Inlet Temperature, °F		
HFP	556.7	556.7
HZP	544.6	545
RCS Pressure, psia	2200	2000
RCS Flow, gpm	322,000	322,000
CEA Worth at Trip, %ΔK/K		
HFP	-5.4	-5.4
HZP	-2.4	-2.4
Doppler Coefficient Multiplier	0.85	0.85
Moderator Temperature Coefficient, $x10^{-4}$ ΔK/K/°F		
HFP	0.5	0.0
HZP	0.5	0.5
Ejected CEA Worth, ΔK/K		
HFP	0.0030	0.0028
HZP	0.0082	0.0082
Ejected Peak, Fq		
HFP	5.145	5.208
HZP	20.58	12.71
Axial Power Peak, Fz		
HFP	1.75	1.75
HZP	2.5	2.5
Delayed Neutron Fraction β, Total		
HFP	0.00482	0.00536
HZP	0.00482	0.00536
High Linear Trip Setpoint, % of 2815 Mwt	124.8	128.1
CEA Ejection Time, sec		
HFP	0.05	0.05
HZP	0.05	0.05

TABLE 11

CEA EJECTION RESULTS

Parameter	Cycle 2 Value	Cycle 10 Value
Total Average Enthalpy of the Hottest Fuel Pin, cal/gm		
HFP	156	157.2
HZP	164	94.5
Total Centerline Enthalpy of the Hottest Fuel Pin, cal/gm		
HFP	267	264.1
HZP	296	141.9
Number of Fuel Pins Having Clad Damage (E_{tot}(avg) ≥ 200 cal/gpm), %		
HFP	0	0
HZP	0	0
Number of Fuel Pins Having Incipient Centerline Melting (E_{tot}(b) ≥ 250 cal/gm), %		
HFP	<0.5	0.32
HZP	<0.5	0
Number of Fuel Pins Having Fully Molten Centerline Melting (E_{tot}(b) ≥ 310 cal/gm), %		
HFP	0	0
HZP	0	0

TABLE 12

CONSERVATIVE ASSUMPTIONS FOR A SINGLE PART LENGTH CEA DROP

Parameter	Cycle 6 Value	Cycle 10 Value
Core Power, Mwt	1450*	1460
Core Inlet Temperature, °F	551	551
RCS Pressure, psia	2200	2000
RCS Flow, gpm	322,000	321,200
Steam Generator Pressure, psia	958	941
CEA Worth at Trip, %ΔK/K	-5.0	Not Applicable
Moderator Temperature Coefficient, $x10^{-4}$ ΔK/K/°F	0.5	0.20
Reactivity Addition, %ΔK/K	0.04	0.04
Number U-tubes Assumed Plugged per Steam Generator	0	841

*The FSAR analysis was based on 100% power; hence is not included for comparison.

TABLE 13

NOMINAL SEQUENCE OF EVENT FOR THE PLCEA DROP FROM 50% POWER EVENT

Time, sec	Event	Setpoint or Value
0.0	Single PLCEA Drops into the Core	- - -
90.4	Main Steam Safety Valves Open	1093 psia
107.8	Maximum RCS Pressure Occurs	≤2422 psia
300.0	Asymptotic Core Power and Heat Flux Reached,	66.5% of 2815 Mwt