DOCKET FILE

Docket No. 50-368 October 5, 1992

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Yelverton:

SUBJECT: ISSUANCE OF AMENDMENT NO. 138 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M84098)

The Commission has issued the enclosed Amendment No. 138 to Facility Operating License No. NPF-6 for the Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 22, 1992, as supplemented by letters dated September 11, and September 14, 1992.

The amendment revises the TSs to increase the allowable pressurizer pressure range from between 2225 and 2275 psia to between 2025 and 2275 psia. The revision also allows a lower low pressurizer pressure setpoint for reactor trip, safety injection, and containment cooling, along with associated Bases changes.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 138 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	NRC/Local PDR	PD4-1 Reading
T. Alexion (2)	M. Virgilio	J. Larkins
P. Noonan	ACRS(10)(MSP315)	OGC(MS15B18)
D. Hagan(MS3206)	G. Hill(4)	Wanda Jones(MS7103)
C. Grimes(MS11E22)	PD4-1 Plant File	OPA(MS2G5)
OC/LFMB(MS4503)	A. B. Beach, RIV	R. Young

OFC	LA:PD4-1	PE:PD4-1	PM:PD4-1	OGC	D:PD4-1
NAME	PNoonan	RYoung	TAlexion	S Hom	JLarkins
DATE	9/21/92	9/21/92	9/22/92	9/28/92	10/5/92

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

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Division of Reactor Projects - III/IV/V
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See next page

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 138
License No. NPF-6

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated July 22, 1992, as supplemented by letters dated September 11 and 14, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 138, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 5, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 138

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
2-5	2-5
B 2-1	B 2-1
B 2-2	B 2-2
B 2-4	B 2-4
3/4 2-14	3/4 2-14
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
B 3/4 2-4	B 3/4 2-4

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
b. Three Reactor Coolant Pumps Operating	*	*
c. Two Reactor Coolant Pumps Operating - Same Loop	*	*
d. Two Reactor Coolant Pumps Operating - Opposite Loops	*	*
3. Logarithmic Power Level - High (1)	≤0.75% of RATED THERMAL POWER	≤ 0.819% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1717.4 psia (2)	≥ 1686.3 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 729.613 psia (3)
8. Steam Generator Level - Low	≥ 23% (4)	≥ 22.111 (4)

* These values left blank pending NRC approval of safety analyses for operation with less than four reactor coolant pumps operating.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High	$\leq$21.0 kw/ft (5)	$\leq$21.0 kw/ft (5)
10 DNBR - Low	$\geq$1.25 (5)	$\geq$1.25 (5)
11. Steam Generator Level - High	$\leq$93.7% (4)	$\leq$94.589% (4)

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\leq 10^{-4}$ of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at $\leq$200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is $\geq$ 500 psia.

(3) Value may be decreased manually during a planned reduction in steam generator pressure provided the margin between the steam generator pressure and this value is maintained at $\leq$ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% of RATED THERMAL POWER.

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.25 for the CE-1 correlation and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

A steady state peak linear heat rate of 21 kw/ft has been established as the Safety Limit to prevent fuel centerline melting during normal operation. Following design basis anticipated operational occurrences, the transient linear heat rate may exceed 21 kw/ft as long as the fuel centerline melt temperature is not exceeded.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kw/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits (i.e., DNBR and centerline fuel melt temperature) are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III of the ASME Code for Nuclear Power Plant Components. (The reactor vessel, steam generators and pressurizer are designed to the 1968 Edition, Summer 1970 Addenda; piping to the 1971 Edition, original issue; and the valves to the 1968 Edition, Winter 1970 Addenda. Section III of this Code permits a maximum transient pressure of 110% (2750) psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.25 and 21.0 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

BASES

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN 305-P, "Functional Design Requirement for a Core Protection Calculator," July 1985; CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator," July 1985; CEN-310-P, "CPC and Methodology Changes for the CPC Improvement Program," October 1985 and CEN-308-P, "CPC/CEAC Software Modifications for the CPC Improvement Program," August 1985.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of $\leq$ 110.712% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of $\leq$ 0.819% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at ≤2370.887 psia which is below the nominal lift setting (2500 psia) of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at ≥1686.3 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at ≤200 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at ≤200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

POWER DISTRIBUTION LIMITS

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1.

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed or placed in the tripped condition for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure -Low	≥1717.4 psia (1)	≥1686.3psia (1)
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 23.3 psia	≤ 23.490 psia
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
4. MAIN STREAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 751 psia (2)	≥ 729.613 psia (2)
5. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons)	Not Applicable	Not applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure - Low	≥ 1717.4 psia (1)	≥1686.3 psia (1)
6. RECIRCULATION (RAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	54,400 ± 2,370 gallons (equivalent to 6.0 ± 0.5% indicated level)	between 51,050 and 58,600 gallons (equivalent to between 5.111% and 6.889% indicated level)
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3120 volts (4)	3120 volts (4)
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	423 ± 2.0 volts with an 8.0 ± 0.5 second time delay	423 ± 4.0 volts with an 8.0 ± 0.8 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level-Low	$\geq$ 23% (3)	$\geq$ 22.111% (3)
c. Steam Generator ΔP-High (SG-A > SG-B)	$\leq$ 90 psi	$\leq$ 99.344 psi
d. Steam Generator ΔP-High (SG-B > SG-A)	$\leq$ 90 psi	$\leq$ 99.344 psi
e. Steam Generator (A&B) Pressure - Low	$\geq$ 751 psia (2)	$\geq$ 729.613 psia (2)

(1) Value may be decreased manually, to a minimum of $\geq$ 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at $\leq$ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is $\geq$ 500 psia.

(2) Value may be decreased manually during a planned reduction in steam generator pressure, provided the margin between the steam generator pressure and this value is maintained at $\leq$ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value, not a trip value. The zero voltage trip will occur in 0.75 $\pm$ 0.075 seconds.

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of any anticipated operational occurrence.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on DNBR includes appropriate uncertainity and penalty factors necessary to provide a 95/95 confidence level that the core power at which a DNBR of less than 1.25 could occur, as calculated by COLSS, is less than or equal to that which would actually be required in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_{xy} measurement uncertainty factor of 1.053, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for rod bow.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 for CEAC operable or Figure 3.2-3 for both CEACs inoperable can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPC.

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses. Safety analyses cover a pressure range from 2000 psia to 2300 psia. The upper and lower allowable limits (2275 and 2025 psia) are adjusted by 25 psi to bound pressure instrumentation measurement uncertainty.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 138 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.,

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated July 22, 1992, as supplemented by letters dated September 11 and 14, 1992, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit No. 2 (ANO-2) Technical Specifications (TSs). The requested changes would increase the allowable pressurizer pressure range and also allow a lower low pressurizer pressure setpoint for reactor trip, safety injection, and containment cooling. The revisions would change Technical Specification (TS) 3.2.8 and associated Bases to allow plant operation with pressurizer pressure between 2025 and 2275 psia. TS 2.1.1 Bases would be clarified with regards to the application of the peak linear heat rate (PLHR) limit to anticipated operational occurrences analysis results. Also, this revision 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 system (CCAS) trip setpoint and allowable values given in ANO-2 TS Table 3.3-4 would be lowered to 1717.4 and 1686.3 psia respectively by this revision.

The September 11 and 14, 1992, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The ANO-2 plant in the past has had small amounts of safety valve leakage below the allowable TS limit for Reactor Coolant System (RCS) leakage, which was considered insignificant. Also, continuous plant operation runs were sufficiently short and plant shutdowns and forced outages due to other reasons allowed the safety valves to be replaced or repaired if necessary. The ANO-2 plant performance has improved as indicated by longer continuous plant operating times. Recently, this problem alone has resulted in plant shutdown for valve replacement or repair. Although modifications have been made to the safety valves and the valves' discharge piping in an attempt to minimize or eliminate valve simmering/leakage, these changes have not eliminated the problem. Initially, valve simmering is characterized by low volume, high velocity, saturated steam leakage across the valve seats. Prolonged simmering tends to allow an increased steam volume. Continued exposure at the higher volume and velocity will typically cause seat damage, and once seat damage occurs, the valve cannot be reseated into a leak tight condition. Once seat damage occurs the leakage rate tends to increase with time in an exponential manner, eventually causing forced plant shutdown with the attendant economic impacts and associated safety concerns.

By reducing the operating pressurizer pressure for a short time period when the valve starts to simmer, it is expected that the safety valve simmering problem can be substantially curtailed or eliminated, thus eliminating 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. TS 3.2.8 specifies the RCS operating pressure bounds and 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. This 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 small pressure reduction (to 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.

3.0 EVALUATION

To avoid simmering of the safety valve, the licensee proposed to revise the ANO-2 TS 3.2.8, Pressurizer Pressure Limiting Condition for Operation (LCO) to allow plant operation in Mode 1 with pressurizer pressure between 2025 psia and 2275 psia (current TS range of values is between 2225 psia and 2275 psia). These lower pressure limits are consistent with other CE plants (Palo Verde, San Onofre, Waterford). In addition, 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 (current TS values are 1766 psia and 1712.757 psia respectively).

The evaluation is covered in three parts as follows; 1) the pressurizer pressure reduction justification - 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, 2) the clarification to the PLHR TS Bases, and 3) the proposed low pressurizer pressure RPS, SIAS, and CCAS trip setpoint and allowable value changes.

3.1 Pressurizer Pressure Reduction

The safety analyses supporting the Chapter 15 FSAR presently bound plant operation with pressurizer pressure between 2200 and 2300 psia. These analyses were reviewed by the licensee to identify which would be adversely affected by plant operation at a lower pressurizer pressure. Four events were found to be affected by lower pressurizer pressure: 1) Loss of External Load/Turbine Trip, 2) Uncontrolled Control Element Assembly (CEA) Withdrawal, 3) CEA Ejection, and 4) Single Part Length CEA Drop. These events were reanalyzed assuming an initial pressurizer pressure of 2000 psia. The plant responses to these events were simulated using 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 for which the Cycle 5 DNBR limit of 1.25 is applicable for the current analyses. The results, using approved codes and critical heat correlation, were verified to be within the acceptance criteria including the specified acceptable fuel design limits (SAFDLs). No fuel cladding damage is predicted for any event, therefore, no changes to the radiological doses were calculated. The results of the four events analyzed are presented below.

Loss of External Load/Turbine Trip

The Loss of External Load/Turbine trip bounding analysis was performed using an initial RCS pressure of 2000 psia. The event was reanalyzed with conservative assumptions. The results included a peak RCS pressure of 2744 psia and steam generator pressure of 1160 psia. These values are within 110 percent of the design limit pressures of 2750 psia and 1210 psia for the RCS and steam generator, respectively, and are therefore acceptable.

Uncontrolled CEA Withdrawal

The uncontrolled CEA bank withdrawal event from both subcritical and 1% power was evaluated at an RCS pressure of 2000 psia. It is noted that the 100% power case is not adversely affected by the lower RCS pressure due to credit taken for the CPC low DNBR trip which is valid over the proposed pressurizer pressure range.

The subcritical CEA bank withdrawal evaluation uses conservative assumptions for the Cycle 10 analysis. The acceptable fuel design limits (DNBR equal or greater than 1.25 and fuel centerline temperature below 4900 degrees F) were easily met with a great deal of margin.

Conservative assumptions for the CEA withdrawal from 1% power event were used in the analysis. The Variable Over Power Trip (VOPT) is the first trip encountered and terminates the reactor power excursion at a lower level than previously calculated. The results show that the minimum DNBR remains above 1.8, and the maximum linear heat rate remains below 17 kw/ft. We find this acceptable as these values are each within the acceptance criteria of 1.25 and 21 kw/ft, respectively.

CEA Ejection

The CEA ejection events from Hot Full Power (HFP) and Hot Zero Power (HZP) were both reevaluated utilizing the new lower RCS pressure limit of 2000 psia.

The STRIKIN-II computer program was used to simulate the heat conduction within a reactor fuel rod and its associated surface heat transfer. Conservative assumptions were used in the CEA ejection analysis. The maximum centerline enthalpy decreased for both cases. However there was a slight increase in the number of fuel pins (0.32%) having incipient centerline melting for the HFP case, but no fuel pins were calculated as having clad damage or fully molten centerline. Therefore we consider the results from this evaluation as acceptable.

Single Part Length CEA Drop

A single part length CEA (PLCEA) drop incident was reevaluated by the licensee to determine the effects of a reduction of RCS pressure to 2000 psia. 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. 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 licensee used conservative assumptions for the PLCEA drop, which produce the maximum power increase that avoids the high pressurizer pressure trip. The data and algorithms of the CPCs were verified by the licensee to be valid for a range of pressurizer pressures which cover the proposed operating pressure range. An uncertainty factor is applied in the COLSS calculation for DNBR to account for instrument uncertainty on the measured parameters used as inputs to the COLSS calculations. The uncertainty factors were reviewed by the licensee and verified to be conservative over the proposed expanded pressurizer pressure range down to 2000 psia.

3.2 PLHR Clarification

The results of the uncontrolled CEA bank withdrawal from subcritical conditions analysis by the licensee indicated a transient peak linear heat rate (PLHR) in excess of the 21 kw/ft limit given in TS 2.1.1.2. The calculated value was a PLHR of less than 28 kw/ft which exceeded 21 kw/ft for less than one second. The limit of 21 kw/ft is specified based on steady state operation fuel centerline melting temperatures. Therefore, higher linear heat rates can occur under transient conditions without resulting in fuel melting. The fuel centerline melt temperature was acceptable for the subcritical CEA bank withdrawal as indicated previously in a paragraph above. The licensee has proposed a clarification to the Bases of TS 2.1.1.2 to ensure the appropriate application of the peak linear heat rate limit. The linear heat rate and analysis results are typically given in kw/ft for use for monitoring by the CPCs. Therefore, the peak linear heat rate limit of 21 kw/ft is appropriate for most situations. However, 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. We find this clarification to be acceptable as it is consistent with other CE plant interpretations of this Technical Specification (Maine Yankee, Waterford) and is also consistent with the interpretation utilized for the CPCs as documented in the methodology and software manuals.

3.3 Lower Low Pressurizer Pressure Setpoint

The effects of operating with a lower low pressurizer pressure setpoint were examined relative to inadvertent operation of the Emergency Core Cooling System (ECCS) 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. 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 and closer to the SIAS setpoint. As a result, a lower low pressurizer pressure trip setpoint of 1717.4 psia (reduced from 1776 psia) is proposed. Safety analyses identified above as being adversely impacted by the reduced pressurizer pressure were reevaluated down to 2000 psia and found acceptable. The proposed limit of 2025 psia is based on the analysis assumption of 2000 psia plus 25 psi which bounds pressure measurement uncertainties. The basis for the 25 psi error was explained in a letter dated September 11, 1992. Due to the potential for an undesirable SIAS actuation following reactor trip when operating below 2150 psia, operation in the range between 2025 psia to 2150 psia for short durations (24 hours) will be administratively controlled. This will allow operator flexibility when attempting to reseat simmering pressurizer code safety valves and yet minimize the exposure to an undesirable SIAS actuation. We find this acceptable based on the supporting analyses and the permissible administrative control.

The licensee also proposed reductions in the low pressurizer pressure RPS and Engineered Safety Feature Actuation System (ESFAS) trip setpoint and allowable values. These reductions are to help prevent an undesirable SIAS following a reactor trip when operating at reduced pressures. The new low pressurizer pressure setpoints and allowable values were based on new instrument error calculations; the safety analysis setpoint assumptions were not changed. The calculations support the proposed low pressurizer pressure setpoint of 1717.4 psia and the allowable value of 1686.3 psia for the RPS, SIAS and CCAS trip functions.

Regarding the new calculations, the licensee used the statistical method of the square root of the sum of squares (SSRS) to determine the sum of random errors in individual components, and in the complete loop. The licensee combined non-random errors algebraically with the sum of random errors to establish the total uncertainty of the instrument loop.

The licensee removed from the calculation the terms for all seismic errors and for the cumulative effects of background radiation. The licensee stated that the concurrent occurrence of an accident and a seismic activity is beyond the design basis of the plant. The licensee also implemented comprehensive procedures at the plant to assess the effects of seismic activity immediately after it occurs and that the plant operators are trained for these procedures. The licensee stated that cumulative effects of background radiation is calibrated out during each calibration and effects of cumulative dose of the background radiation for a period between the two successive calibrations is very negligible. Therefore, the licensee did not consider the cumulative effects of background radiation in the calculation. The staff found these explanations acceptable.

The licensee used a currently accepted methodology for calculating setpoints. In addition, the licensee uses this methodology for calculation 91-EQ-2002-02, Revision 0, "Loop Error, Setpoint, and Time Response Analysis for Narrow Range Containment Building Pressure ESFAS and RPS Trip Functions," which the staff recently approved for another TS amendment. The licensee has not committed to strict compliance with the guidance in ISA-S67.04-1988 "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants." However, the licensee considered the guidelines of this ISA standard in calculating the loop errors, periodic test errors, and allowable values associated with the low pressurizer pressure setpoints. The calculation methodology is acceptable to the staff.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 37567). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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