Docket No. 50-368 April 11, 1994

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. 157 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M87066)

The Commission has issued the enclosed Amendment No. 157 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, 1993, as supplemented by letter dated October 20, 1993.

The amendment removes the cycle-specific variables from the TSs and controls them under a new document called the Core Operating Limits Report (COLR), in accordance with generic letter (GL) 88-16. However, your proposed shutdown margin definition, which is not related to GL 88-16, is denied since it does not accurately define the starting point for shutdown margin.

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. A copy of the Notice of Partial Denial of Amendment to be published in the Federal Register will be provided to you under separate cover.

Sincerely,

ORIGINAL SIGNED BY:

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

Enclosures:
1. Amendment No. 157 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

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*See previous concurrence

OFC	LA:PD4-1	PM:PD4-1	(A)BC/SRXB*	OGC*	D:PD4-1
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Sincerely,

Thomas W. Alexion

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Enclosures:
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2. Safety Evaluation

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See next page

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REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the CORE OPERATING LIMITS REPORT with:

a. CEA insertion between the Long Term Steady State Insertion Limit and the Transient Insertion Limit restricted to:

1. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day intervals, and

2. ≤ 14 Effective Full Power Days per calendar year.

b. CEA insertion between the Short Term Steady State Insertion Limit and the Transient Insertion Limit shall be restricted to ≤ 4 hours per 24 hour interval.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the CORE OPERATING LIMITS REPORT.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limit within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

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* See Special Test Exceptions 3.10.2 and 3.10.4.

With K_{eff} ≥1.0.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Short Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 4 hours per 24 hour interval, operation may proceed provided any subsequent increase in thermal power is restricted to $\leq$ 5% of rated thermal power per hour.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limit at least once per 12 hours except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours.

ACTION: (Continued)

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

    a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

    b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours.

f. With one full length or part length CEA trippable but misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with the limits specified in the CORE OPERATING LIMITS REPORT and within one hour, the misaligned CEA is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

    a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

    b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With one full length CEA trippable but inoperable due to causes other than addressed by ACTION (a), above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

c. With one full length CEA trippable but inoperable due to causes other than addressed by ACTION (a), above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in full length CEA group 6, operation in MODES 1 and 2 may continue.

d. With more than one full length or part length CEA trippable but inoperable due to causes other than addressed by ACTION a, above, restore the inoperable CEA(s) to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

e. With one or more full length or part length CEAs trippable but misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with the limits specified in the CORE OPERATING LIMITS REPORT and within 1 hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

*See Special Test Exceptions 3.10.2 and 3.10.4.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one of the following sources with a minimum solution temperature of 55°F.

1. One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, or

2. Two boric makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A contained borated water volume of between 464,900 and 500,500 gallons (equivalent to an indicated tank level of between 91.7% and 100%, respectively),

2. Between 2500 and 3000 ppm of boron,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 110°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the make up tank(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least that specified in the CORE OPERATING LIMITS REPORT at 200°F; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each of the above required borated water sources shall be demonstrated OPERABLE:

BORIC ACID MAKEUP PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid makeup pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid makeup pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid makeup pump(s) in Specification 3.1.2.2 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least that specified in the CORE OPERATING LIMITS REPORT at 200°F; restore the above required boric acid pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional Surveillance Requirements other than those required by Specification 4.0.5.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least that specified in the CORE OPERATING LIMITS REPORT at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE, depending on the volume available in the boric acid makeup tanks.

a. If the contents of ONE boric acid makeup tank meet the volume requirements of Figure 3.1-1, two of the following three flow paths to the Reactor Coolant System shall be OPERABLE:

1. One flow path from the appropriate boric acid makeup tank via a boric acid makeup pump and a charging pump.
2. One flow path from the appropriate boric acid makeup tank via a gravity feed connection and a charging pump.
3. One flow path from the refueling water tank via a charging pump.

OR

b. If the contents of Both boric acid tanks are needed to meet the volume requirements of Figure 3.1-1, four of the following five flow paths to the Reactor Coolant System shall be OPERABLE:

1. One flow path from boric acid makeup tank A via a boric acid makeup pump and a charging pump.
2. One flow path from boric acid makeup tank B via a boric acid makeup pump and a charging pump.
3. One flow path from boric acid makeup tank A via a gravity feed connection and a charging pump.
4. One flow path from boric acid makeup tank B via a gravity feed connection and a charging pump.
5. One flow path from the refueling water tank via a charging pump.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With any of the boron injection flow paths to the Reactor Coolant System required in (a) or (b) above inoperable, restore the inoperable flow path to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least that specified in the CORE OPERATING LIMITS REPORT at 200°F within the next 6 hours; restore the flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT. The maximum upper design limit shall be:

a. Less positive than $+0.5 \times 10^{-4}$ Δk/k/°F whenever THERMAL POWER is ≤70% of RATED THERMAL POWER, and

b. Less positive than 0.0 Δk/k/°F whenever THERMAL POWER is >70% of RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, prior to reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

c. At any THERMAL POWER, within 14 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 525°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 525°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 525°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 535°F.

#With $K_{eff} \geq 1.0$.

*See Special Test Exception 3.10.5.

SHUTDOWN MARGIN-T_{avg} ≤ 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN-T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of (e) below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2815 MWt.

OPERATIONAL MODE - MODE

1.4 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - The injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - The injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels - The exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

AZIMUTHAL POWER TILT - T_q

1.17 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18 DOSE EQUIVALENT I-131 shall be that concentration of 1-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.19 - shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL SHAPE INDEX

1.22 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

REACTOR TRIP SYSTEM RESPONSE TIME

1.23 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.24 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

SOFTWARE

1.26 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation and procedures.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.27 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

SOURCE CHECK

1.28 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to the radioactive source.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.29 An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints.

LIQUID RADWASTE TREATMENT SYSTEM

1.30 A LIQUID RADWASTE TREATMENT SYSTEM is a system designed and installed to reduce radioactive liquid effluents from the unit. This is accomplished by providing for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the environment.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents from the plant by collecting offgases from radioactive systems and providing for decay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Atmospheric cleanup systems that are Engineered Safety Feature (ESF) actuated are not considered to be VENTILATION EXHAUST TREATMENT SYSTEMS.

MEMBER(S) OF THE PUBLIC

1.33 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

PURGE-PURGING

1.34 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to reduce airborne radioactive concentrations in such a manner that replacement air or gas is required to purify the confinement.

EXCLUSION AREA

1.35 The EXCLUSION AREA is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

UNRESTRICTED AREA

1.36 An UNRESTRICTED AREA shall be any area at or beyond the exclusion area boundary.

CORE OPERATING LIMITS REPORT

1.37 The CORE OPERATING LIMITS REPORT is the ANO-2 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.9.5. Plant operation within these operating limits is addressed in individual specifications.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 300°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 300°F
3. HOT STANDBY	< 0.99	0	$\geq$ 300°F
4. HOT SHUTDOWN	< 0.99	0	300°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

DEFINITIONS

SECTION	PAGE

1.0 DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>INDEX</u>

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

(continued from previous page)

REMOVE PAGES	INSERT PAGES
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-10a	--
3/4 2-11	--
3/4 2-12	--
3/4 2-13	--
3/4 2-14	--
3/4 10-2	3/4 10-2
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
6-16	6-16
6-17	6-17
6-18	6-18
6-19	6-19
6-19a	--
6-20	6-20
6-21	6-21
--	6-21a

ATTACHMENT TO LICENSE AMENDMENT NO. 157

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
I	I
II	II
IV	IV
V	V
XVII	XVII
1-1	1-1
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-5	3/4 1-5
3/4 1-8	3/4 1-8
3/4 1-10	3/4 1-10
3/4 1-12	3/4 1-12
3/4 1-15	3/4 1-15
3/4 1-17	3/4 1-17
3/4 1-18	3/4 1-18
3/4 1-19a	--
3/4 1-25	3/4 1-25
3/4 1-27	3/4 1-27
3/4 1-28	--
3/4 1-29	--
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a	--
3/4 2-8	3/4 2-8

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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 157, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William D. Beckner

William D. Beckner, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 11, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
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, 1993, as supplemented by letter dated October 20, 1993, 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.

6.0 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 and changes surveillance requirements. 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 (58 FR 46230). 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.

7.0 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.

Principal Contributors: T. Huang, SRXB
T. Alexion, PD IV-1

Date: April 11, 1994

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The part length CEA group shall be limited to the insertion limits specified in the CORE OPERATING LIMITS REPORT with PLCEA insertion between the Long Term Steady State Insertion Limit and the Transient Insertion Limit restricted to:

a. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

b. ≤ 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODE 1*.

ACTION:

a. With the part length CEA groups inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the part length CEA group to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the PLCEA group position as specified in the CORE OPERATING LIMITS REPORT.

b. With the part length CEA groups inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

   1. Restore the part length group to within the Long Term Steady State Insertion limits within two hours, or

   2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENT

4.1.3.7 The position of the part length CEA group shall be determined to be within the Transient Insertion Limit at least once per 12 hours. The accumulated time during which the part length CEA group is inserted beyond the Long Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours.

*See Special Test Exception 3.10.2.

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

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

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but within the limit specified in the CORE OPERATING LIMITS REPORT, within two hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed the limit specified in the CORE OPERATING LIMITS REPORT:

   1. Due to misalignment of either a part length or full length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

   2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Linear Power Level - High trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR limit shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and at least one CEAC is operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13.0% (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and at least one CEAC is operable); or

d. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

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

ACTION:

a. With COLSS in service and the DNBR limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

   1. Restore the DNBR to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the DNBR limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the DNBR to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC Channel (when COLSS is out of service).

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

ACTION:

a. With COLSS in service and the linear heat rate limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limit and either:

   1. Restore the linear heat rate to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the linear heat rate limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the linear heat rate to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

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

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTOR by a factor equivalent to $\geq F^m_{xy}/F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the CLOSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}); or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}), obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 days of accumulated operation in MODE 1.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 120.4×10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature (T_c) shall be maintained between 542°F and 554.7°F.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the Reactor Coolant Cold Leg Temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit at least once per 12 hours.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

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.

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.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The linear heat rate limit shall be maintained by either:

   1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

   2. Operating within the region of acceptable operation as specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service.)

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With any of the above limits being exceeded while any of the above requirements are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of the above Specification, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which any of the above requirements are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within its limits during PHYSICS TESTS above 5% of RATED THERMAL POWER in which any of the above requirements are suspended.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident, and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} ≤200°F, the reactivity transients resulting from any postulated accident are minimal and the shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 6,650 cubic feet in approximately 25 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentrations associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle. The MTC limits defined in the Technical Specification are maximum upper design limits. Actual operating limits are specified in the CORE OPERATING LIMITS REPORT.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operations. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup pumps, 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of that specified in the CORE OPERATING LIMITS REPORT after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 and a small fraction of the borated water from the refueling water tank required in Specification 3.1.2.8.

The requirement in Technical Specification 3.1.2.8 for a minimum contained volume of 464,900 gallons of 2500-3000 ppm borated water in the refueling water tank ensures the capability for borating the RCS to the desired concentration. The value listed is consistent with the plant ECCS requirements.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a sufficient SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either borated water from the refueling water tank or boric acid solution from the boric acid makeup tank(s) in accordance with the requirements of Specification 3.1.2.7.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics. The 61,370 gallon limit for the refueling water tank is based upon having an indicated level in the tank of at least 7.5%.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 8.8 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA or a large misalignment (≥ 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN. CEAs that are confirmed to be inoperable due to problems other than addressed by ACTION a of Specification 3.1.3.1 will not impact SHUTDOWN MARGIN as long as their relative positions satisfy the applicable alignment requirements.

For small misalignments (< 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing 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 operating limit power level based on linear heat rate, 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 specified in the CORE OPERATING LIMITS REPORT can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

3/4.2.2 RADIAL PEAKING FACTORS

Limiting the values of the planar radial peaking factors (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured planar radial peaking factors (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured planar radial peaking factors. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than the limit specified in the CORE OPERATING LIMITS REPORT is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\theta - \theta_o)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

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 uncertainty 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 specified in the CORE OPERATING LIMITS REPORT 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.

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5. Reports required on an annual basis shall include:

a. A tabulation on an annual basis for the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received form external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).

c. Documentation of all challenges to the pressurizer safety valves.

d. A diesel generator data report which provides the number of valid tests and the number of valid failures for each diesel generator.

e. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded the results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Administrator of the Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Inoperable Fire Detection Instrumentation

f. Inoperable Fire Suppression Systems

g. Deleted.

h. Radioactive Effluents, Specifications 3.11.1.1, 3.11.1.2, 3.11.1.3, 3.11.2.2, 3.11.2.3, 3.11.2.4, 3.11.2.5, and 3.11.3.

This report shall include the following:

1) Description of occurrence.

2) Identify the cause(s) for exceeding the limit(s)

3) Explain corrective action(s) taken to mitigate occurrence.

4) Define action(s) taken to prevent recurrence.

5) Summary of consequence(s) of occurrence.

6) Describe levels exceeding 40CFR190 in accordance with 10CFR20.405(c).

i. Inoperable Containment Radiation Monitors, Specification 3.3.3.1.

j. Steam Generator Tubing Surveillance -- Category C-3 Results, Specification 4.4.5.5.

k. Maintenance of Spent Fuel Pool Structural Integrity, Specification 3.7.12.

l. Radiological Environmental Monitoring Sample Analysis, Specification 3.12.1.

m. Unplanned Offsite Release during one hour period of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. This report shall be submitted within 30 days of the occurrence of the event and shall include the following information:

1. Description of the occurrence.

2. Identify the cause(s) of exceeding the limit(s).

3. Explain corrective action(s) taken to mitigate occurrence.

4. Define action(s) taken to prevent recurrence.

5. Summary of the consequence(s) of occurrence.

n. Inoperable Reactor Vessel Level Monitoring System (RVLMS), Specification 3.3.3.6, Table 3.3-10 Item 14.

o. Inoperable Main Steam Line Radiation Monitors, Specification 3.3, Table 3.3-6.

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.3 Routine radioactive effluent release reports covering the operating of the unit during the previous 6 months of operations shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.3.1 The radioactive effluent release report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste release from the unit. The data will be summarized on a quarterly basis following the format of Regulatory Guide 1.21, Revision 1.

6.9.3.2 Any changes in the OFFSITE DOSE CALCULATION MANUAL and PCP shall be included in the semiannual report for the period in which the change(s) was made effective.

6.9.3.3 The radioactive effluent release reports shall include the following information for all unplanned releases to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents:

1. Description of the occurrence.

2. Identify the cause(s) for exceeding the limit(s).

3. Explain corrective actions taken to mitigate occurrence.

4. Define action(s) taken to prevent recurrence.

5. Summary of consequence(s) of occurrence.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste system, the submittal shall specify the releases of radioactive material from each unit.

6.9.3.4 The first report filed each year shall contain:

1. A summary of the hourly meteorological data collected over the previous calendar year. In lieu of including this summary in the report, the data may be retained by the licensee for NRC review and noted as such in the report.

2. A summary of radiation doses due to radiological effluent during the previous calendar year calculated in accordance with the methodology specified in the OFFSITE DOSE CALCULATION MANUAL.

3. The radiation dose to members of the public due to their activities inside the site boundary. This calculated dose shall include only those dose contributions directly attributed to operation of the unit and shall be compared to the limits specified in 40 CFR 190.

6.9.3.5 The first report filed each year shall contain description of licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) during the previous calendar year.*

*This information may be included in the annual FSAR update in lieu of inclusion in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.4 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of year.

a. The annual radiological environmental operating report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

b. The annual radiological environmental operating reports shall include summarized and tabulated results of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

c. The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

CORE OPERATING LIMITS REPORT

6.9.5 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or any remaining part of a reload cycle.

6.9.5.1 The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at ANO-2, specifically:

1) "The ROCS and DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, and 3.1.3.6 for Regulating CEA Insertion Limits).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6 for Regulating CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

3) "Statistical Combination of Uncertainties, Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One Unit 2," CEN-139(A)-P, November 1980 (Methodology for Specification 3.2.4 for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 1, February 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 2-P, July 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

7) "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CEN-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Calculational Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

CORE OPERATING LIMITS REPORT

9) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating CEA Insertion Limits, and 3.1.3.7 for Part Length CEA Insertion Limits).

10) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.9.5.1.4, 6.9.5.1.5, and 6.9.5.1.8 methodologies.

11) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.9.5.1.6 methodology.

12) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.9.5.1.9 methodology.

6.9.5.2 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.5.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

(1) The definition section of the TSs was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with reference to the COLR that provides these limits.

(a) Specification 3/4.1.1

Shutdown margin limits for this specification (T_{avg} > 200°F and T_{avg} ≤ 200°F) are specified in the COLR.

(b) Specification 3.1.1.4

The moderator temperature coefficient (MTC) limits for this specification are specified in the COLR.

(c) Specification 3.1.3.1

The core power limits for the moveable control rod assemblies - CEA position deviation are specified in the COLR.

(d) Specifications 3.1.3.6 and 3.1.3.7

The regulating CEA insertion limits and part length CEA insertion limits for these specifications are specified in the COLR.

(e) Specifications 3/4.2.1 and 3.10.2.b.2

The linear heat rate limit for these specifications is specified in the COLR.

(f) Specification 3.2.3

The maximum azimuthal power tilt limits for this specification are specified in the COLR.

(g) Specification 3/4.2.4

The DNBR margin related limits for this specification are specified in the COLR.

(h) Specification 3.2.7

The axial shape index range limits for this specification are specified in the COLR.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 157 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, 1993, as supplemented by letter dated October 20, 1993, 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 remove the cycle-specific variables from the TSs and control them under a new document called the Core Operating Limits Report (COLR), in accordance with Generic Letter (GL) 88-16. The proposed amendment would also revise the definition of shutdown margin (SDM) in Technical Specification 1.13 to achieve consistency with NUREG-1432, Revision 0, "Revised Standard Technical Specifications (RSTS) for CE (Combustion Engineering) Plants.

The October 20, 1993, supplement provided an additional page that was inadvertently omitted from the proposed TSs. This additional page would also reference the COLR and does not change the initial proposed no significant hazards consideration determination.

2.0 DISCUSSION

The proposed changes would modify TSs having cycle-specific parameter limits by replacing the values of those limits with a reference to a COLR for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TSs. Guidance on the proposed changes was developed by NRC and provided to all power reactor licensees and applicants by GL 88-16, dated October 4, 1988.

The request to change the SDM definition is unrelated to GL 88-16.

3.0 EVALUATION

3.1 COLR

The proposed changes to the TSs are in accordance with the guidance provided by GL 88-16 and are addressed below.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.9 is revised to include the COLR under the reporting requirements of the Administrative Control section of the TSs. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the NRC-approved methodologies be used in establishing the values of these limits for the relevant specifications and that the values be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April 1983, (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, and 3.1.3.6 for Regulating CEA Insertion Limits).

(b) CENPD-190-A, "C-E Method for Control Element Assembly Ejection Analysis," January 1976, (Methodology for Specification 3.1.3.6 for Regulating CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

(c) CEN-139(A)-P, "Statistical Combination of Uncertainties, Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One Unit 2," November 1980, (Methodology for Specification 3.2.4 for DNBR Margin and 3.2.7 for ASI).

(d) CENPD-132-P, " Calculational Methods for the CE Large Break LOCA Evaluation Model," August 1974 - Approved June 1975; Supplement No. 1, February 1975 - Approved June 1975; No. 2, July 1975 - approved December 1975, (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

(e) CENPD-137-P, "Calculational Methods for the CE Small Break LOCA Evaluation Model," August 1974 - Approved June 1975, (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

(f) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 - Approved March 1984, (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating CEA Insertion Limits, and 3.1.3.7 for Part Length CEA Insertion Limits).

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits. Furthermore, the licensee should identify the applicable specifications to be relocated to the COLR under each approved methodology cited above.

3.2 SDM Definition

The licensee's proposed revision to the definition of SDM is denied because the licensee did not propose all portions of the SDM definition in NUREG-1432, Revision 0, "Revised Standard Technical Specifications for CE Plants" (hereafter referred to as the RSTS). Specifically, the licensee did not include the RSTS statement, "In Modes 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level." The licensee's reason for not including this portion, as discussed in the application, is that this RSTS statement is not applicable to the ANO-2 SDM calculation method, and that another specification verifies the SDM in Modes 1 and 2 by verifying that the CEA group withdrawal is within the transient insertion limits.

The staff position is that the above RSTS statement is applicable and is needed to accurately define the starting point for SDM. The verification of the CEA group withdrawal assists in ensuring the SDM is met, however, the staff is of the opinion that how SDM is met and the definition of SDM are two different concepts that should be separated.

Similarly, the licensee did not include the RSTS statement, "There is no change in part length CEA position." In the application, the licensee offered no reason for not including this statement. The staff position is that this statement is also necessary to accurately define the starting point for the SDM definition.

4.0 SUMMARY

On the basis of its review, the NRC staff finds that the licensee provided an acceptable response to the items in GL 88-16 on modifying cycle-specific parameter limits in the TSs. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC-approved methodologies, the NRC staff finds that this change has no impact on plant safety.

Regarding the licensee's proposed SDM definition, the staff is denying this part of the application since it did not accurately define the starting point for shutdown margin. A Notice of Partial Denial of Amendment to Facility Operating License and Opportunity for Hearing is being prepared under separate cover and will appear in the Federal Register.

5.0 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.