Docket No. 50-313

April 26, 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. 171 TO FACILITY OPERATING LICENSE NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M86005)

The Commission has issued the enclosed Amendment No. 171 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 24, 1993.

The amendment corrects typographical errors in the ANO-1 TSs. The errors were introduced in the original ANO-1 TSs and in subsequent amendments. These changes are administrative in nature and are intended to improve the readability of the TSs without changing the meaning or intent of any specifications.

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:

George Kalman, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 171 to DPR-51
2. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	NRC/Local PDR	PD4-1 Reading	OGC (15B18)
E. Adensam	P. Noonan	G. Kalman (2)	OPA (2G5)
D. Hagan (3206)	G. Hill (2)	W. Johnson, RIV	J. Roe
C. Grimes (11E22)	ACRS (10) (P315)	OC/LFMB (4503)	W. Beckner

OFC	LA:PD4-1	PM:PD4-1	OGC	D:PD4-1
NAME	PNoonan	GKalman/vw	C Marco	WBeckner
DATE	3/31/94	4/5/94	4/15/94	4/26/94
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 26, 1994

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George Kalman

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Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

cc:

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated February 24, 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.

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. DPR-51 is hereby amended to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 171, 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

William 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 26, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace 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.

REMOVE PAGES	INSERT PAGES
4	4
15b-3	15b-3
16	16
16a	16a
25	25
36	36
40a	40a
43	43
48a	48a
72b-1	72b-1
80	80
85a	85a
105	105
110ss	110ss
110tt	110tt
113	113
140	140

1.5.4 Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses. Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent

$$100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}}-1\right)$$

The power in any quadrant is determined from the power range channel displayed on the console for that quadrant. The average power is determined from an average of the outputs of the power range channels. If one of the power range channels is out of service, the remaining three operable power range channels or the incore detectors will be used to determine the average power. The quadrant power tilt limits as a function of power are stated in Specification 3.5.2.4.

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

initiated or that higher modes of operation are not entered when corrective action is being taken to obtain compliance with a Specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with Action requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a mode change. Therefore, in this case, if the requirements for continued operation have been met in accordance with the requirements of the specification, then entry into that mode of operation is permissible. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with Action requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower mode of operation. For the purpose of compliance with this specification the term 'shutdown' is defined as a required reduction in the REACTOR OPERATING CONDITION.

3.0.5 Delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 7 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.

3.1.1.2 Steam Generator

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.

3.1.1.3 Pressurizer Safety Valves

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

3.1.1.5 Reactor Coolant Loops

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.1.1.6 Decay Heat Removal

With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition, at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:*

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.

3.1.5 Chemistry

Applicability

Applies to the limiting conditions of reactor coolant chemistry for continuous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities in the reactor coolant.

Specification

3.1.5.1 The following limits shall not be exceeded for the listed reactor coolant conditions.

Contaminant	Specification	Reactor Coolant Conditions
Oxygen as O_2	0.10 ppm max	above 250°F
Chloride as Cl^-	0.15 ppm max	above cold shutdown conditions
Fluoride as F^-	0.15 ppm max	above cold shutdown conditions

3.1.5.2 During operation above 250°F, if any of the specifications in 3.1.5.1 is exceeded, corrective action shall be initiated within 8 hours. If the concentration limit is not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.3 During operations between 250°F and cold shutdown conditions, if the chloride or fluoride specification in 3.1.5.1 are exceeded, corrective action shall be initiated within 8 hours to restore the normal operating limits. If the specifications are not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.4 If the oxygen concentration and either the chloride or fluoride concentration of the primary coolant system exceed 1.0 ppm, the reactor shall be immediately brought to the hot shutdown condition using normal shutdown procedures, and action is to be taken immediately to return the system to within normal operation specifications. If specifications given in 3.1.5.1 have not been reached in 12 hours, the reactor shall be brought to a cold shutdown condition using normal procedures.

Bases

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack ([1],[2]).

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Objectivity

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

3.3.1 The following equipment shall be operable whenever containment integrity is established as required by Specification 3.6.1:

(A) One reactor building spray pump and its associated spray nozzle header.

(B) One train of reactor building emergency cooling.

(C) Two out of three service water pumps shall be operable, powered from independent essential buses, to provide redundant and independent flow paths.

(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable.

(E) Both low pressure injection coolers and their cooling water supplies shall be operable.

(F) Two Borated Water Storage Tank (BWST) level instrument channels shall be operable.

(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.

(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.

3.4.3 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig.

3.4.4 Components required to be operable by Specifications 3.4.1, 3.4.2, and 3.4.3 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specifications 3.4.1, 3.4.2 and 3.4.3 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specifications 3.4.1, 3.4.2, and 3.4.3 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.4.5 If the condition specified in 3.4.1.4 cannot be met:

1. With one EFW flow path inoperable, the unit shall be brought to HOT SHUTDOWN within 36 hours, and if not restored to an operable status within the next 36 hours, the unit shall be brought to COLD SHUTDOWN within the next 12 hours or at the maximum safe rate.

2. If both EFW trains are inoperable, restore one train to operable status within one hour or be in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the next 12 hours or at the maximum safe rate.

3. If both EFW trains and the AFW pump are inoperable, the unit shall be immediately run back to ≤5% full power with feedwater supplied from the MFW pumps. As soon as an EFW train or the AFW train is operable, the unit shall be placed in COLD SHUTDOWN within the next 12 hours or at the maximum safe rate.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required simultaneou all other engineering and unc ainty factors are also at their limits.* Conservatism is introduced by application of:

a. Nuclear uncertainty factors.
b. Thermal calibration.
c. Fuel densification effects.
d. Hot rod manufacturing tolerance factors.
e. Fuel rod bowing.

The 20 ±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower parts of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping bank)

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full-out position ([1]). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis ([2]) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot zero power, would result in a lower transient peak thermal power and therefore less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

Control rod Groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 20%. The normal position at power is for Groups 6 and 7 to be partially inserted.

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*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
47. RCS Subcooling Margin Monitor	D	NA	R	
48. Electromatic Relief Valve Flow Monitor	D	NA	R	
49. Electromatic Relief Block Valve Position Indicator	D	NA	R	
50. Pressurizer Safety Valve Flow Monitor	D	NA	R	
51. Prèssurizer Water Level Indicator	D	NA	R	
52. Control Room Chlorine Detector	D	M	R	
53. EFW Initiation				
a. Manual	NA	M	NA	
b. SG Low Level, SGA or B	S	M	R	
c. Low Pressure SGA or B	S	M	R	
d. Loss of both MFW Pumps and PWR > 10%	S	M	R	

Where	(L_a)	Design Basis Accident Leakage Rate at Pressure P_a
	(L_t)	Maximum Allowable Test Leakage Rate at Reduced Test Pressure P_t Under Test Condition
	(L_{ao})	Maximum allowable operational leakage rate at pressure P_a
	(L_{to})	Maximum allowable leakage rate at pressure P_t
	(L_{am})	Initial Measured Leakage Rate at Pressure P_a
	(L_{tm})	Initial Measured Leakage Rate at Pressure P_t
	(P_a)	Peak Test Pressure of 59 psig
	(P_t)	Reduced Test Pressure of 30 psig

4.4.1.1.3 Conduct of Tests

a. Leakage rate tests should not be started until essential temperature equilibrium has been attained. Containment test conditions should stabilize for a period of about four hours prior to the start of a leakage rate test.

b. The leakage rate test period shall extend to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, the agreed upon shorter period may be used.

c. Test accuracy shall be verified by supplementary means, such as measuring the quantity of air required to return to the starting point or by imposing a known leak rate to demonstrate the validity of measurements.

d. Closure of reactor building isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercise or adjustment.

4.4.1.1.4 Frequency of Test

After the initial preoperational leakage rate test, a set of three integrated leak rate tests shall be performed at approximately equal intervals during each 10 year service period, with the third test of each set coinciding with the end of each 10-year service period. The test may coincide with the plant inservice inspection shutdown periods.

Should the inspection of one of the wires reveal any significant physical change (pitting or loss of area), additional wires shall be removed from the applicable surveillance tendons and inspected to determine the extent and cause of change. The sheathing filler will be sampled and inspected for changes in physical appearance. (See Applicable Acceptance Criteria in Section 4.4.2.1.3)

4.4.2.1.3 Acceptance Criteria

The Reactor Building Post Tensioning System shall be considered acceptable if the following acceptance criteria are met.

1. Each surveillance tendon has a normalized lift-off force equaling or exceeding its expected prestress force for the time of the test. See Figures 4.4.2-1, -2, and -3. If the normalized lift-off force of any one tendon in a group lies between the expected prestress force and the lower bound prestress force, an adjacent tendon on each side shall be checked for lift-off force. If both of these tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. If either of the adjacent tendons is found unacceptable, it shall be considered evidence of possible abnormal degradation of the containment structure. (See TS 6.12.5)

   If the normalized lift-off force of any single tendon lies below the lower bound prestress force, the occurrence should be considered evidence of possible abnormal degradation of the containment structure. (See TS 6.12.5)

2. The wires removed from three detensioned surveillance tendons (one dome, one vertical and one hoop) shall be inspected for corrosion, cracks, or other damage over the entire length of the wire. The presence of abnormal corrosion, cracks, or other damage shall be considered evidence of possible abnormal degradation of the containment structure. (See TS 6.12.5)

   A minimum of three (3) wire samples cut from each removed wire (one from each end and one at mid length) shall be subjected to a tensile test. Failure of any one of these wire samples to meet a minimum ultimate tensile strength of 240 ksi shall be considered evidence of possible abnormal degradation of the containment structure. (See TS 6.12.5)

3. Sheathing Filler material samples from each surveillance shall be considered acceptable provided the results of the tests performed on the samples fall within the following limits.

1.	Water Soluble Chlorides	less than	10 ppm
2.	Water Soluble Nitrates	less than	10 ppm
3.	Water Soluble Sulfides	less than	10 ppm
4.	Water Content	less than	10% Dry Weight

4.8 EMERGENCY FEEDWATER PUMP TESTING

Applicability

Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.

Objective

To verify that the emergency feedwater pump and associated valves are operable.

Specification

4.8.1 Each EFW train shall be demonstrated operable:

a) By verifying on a STAGGERED TEST BASIS:

1. at least once per 31 days or upon achieving hot shutdown following a plant heatup and prior to criticality, that the turbine-driven pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm through the test loop flow path.

2. at least once per 31 days by verifying that the motor driven EFW pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm thorough the test loop flow path.

b) At least once per 31 days by verifying that each valve (manual, power operated or automatic) in each EFW flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

c) Prior to exceeding 280°F reactor coolant temperature and after any EFW flowpath manual valve alterations by verifying that each manual valve in each EFW flowpath which, if mispositioned may degrade EFW operation, is locked in its correct position.

d) At least once per 92 days by cycling each motor-operated valve in each flowpath through at least one complete cycle.

e) At least once per 18 months by functionally testing each EFW train and:

1) Verifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actuation signal.

4.30 RADIOLOGICAL ENVIRONMENTAL MONITORING

4.30.1 Radiological Environmental Monitoring Program Description

Applicability

Applies at all times.

Objective

To provide information on the radiological effects of station operation on the environment.

Specification

4.30.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 4.30-1 and shall be analyzed pursuant to the requirements of Tables 4.30-1 and 4.30-2. The sample locations shall be listed in Table 4-1 in the ODCM.

4.30.1.2 a. With the radiological environmental monitoring program not being conducted as specified in Table 4.30-1, prepare and submit to the Commission in the Annual Radiological Environmental Report a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if specimens are not obtainable due to hazardous conditions, seasonal unavailability, or to malfunction of sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.)

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at one or more of the locations specified in Table 4.30-1 exceeding the limits of Table 4.30-3 when averaged over any calendar quarter, prepare and submit to the Commission, within 30 days from the end of the affected quarter, a Special Report which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits of Table 4.30-3 to be exceeded, and defines the actions taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.25.1.2 and 3.25.2.2. When more than one of the radionuclides in Table 4.30-3 are detected in the sampling medium, this Special Report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{reporting level (1)}} + \frac{\text{Concentration (2)}}{\text{reporting level (2)}} + \dots \geq 1.0$$

When radionuclides other than those in Table 4.30-3 are detected and are the result of plant effluents, this Special Report shall be submitted if the potential annual dose to a member of the public is equal to or greater than the calendar year limits of Specifications 3.25.1.2 and 3.25.2.2. This Special Report is not required if the measured level of radioactivity was not the result of plant effluents, however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Report.

c. With milk or fresh leafy vegetable samples unavailable from any of the sample locations required by Table 4.30-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the causes of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised table for the ODCM reflecting the new location(s).

d. The provisions of Specification 3.0.3 are not applicable.

4.30.1.3 The results of analyses performed on the radiological environmental monitoring samples shall be summarized in the Annual Radiological Environmental Report.

Bases:

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluents monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.30-2 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. ([1])

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. ([2])

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. ([3])

REFERENCES:

(1) FSAR Section 5.1

(2) FSAR Section 5.2.5

(3) FSAR Section 6.5

6.12 REPORTING REQUIREMENTS

6.12.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the appropriate NRC Regional Office unless otherwise noted.

6.12.2 Routine Reports

6.12.2.1 Startup Report

A summary report of plant startup and power escalation testing shall be submitted following: 1) receipt of an operating license, 2) amendment to the license involving a planned increase in power level, 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and 4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within 1) 90 days following completion of the startup test program, 2) 90 days following resumption or commencement of commercial power operation, or 3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

6.12.2.2 Occupational Exposure Data Report 1/

An Occupational Exposure Data Report for the previous calendar year shall be submitted prior to March 1 of each year. The report shall contain a tabulation on an annual basis of 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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated February 24, 1993, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit No. 1 (ANO-1) Technical Specifications (TSs). The requested changes correct a number of typographical, grammatical and format errors in various sections of the TSs. In the course of reviewing the errors identified by the licensee, the NRC staff found several discrepancies in the revised pages of the proposed TSs submitted by the licensee. Following discussion with the licensee these discrepancies were resolved and the proposed TSs were revised accordingly. In all cases the revisions were administrative in nature and did not change the meaning or intent of the TSs.

2.0 EVALUATION

The administrative errors identified by the licensee were reviewed by the NRC staff to ensure that the proposed changes do not change the meaning or intent of the TSs. The proposed revisions were reviewed for clarity and thoroughness. Minor changes recommended by the NRC staff to resolve discrepancies in the licensee submittal were adopted and included in the revised TSs.

The changes do not provide any relief from or add requirements to the TSs, or change the intended operation or administrative requirements of the plant or its design basis. Based on the above considerations, the staff finds that the changes are appropriate, and the proposed amendment is acceptable.

3.0 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.

4.0 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 67843). 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.

5.0 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 Contributor: G. Kalman

Date: April 26, 1994