July 26, 1993

Docket No. 50-368

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. 149 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M86420)

The Commission has issued the enclosed Amendment No. 149 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 May 7, 1993.

The amendment corrects typographical errors that were introduced in the original TSs and in subsequent amendments.

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

Sincerely,

ORIGINAL SIGNED BY:

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

Enclosures:
1. Amendment No. 149 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

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

OFC	LA/PD4-1	PM/PD4-1	BC/SCSB	OGC	(A)D/PD4-1
NAME	PNoonan	TAlexion/vw	RBarrett	A. Jorgensen	HRood
DATE	6/16/93	6/22/93	6/25/93	7/6/93	7/26/93
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

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

July 26, 1993

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

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See next page

Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 2

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
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 May 7, 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.

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

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 149, 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

Harry Rood

Harry Rood, Acting Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: July 26, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 149

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
1-7	1-7
3/4 1-1	3/4 1-1
3/4 1-8	3/4 1-8
3/4 1-16	3/4 1-16
3/4 1-17	3/4 1-17
3/4 3-5a	3/4 3-5a
3/4 3-17	3/4 3-17
3/4 3-45	3/4 3-45
3/4 6-15	3/4 6-15
3/4 6-17	3/4 6-17
3/4 7-38	3/4 7-38
3/4 8-5	3/4 8-5
3/4 10-3	3/4 10-3
3/4 11-1	3/4 11-1
3/4 11-9	3/4 11-9
B 3/4 0-3	B 3/4 0-3
B 3/4 3-4	B 3/4 3-4
B 3/4 4-1	B 3/4 4-1
6-13	6-13

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.

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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥5.5% Δk/k.

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

ACTION:

With the SHUTDOWN MARGIN <5.5% Δk/k, 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 ≥5.5% Δk/k.

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.

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 $<5\% \ \Delta k/k$ 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.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume in each water source, and

3. Verifying the boric acid makeup tank(s) solution temperature is greater than 55°F.

b. At least once per 24 hours by verifying the RWT temperature.

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 Figure 3.1-1A 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.

ACTION STATEMENTS

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to both CEACs inoperable.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "OFF" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn, except as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in their group.

ACTION 6 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	$\leq$ 0.40 seconds*
3. Logarithmic Power Level - High	$\leq$ 0.40 seconds*
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Pressurizer Pressure - Low	$\leq$ 0.90 seconds
6. Containment Pressure - High	$\leq$ 1.59 seconds
7. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
8. Steam Generator Level - Low	$\leq$ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 2.58 seconds*
b. CEA Positions	$\leq$ 1.58 seconds**

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

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

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

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

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

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

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded.

APPLICABILITY: During releases via this pathway.

ACTION:

a. With the following gaseous effluent monitoring instrumentation channels alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel.

   1. Waste Gas Holdup System Noble Gas Activity Monitor (during periods of gaseous releases).

   2. Containment Purge and Ventilation System Noble Gas Activity Monitor (during periods of containment building PURGE).

b. With less than the minimum number of monitoring instrumentation channels OPERABLE, take the action shown in Table 3.3-12.

c. Return the instruments to OPERABLE status within 30 days or, in lieu of any other report, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected.

d. The provisions of Specifications 3.0.3 and 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
1. Waste Gas Holdup System				
a. Noble Gas Activity Monitor (provides alarm and automatic termination of release)	1	*	Radioactivity	25
b. Effluent System Flow Monitor	1	*	System Flow	26
2. Containment Purge and Ventilation System				
a. Noble Gas Activity Monitor	1	*	Radioactivity	27,29
b. Iodine Sampler Cartridge	1	*	Verify Presence of Cartridge	28
c. Particulate Sampler Filter	1	*	Verify Presence of Filter	28
d. Effluent System Flow Monitor	1	*	System Flow	26
e. Sampler Flow Monitor	1	*	Sampler Flow	26

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling group shall be demonstrated OPERABLE:

a. At least once per 14 days by:

1. Verifying that service water flow rate to the group of cooling units is $\geq$ 1250 gpm and that each unit in the group has an operable fan; or that one unit in the group has a service water flow rate of $\geq$ 1250 gpm and an operable fan.

2. Addition of a biocide to the service water during the surveillance in 4.6.2.3.a.1 above, whenever service water temperature is between 60°F and 80°F.

b. At least once per 31 days by:

1. Starting (unless already operating) each operational cooling unit from the control room.

2. Verifying that each operational cooling unit operates for at least 15 minutes.

c. At least once per 18 months by verifying that each cooling unit starts automatically on a CCAS test signal.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.1.4 Prior to exceeding conditions which require establishment of reactor building integrity per Specification 3.6.1.1, the leak rate of the containment purge supply and exhaust isolation valves listed in Table 3.6-1 Part B shall be verified to be within acceptable limits per Specification 4.6.1.2, unless the test has been successfully completed within the last three months.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES.

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
A. CONTAINMENT ISOLATION			
2P7	2CV-5852-2#	"A" S/G Sample Isolation (outside)	≤ 20
	2CV-5859-2#	"B" S/G Sample Isolation (outside)	≤ 20
2P8	2SV-5833-1	RCS & Pressurizer Sample Isolation (inside)	≤ 20
	2SV-5843-2	RCS & Pressurizer Sample Isolation (outside)	≤ 20
2P9	2CV-6207-2	H.P. Nitrogen to SI Tanks (outside)	≤ 20
2P14	2CV-4821-1	CVCS L/D Isolation (inside)	≤ 35
	2CV-4823-2	CVCS L/D Isolation (outside)	≤ 20
2P18	2CV-4846-1	RCP Seal Return Isolation (inside)	≤ 25
	2CV-4847-2	RCP Seal Return Isolation (outside)	≤ 20
2P31	2CV-2401-1	Containment Vent Header (inside)	≤ 20
	2CV-2400-2	Containment Vent Header (outside)	≤ 20
2P37	2SV-5878-1	Quench Tank Liquid Sample (inside)	≤ 20
	2SV-5871-2	Quench Tank Liquid Sample (outside)	≤ 20
	2SV-5876-2	SI Tanks Sample Isolation (outside)	≤ 20
2P39	2CV-4690-2	Quench Tank Makeup & Demin Water Supply Isolation (outside)	≤ 20
2P40	2CV-3200-2	Fire Water Isolation (outside)	≤ 20
2P41	2CV-6213-2	L.P. Nitrogen Supply Isolation (outside)	≤ 20
2P51	2CV-3852-1	Chilled Water Supply Isolation (outside)	≤ 20
2P52	2CV-5236-1	CCW to RCP Coolers Isolation (outside)	≤ 20
2P59	2CV-3850-2	Chilled Water Return Isolation (inside)	≤ 20
	2CV-3851-1	Chilled Water Return Isolation (outside)	≤ 20
2P60	2CV-5254-2	CCW from RCP Coolers Isolation (inside)	≤ 20
	2CV-5255-1	CCW from RCP Coolers Isolation (outside)	≤ 20

PLANT SYSTEMS

3/4.7.11 FIRE BARRIERS

LIMITING CONDITION FOR OPERATION

DELETED

PLANT SYSTEMS

3/4.7.12 SPENT FUEL POOL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.7.12 The structural integrity of the spent fuel pool shall be maintained in accordance with Specification 4.7.12.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the structural integrity of the spent fuel pool not conforming to the above requirements, in lieu of any other report, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days of a determination of such non-conformity.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 Inspection Frequencies - The structural integrity of the spent fuel pool shall be determined per the acceptance criteria of Specification 4.7.12.2 at the following frequencies:

a. At least once per 92 days after the pool is filled with water. If no abnormal degradation or other indications of structural distress are detected during five consecutive inspections, the inspection interval may be extended to at least once per 5 years.

b. Within 24 hours following any seismic event which actuates or should have actuated the seismic monitoring instrumentation of Specification 3.3.3.3.

4.7.12.2 Acceptance Criteria - The structural integrity of the spent fuel pool shall be determined by a visual inspection of at least the interior and exterior surfaces of the pool, the struts in the tilt pit, the surfaces of the separation walls, and the structural slabs adjoining the pool walls. This visual inspection shall verify no changes in the concrete crack patterns, no abnormal degradation or other signs of structural distress (i.e, cracks, bulges, out of plumbness, leakage, discolorations, efflorescence, etc.).

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. A day fuel tank containing a minimum volume of 280 gallons of fuel (equivalent to 50% of total tank volume),

   2. A fuel storage system containing a minimum volume of 22,500 gallons of fuel (equivalent to 100% of total tank volume), and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENT

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for Requirement 4.8.1.1.2a.5.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

4160	volt Emergency Bus # 2A3
4160	volt Emergency Bus # 2A4
480	volt Emergency Bus # 2B5
480	volt Emergency Bus # 2B6
120	volt A.C. Vital Bus # 2RS1
120	volt A.C. Vital Bus # 2RS2
120	volt A.C. Vital Bus # 2RS3
120	volt A.C. Vital Bus # 2RS4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment and indicated power availability.

SPECIAL TEST EXCEPTIONS

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Table 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at ≤ 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site in liquid effluents to the discharge canal shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration released shall be limited to 2×10^{-4} μCi/ml.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released exceeding the above limits, immediately initiate actions to restore concentrations to within the above limits. Provide notification to the Commission within 24 hours and in lieu of any other report, submit a Special Report pursuant to Specification 6.9.2.h within 30 days.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analyses program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSES PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analyses Frequency	Type of Activity Analyses	Lower Limit of Detection (LLD) (uCi/ml) (a)
A. Batch Waste Release (d)	P Each Batch	P Each Batch	γ isotopic (e)	5×10^{-7} (b)
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emmitters)	1×10^{-5}
	P Each Batch	M Composite (c)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite (c)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the lower limit of detection as defined above (as picocurie per unit mass or volume).

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficiency (as counts per transformation).

TABLE 4.11.2 (Continued)

TABLE NOTATION

a. The Lower Limit of Detection (LLD) is defined in Table Notation a. of Table 4.11-1 of Specification 3.11.1.1.

b. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Release Report.

c. Tritium grab samples shall be taken from the Reactor Building ventilation exhaust at least once per 24 hours when the refueling canal is flooded.

d. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel area, whenever spent fuel is in the spent fuel pool.

e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

f. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from the sampler).

g. For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportional to the magnitude of the gamma yield (i.e., 1 x E-4/I, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the MPC value specified in 10 CFR 20, Appendix B, Table II, Column I.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The dose due to noble gases released in gaseous effluents from ANO-2 to UNRESTRICTED AREAS (See Figure 5.1-3) shall be:

a During any calendar quarter, less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year, less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report, submit a Special Report pursuant to Specification 6.9.2.h within 30 days

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions for noble gases for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirements within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2 was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24 hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24 hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24 hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated. If the ACTION requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not allow a grace period before a device, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

1. The instrument indicates measured levels above the alarm/trip setpoint.

2. Power to the detector is lost.

3. The instrument indicates a downscale failure.

For the containment purge and the waste gas holdup system noble gas activity monitors, the CHANNEL FUNCTIONAL TEST also demonstrates the automatic isolation of the release pathway occurs if the instrument indicates above the trip setpoint.

The initial CHANNEL CALIBRATION is performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards permit calibrating the system over its intended range of energy and measurment range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration are used.

3.4.3.3.10 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

For the radioactive liquid effluent instrumentation surveillance requirements, the channel test demonstrates that automatic isolation of this pathway and control room alarm annunciation occur if the instrument indicates measured levels above the trip setpoint. The channel test demonstrates that alarm annunciation occurs if any of the following conditions exist:

1. Power to the detector is lost.

2. The instrument indicates a downscale failure.

3. Instrument controls are not set in the operate mode.

The initial CHANNEL CALIBRATION is performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above the limits specified by Specification 3.2.4 during all normal operations and anticipated transients.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 420,000 lbs. per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Vice President, Operations ANO and the SRC shall be notified within 24 hours.

c. The Nuclear Regulatory Commission shall be notified pursuant to 10CFR50.72 and a report submitted pursuant to the requirements of 10CFR50.36 and Specification 6.6.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants. These procedures should include provisions to assure that sufficient margin is maintained in CPC Type I addressable constants to avoid excessive operator interaction with the CPCs during reactor operation.

NOTE: Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P that has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

h. New and spent fuel storage.

i. ODCM and PCP implementation.

j. Postaccident sampling (includes sampling of reactor coolant, radioactive iodines and particulates in plant gaseous effluent, and the containment atmosphere).

6.8.2 Each procedure of 6.8.1 above, and changes in intent thereto, shall be reviewed by the PSC and approved by the General Manager, Plant Operations, Plant Manager, ANO-2 or responsible Major Department Head prior to implementation and reviewed periodically as set forth in administrative procedures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.149 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 May 7, 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 correct typographical errors that were introduced in the original TSs and in subsequent amendments.

2.0 EVALUATION

The proposed amendment would correct typographical errors and provide clarification (i.e., renumbering, placing paragraph designators in parenthesis, correcting misspellings, rephrasing, correcting references). The staff has reviewed the changes and found that the changes do not affect the intent of any specification. Also, the proposed 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.

In addition, the staff suggested to the licensee that the proposed revision of TS 4.6.2.3 is still unclear, since it may imply that both groups of cooling units shall be demonstrated operable to the same criteria. However, there are two different criteria for demonstrating operability and it is possible that one group is demonstrated operable to one criteria, while at the same time the other group is demonstrated operable to the other criteria. Accordingly, the licensee requested (per telecon) that the staff change the words "each group" to "the group". The staff agrees that the words "the group" provides the desired flexibility and has made this change to the TSs as requested.

Based on the above, the staff finds that the proposed amendment is acceptable.

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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 34076). 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: T. Alexion, PDIV-1

Date: July 26, 1993