March 31, 1992

DISTRIBUTION:

Docket File	NRC/Local PDR	PD4-1 r/f
SPeterson(2)	MVirgilio	JLarkins
PNoonan(2)	CGrimes	TAlexion(2)
TWesterman	DHagan	OGC
GHill(8)	Wanda Jones	PD4-1 p/f
ACRS (10)	OPA	OC/LFMB
SRuffin	JNorberg	

Docket Nos. 50-313
and 50-368

Mr. Neil S. Carns
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Carns:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 158 AND 132 TO FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6 - ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 (TAC NOS. M81999 AND M82000)

The Commission has issued the enclosed Amendment Nos. 158 and 132 to Facility Operating License Nos. DPR-51 and NPF-6 for the Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 15, 1991, as supplemented March 13, 1992.

The amendments revise the Operating Licenses (OLs) and TSs for ANO-1&2. The changes to the OLs are to add the NRC's standard OL condition for Fire Protection. The changes to the TSs will relocate the Fire Protection requirements from the TSs to the respective Safety Analysis Reports (SARs) verbatim. These changes were prepared in acordance with Generic Letters 86-10 and 88-12.

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 Sheri R. Peterson for:
Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV and V
Office of Nuclear Reactor Regulation

Original signed by:
Sheri R. Peterson, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 158 to DPR-51
2. Amendment No. 132 to NPF-6
3. Safety Evaluation

NRC FILE CENTER COPY

cc w/enclosures:
See next page

* See previous concurrence

OFC	:LA/PD4-1	:I/PD4-1 *	:PM/PD4-1 *	:PM/PD4-1	:OGC*	:D:PD4-1 *
NAME	:PNoonan	:SRuffin/vmw	:TAlexion	:SPeterson	:JHull	:JLarkins
DATE	:3/30/92	: 2 / 10/92	: 2 / 10/92	:3/30/92	: 2/26/92	: 3 / 4 /92

OFFICIAL RECORD COPY Document Name: ANO1/2 AMD M81999/2000

9204090050 920331
PDR ADOCK 05000313
P PDR

DF01
1/1

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

March 31, 1992

Docket Nos. 50-313
and 50-368

Mr. Neil S. Carns
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Carns:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 158 AND 132 TO FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6 - ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 (TAC NOS. M81999 AND M82000)

The Commission has issued the enclosed Amendment Nos. 158 and 132 to Facility Operating License Nos. DPR-51 and NPF-6 for the Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 15, 1991, as supplemented March 13, 1992.

The amendments revise the Operating Licenses (OLs) and TSs for ANO-1&2. The changes to the OLs are to add the NRC's standard OL condition for Fire Protection. The changes to the TSs will relocate the Fire Protection requirements from the TSs to the respective Safety Analysis Reports (SARs) verbatim. These changes were prepared in acordance with Generic Letters 86-10 and 88-12.

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,

Sheri R. Peterson
for Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV and V
Office of Nuclear Reactor Regulation

Sheri R. Peterson
Sheri R. Peterson, Project Manager
Proejct Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 158 to DPR-51
2. Amendment No. 132 to NPF-6
3. Safety Evaluation

cc w/enclosures:
See next page

cc:

Mr. Donald C. Hintz
Executive Vice President
and Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, Mississippi 39286

Mr. Jerry Yelverton
General Manager, Plant Operations
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Mr. Nicholas S. Reynolds
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

Mr. Robert B. Borsum
Licensing Representative
B&W Nuclear Technologies
1700 Rockville Pike, Suite 525
Rockville, Maryland 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
1 Nuclear Plant Road
Russellville, Arkansas 72801

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Honorable Joe W. Phillips
County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

Ms. Greta Dicus, Director
Division of Radiation Control and
Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, Arkansas 72205-3867

Mr. John R. McGaha
Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, Mississippi 39286

Mr. Robert B. McGehee
Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, Mississippi 39205

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering Nuclear Power
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. James J. Fisicaro
Director, Licensing
Entergy Operations, Inc.
Route 3, Box 137G
Russellville, Arkansas 72801

Admiral Kinnaird R. McKee, USN (Ret)
214 South Morris Street
Oxford, Maryland 21654

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
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 October 15, 1991, as supplemented March 13, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

9204090054 920331
PDR ADOCK 05000313
P PDR

2. Accordingly, the license is amended, as indicated in the attachment to this license amendment, and Paragraph 2.C.(8) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   (8) FIRE PROTECTION

   EOI shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992 , subject to the following provision:

   The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. In addition, the license is also 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 158, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. The license amendment is effective as of 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

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

Attachment:
Changes to the License
and Technical Specifications

Date of Issuance: March 31, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 158

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Revise the following pages of the License and 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
License page 5	License page 5
53d	53d
53e	53e
66m	66m
66n	66n
66o	66o
66p	66p
66q	66q
110p	110p
110q	110q
110r	110r
110s	110s
110t	110t
110u	110u
110v	110v
110w	110w
118	118
122	122
146b	146b

(7) Secondary Water Chemistry Monitoring

A secondary water chemistry monitoring program shall be implemented to minimize steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedures for the recording and management of data;

5. Procedures defining corrective actions for off-control point chemistry conditions; and

6. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate a corrective action.

(8) FIRE PROTECTION

EOI shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992 , subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license is effective as of the date of issuance and shall expire at midnight, May 20, 2014.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by:
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A and B - Technical
Specifications

Date of Issuance: May 21, 1974

3.5.5 Fire Detection Instrumentation

DELETED

Table 3.5-5

SAFETY-RELATED AREAS PROTECTED BY HEAT/SMOKE DETECTORS

DELETED

3.17 Fire Suppression Water System

DELETED

3.18 FIRE SUPPRESSION SPRINKLER SYSTEM

DELETED

3.19 CONTROL ROOM AND AUXILIARY CONTROL ROOM HALON SYSTEMS

DELETED

3.20 FIRE HOSE STATIONS

DELETED

3.21 FIRE BARRIERS

DELETED

4.19 FIRE DETECTION INSTRUMENTATION

DELETED

4.20 Fire Suppression Water System

DELETED

DELETED

Bases

DELETED

4.21 SPRINKLER SYSTEMS

DELETED

4.22 Control Room and Auxiliary Control Room Halon Systems

DELETED

4.23 Fire Hose Stations

DELETED

4.24 Fire Barriers

DELETED

Table 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM SHIFT CREW COMPOSITION #

UNIT 1

LICENSE CATEGORY	ABOVE COLD SHUTDOWN	COLD AND REFUELING SHUTDOWNS
SOL	2	1*
OL	2	1
NON-LICENSED	3	1
SHIFT TECHNICAL ADVISOR	1	None required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

Additional Requirements:

1. At least one licensed Operator shall be in the control room when fuel is in the reactor.

2. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

3. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

4. All refueling operations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

5. DELETED

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager, ANO-1, General Manager, Plant Operations or the Safety Review Committee.

i. Review of the Plant Security Plan and implementing procedures and submittal of recommended changes to the General Manager, Plant Operations and the Safety Review Committee.

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the General Manager, Plant Operations and the Safety Review Committee.

k. Review of changes to the Offsite Dose Calculation Manual and the Process Control Program.

l. Review of changes to the Fire Protection Program and implementing procedures and submittal of recommended changes to the General Manager, Plant Operations and Safety Review Committee.

AUTHORITY

6.5.1.7. The Plant Safety Committee shall:

a. Recommend in writing their approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President, Operations ANO and the Safety Review Committee of disagreement between the PSC and the Plant Manager, ANO-1 or the General Manager, Plant Operations; however, the General Manager, Plant Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Plant Safety Committee shall maintain written minutes of each PSC meeting that, at a minimum, document the results of all PSC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Plant Manager, ANO-1, General Manager, Plant Operations and Chairman of the Safety Review Committee.

6.5.2 Safety Review Committee (SRC)

FUNCTION

6.5.2.1 The Safety Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry

h. Inoperable Fire Detection Instrumentation

i. Inoperable Fire Suppression Systems

j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H.

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1

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. 132
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 October 15, 1991, as supplemented March 13, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended, as indicated in the attachment to this license amendment, and Paragraph 2.C.(3)(b) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (b) Fire Protection

   EOI shall implement and maintain in effect all provisions of the approved fire protection program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992 , subject to the following provision:

   The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. In addition, the license is also 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. 132 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. The license amendment is effective as of 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

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

Attachment:
Changes to the License
and Technical Specifications

Date of Issuance: March 31, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the following pages of the License and 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
License pages 4 and 5	License pages 4 and 5
3/4 3-43	3/4 3-43
3/4 3-44	3/4 3-44
3/4 7-29	3/4 7-29
3/4 7-30	3/4 7-30
3/4 7-31	3/4 7-31
3/4 7-32	3/4 7-32
3/4 7-33	3/4 7-33
3/4 7-34	3/4 7-34
3/4 7-35	3/4 7-35
3/4 7-36	3/4 7-36
3/4 7-37	3/4 7-37
B 3/4 3-3	B 3/4 3-3
B 3/4 7-6	B 3/4 7-6
B 3/4 7-7	B 3/4 7-7
6-2	6-2
6-5	6-5
6-7	6-7
6-19	6-19

(b) Fire Protection

EOI shall implement and maintain in effect all provisions of the approved fire protection program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992 , subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(c) Less Than Four Reactor Coolant Pump Operation

EOI shall not operate the reactor in operational Modes 1 and 2 with fewer than four reactor coolant pumps in operation, except as allowed by Special Test Exception 3.10.3 of the facility Technical Specifications.

2.C.(3)(d) Deleted per Amendment 24, 6/19/81.

(e) AP&L shall complete the following modifications by the indicated dates in accordance with the staff's findings as set forth in the fire protection evaluation report, NUREG-0223 "Fire Protection Safety Evaluation Report."

Implementation Dates for Proposed Modifications

Applicable Section of NUREG-0223		Date
3.1	Portable Radio Communication Equipment	March 31, 1979
3.2	Separation of Power Cables in Manholes	*
3.3	Protection from Water Spray	*
3.4	Protection of Redundant Cables in the MCC Room (2096-M)	December 30, 1978
3.5	Protection of Redundant Cables in the Hallway - Elevation 372 (2109-U)	*, **
3.6	Protection of Redundant Cables in the Cable Spreading Room (2098-L)	*
3.7	Protection of Redundant Cables in the Switchgear Room (2100-Z)	*
3.8	Protection of Redundant Cables in the Electrical Equipment Room (2091-BB)	September 30, 1978

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

DELETED

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

DELETED

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

DELETED

ACTION (Continued)

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

DELETED

SURVEILLANCE REQUIREMENTS

DELETED

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

DELETED

TABLE 3.7-7

FIRE HOSE STATIONS

DELETED

3/4.7.11 FIRE BARRIERS

LIMITING CONDITION FOR OPERATION

DELETED

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 internal 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.).

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations."

The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. A minimum of two operable level sensors in the upper plenum region and one operable level sensor in the dome region are required for RVLMS channel operability. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

DELETED

following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

3/4.7.8 SHOCK SUPPRESSORS (SNUBBERS)

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies based upon the number of INOPERABLE snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories and the previous inspection interval as specified in NRC Generic Letter 90-09, "Alternative Requirements For Snubber Visual Inspection Intervals and Corrective Actions". Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the result of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation is performed to determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

If a review and evaluation of an INOPERABLE snubber is performed and documented to justify continued operation and provided that all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in areas which have high radiation fields during shutdown or in especially difficult to remove locations may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

DELETED

3/4.7.11 FIRE BARRIERS

DELETED

3/4.7.12 SPENT FUEL POOL STRUCTURAL INTEGRITY

The reinforcing steel in the walls of the spent fuel pool was erroneously terminated into the front face instead of the rear face of the adjoining walls during construction of the spent fuel pool. Therefore, the specified structural integrity inspections of the spent fuel pool are required to be performed to ensure that the pool remains safe for use and that it will adequately resist the imposed loadings. If no abnormal degradation is observed during the first five inspections, the inspection interval for subsequent routine inspections may be extended to at least once per 18 months or longer if justified by observed performance of the pool.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. DELETED

g. Administrative control shall be established to limit the amount of overtime worked by plant staff performing safety-related functions. These administrative controls shall be in accordance with the guidance provided by the NRC Policy Statement on working hours (Generic Letter No. 82-12).

h. The Manager, Operations and Shift Supervisor shall hold a senior reactor operator license.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager, Training and Emergency Planning and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 DELETED

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY COMMITTEE (PSC)

FUNCTION

6.5.1.1 The Plant Safety Committee shall function to advise the General Manager, Plant Operations and Plant Manager, ANO-2 on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Committee shall be composed of eight members of ANO onsite management organization (except as discussed under 6.5.1.3) at the superintendent level or above. The PSC Chairman shall ensure that adequate expertise is present during meetings to evaluate material before the PSC.

In addition, the General Manager, Plant Operations shall designate in writing a PSC Chairman and at least one Alternate Chairman.

6.5.1.3 If Core Protection Calculator (CPC) Software is being reviewed a nuclear software expert shall be present as a voting member. If one of the members of the Plant Safety Committee meets the qualification requirements for this position, the requirement to have this member is satisfied. This membership may be filled by two appropriately qualified individuals who shall ballot with a single combined vote. Generic qualifications for this membership shall be as follows:

One Individual

The Nuclear Software Expert shall have as a minimum a Bachelor's degree in Science or Engineering, Nuclear preferred (in accordance with ANSI N18.1). In addition, he shall have a minimum of four years of technical experience, of which a minimum of two years shall be in Nuclear Engineering and a minimum shall be in Software Engineering. (Software Engineering is that branch of science and technology which deals with the design and use of software. Software Engineering is a discipline directed to the production and modification of computer programs that are correct, efficient, flexible, maintainable, and understandable, in reasonable time spans, and at reasonable costs.) The two years of technical experience in Software Engineering may be general software experience not necessarily related to the software of the Core Protection Calculator System. One of these two years of experience shall be with certified computer programs.

Two Individuals

One of the individuals shall meet the requirements of the Nuclear Engineering portion of the above. The second individual shall have a Bachelor of Science degree (digital computer speciality) and meet the Software Engineering requirements of the above.

The membership (the Nuclear Software Expert or the Digital Computer Specialist) shall be knowledgeable of the Core Protection Calculator System with regard to:

a. The software modules, their interactions with each other and with the data base.

b. The relationship between operator's module inputs and the trip variables.

c. The relationship between sensor input signals and the trip variable.

d. The design basis of the Core Protection Calculator System.

e. The approved software change procedure and documentation requirements of a software change.

f. The security of the computer memory and access procedures to the memory.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses, and reports thereon as requested by the Plant Manager, ANO-2, General Manager, Plant Operations or the Safety Review Committee.

i. Review of the Plant Security Plan and implementing procedures and submittal of recommended changes to the General Manager, Plant Operations and the Safety Review Committee.

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the General Manager, Plant Operations and Safety Review Committee.

k. Review of changes to the Offsite Dose Calculation Manual and Process Control Program.

l. Review of changes to the Fire Protection Program and implementing, procedures and submittal of recommended changes to the General Manager, Plant Operations and Safety Review Committee.

AUTHORITY

6.5.1.8 The Plant Safety Committee shall:

a. Recommend in writing their approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President, Operations ANO and the Safety Review Committee of disagreement between the PSC and the Plant Manager, ANO-2 or the General Manager, Plant Operations; however, the General Manager, Plant Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.9 The Plant Safety Committee shall maintain written minutes of each PSC meeting that, at a minimum, document the results of all PSC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Plant Manager, ANO-2, General Manager, Plant Operations and Chairman of the Safety Review Committee.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The Safety Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The Safety Review Committee shall be composed of a Chairman and eight to twelve members which collectively have the experience and competence required by ANSI/ANS-3.1-1981 to review problems in the areas specified in Section 6.5.2.1, a-h.

The Vice President, Operations ANO shall designate, in writing, the Chairman and all SRC members.

The Chairman shall designate, in writing, the alternate Chairman in the absence of the SRC Chairman.

SPECIAL REPORTS

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

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Inoperable Fire Detection Instrumentation

f. Inoperable Fire Suppression Systems

g. Deleted.

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

This report shall include the following:

1) Description of occurrence.

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

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

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

5) Summary of consequence(s) of occurrence.

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

i. Inoperable Containment Radiation Monitors, Specification 3.3.3.1.

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

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

1. Radiological Environmental Monitoring Sample Analysis, Specification 3.12.1.

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

   1. Description of the occurrence.

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

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

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

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

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

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

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

---

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 158 AND 132 TO

FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 AND 2

DOCKET NOS. 50-313 AND 50-368

1.0 INTRODUCTION

By letter dated October 15, 1991, as supplemented March 13, 1992, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Arkansas Nuclear One, Units 1 & 2 (ANO 1&2), Operating License (OL) and Technical Specifications (TSs). The requested changes would add the NRC's standard Operating License Condition (OLC) noted in Generic Letter 86-10. In addition, the proposed changes would move requirements for fire detection systems, fire suppression systems, fire barriers, and fire brigade staffing requirements, as recommended by Generic Letter 88-12 from the TSs, to the respective Safety Analysis Reports (SARs) verbatim. The proposed OLC would reference the NRC-approved Fire Protection Program in the SAR and allow changes to this program provided the changes would not adversely affect fire protection effectiveness. The proposed changes would also modify the administrative control requirements of the TSs to add requirements for the Fire Protection Program that are similar to requirements for other programs implemented by a license condition. The new requirement would be added to TS 6.0, "Administrative Controls," requiring the Plant Safety Committee (PSC) to review changes to the approved Fire Protection Program. Guidance on these proposed changes to TSs was provided to all power reactor licensees and applicants by Generic Letter 88-12, dated August 2, 1988.

The additional information contained in the supplemental letter dated March 13, 1992, was clarifying in nature and, thus, within the scope of the initial Federal Register notice and did not affect the staff's proposed no significant hazards consideration determination.

2.0 EVALUATION

The proposed changes to the fire protection OLC and TSs are consistent with the guidance in NRC Generic Letter 86-10, "Implementation of Fire Protection Requirements," and NRC Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications." These generic letters recommended that the appropriate fire protection TSs be removed from the TSs and incorporated into the SAR for the facility. Although a comprehensive Fire Protection Program is essential to plant safety, the basis for the recommendation is that many details of this program that are currently addressed in the TSs can be modified without affecting nuclear safety. These details, that are presently included in TSs and which are removed by this amendment, do not constitute performance requirements necessary to ensure safe operation of the facility

9204090056 920331
PDR ADOCK 05000313
P PDR

and, therefore, do not warrant being included in TSs. At the same time, suitable administrative controls ensure that there will be careful review and analysis by competent individuals of any changes in the Fire Protection Program to ensure that nuclear safety is not adversely affected. These controls include: (1) the TSs administrative controls that are applicable to the Fire Protection Program; (2) the licensee condition on implementation of, and subsequent changes to, the Fire Protection Program; and (3) the 10 CFR 50.59 criteria for evaluating changes to the Fire Protection Program as described in the SAR. The generic letters also recommended that the specific revision of the SAR which incorporated the Fire Protection Program and the staff's specific safety evaluation report which approved the Fire Protection Program be referenced in a new OLC.

The specific details relating to fire protection requirements removed from the TSs by this amendment include those specifications for fire detection systems, fire suppression systems, fire barriers, and fire brigade staffing requirements. The administrative control requirements already include Fire Protection Program implementation as an element for which written procedures must be established, implemented, and maintained. The audit responsibilities of the PSC were expanded to include the review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the General Manager, Plant Manager, Plant Operations and Safety Review Committee.

The proposed fire protection OLC 2.C.(8) and 2.C.(3)(b) for Unit 1 and Unit 2, respectively, states:

> "EOI shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992 , subject to the following provision:
>
> > The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of fire."

The TS changes proposed by the licensee are in accordance with guidance provided by Generic Letter 88-12, as addressed in the items below:

<u>Unit 1</u>

Specification 3.5.5, Fire Detection Instrumentation, its associated Surveillance Requirements, and Bases were removed.

Table 3.5-5, Safety-Related Areas Protected By Heat/Smoke Detectors, was removed.

Specification 3.17, Fire Suppression Water System, its associated Surveillance Requirements, and Bases were removed.

Specification 3.18, Fire Suppression Sprinkler System, its associated Surveillance Requirements, and Bases were removed.

Specification 3.19, Control Room and Auxillary Control Room Halon Systems, its associated Surveillance Requirements, and Bases were removed.

Specification 3.20, Fire Hose Stations, its associated Surveillance Requirements, and Bases were removed.

Specification 3.21, Fire Barriers, its associated Surveillance Requirements, and Bases were removed.

Administrative Controls Specification 6.5.1.2, Composition, Additional Requirements, Item 5, regarding fire brigade staffing, was removed.

Administrative Controls Specification 6.5.1.6, Responsibilities, Item 1, regarding review of changes to the Fire Protection Program, was added.

Unit 2

Specification 3.3.3.8, Fire Detection Instrumentation, its associated Surveillance Requirements, and Bases were removed.

Table 3.3-11, Fire Detection Instruments, was removed.

Specification 3/4.7.10, Fire Suppression Water System, Fire Suppression Spray and/or Sprinkler System, Fire Hose Stations, its associated Surveillance Requirements, and Bases were removed.

Table 3.3-7, Fire Hose Stations, was removed.

Specification 3/4.7.11, Fire Barriers, its associated Surveillance Requirements, and Bases, were removed.

Administrative Controls Specification 6.2.2, Facility Staff, Item f, regarding fire brigade staffing, was removed.

Administrative Controls Specification 6.4.2, regarding a training program for fire brigade staff, was removed.

Administrative Controls Specification 6.5.1.7, Responsibilities, Item 1, regarding review of changes to the Fire Protection Program, was added.

By letter dated October 25, 1991, the licensee confirmed that the NRC-approved Fire Protection Program has been incorporated into the SAR as required by Generic Letter 86-10. The licensee also confirmed that the operational conditions, remedial actions, and test requirements associated with the removed fire protection TSs have been included in the Fire Protection Program and incorporated into the SAR. This is in accordance with the guidance of Generic Letter 88-12.

On the basis of its review of the above items, the staff concludes that the licensee has met the guidance of Generic Letter 88-12. Therefore, the staff finds the proposed changes 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 comment.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 in surveillance requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 60116). Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Prinicpal Contributors: Steve Ruffin
Thomas Dunning
Thomas Alexion
Sheri Peterson

Date: March 31, 1992