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.

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) <u>FIRE PROTECTION</u>

   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) <u>Technical Specifications</u>

   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 <u>Control Room and Auxiliary Control Room Halon Systems</u>

DELETED

4.23 Fire Hose Stations

DELETED

4.24 <u>Fire Barriers</u>

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

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

DELETED

DELETED

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

DELETED

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

DELETED

E' ' DETECTION INSTRUMENTATION

I TING CONDITION FOR OPERATION

DELETED

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

DELETED

PLANT SYSTEMS

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

DELETED

PLANT SYSTEMS

ACTION (Continued)

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

DELETED

SURVEILLANCE REQUIREMENTS

DELETED

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

DELETED

TABLE 3.7-7

FIRE HOSE STATIONS

DELETED

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 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 t' control room to 5 rem or less whole body, or its equivalent. This limita is consistent with the requirements of General Design Criteria 19 o pendix "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. A retraining and replacement training program for the unit staff shall be ...tained 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 Ope...ions 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.

l. Radiological Environmental Monitoring Sample Analysis, Specification 3.12.1.

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

1. Description of the occurrence.
2. Identify the cause(s) of exceeding the limit(s).
3. Explain corrective action(s) taken to mitigate occurrence.
4. Define action(s) taken to prevent recurrence.
5. Summary of the consequence(s) of occurrence.

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

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.