ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

UNIT 1:

Technical Specification 6.5.2.2
SRC Composition

This change is requested due to reorganization.

Technical Specification Figure 6.2-2
Functional Organization for Plant Operation

This change is requested due to reorganization.

Technical Specification Figure 6.2-1
Management Organization Chart

This change is requested due to a position title change.

Technical Specification 6.5.1.7.1.c
Plant Safety Committee Authority

This change is requested due to a position title change.

Technical Specification 6.6.1
Reportable Occurrence Action

This change is requested due to a position title change.

UNIT 2:

Technical Specification 2.2.1 Bases

a. Linear Power Level - High

The Linear Power Level - High trip value was changed per Technical Specification Amendment No. 24 to 110.712%, but was not changed in the corresponding Technical Specification Bases.

b. Pressurizer Pressure-High

The Pressurizer Pressure - High trip value was changed per Technical Specification Amendment No. 24 to 2370.887 psia, but was not changed in the corresponding Technical Specification Bases.

c. Pressurizer Pressure-Low

The Pressurizer Pressure - Low trip value was changed per Technical Specification Amendment No. 24 to 1712.757 psia, but was not changed in the corresponding Technical Specification Bases.

Technical Specification 2.2.2 Bases
CPC Addressable Constants

AP&L Letter 2CANØ58113 was a prelimary submittal made as part of Technical Specification Amendment 24 describing the methodology for determination of CPC addressable constant values. The final submittal to the NRC is documented in Middle South Services Topical Report MSS-NA2-P," Arkansas Nuclear One - Unit 2 Core Protection Calculator Addressable Constant Determination Methodology", dated August, 1981.

Technical Specification 4.2.4.2
DNBR Margin Surveillance Requirements

Typographical error, Figure 3.2-3 is for COLSS in service and Figure 3.2-4 is for COLSS out of service.

Technical Specification 4.2.7
Axial Shape Index Surveillance Requirements

Typographical error, surveillance requirement for specification 3.2.7 should be numbered 4.2.7.

Technical Specification 4.2.8
Pressurizer Pressure Surveillance Requirements

Typographical error, surveillance requirement for specification 3.2.8 should be numbered 4.2.8.

Technical Specification Table 3.3-1
Action Statement 5

Typographical error, there is no 1.a) of Action 5 of Table 3.3-1. The CEA group 6 insertion allowed in No. 3 of Action 5 is permitted by 2.a) of Action 5.

Technical Specification 4.3.3.2
Incore Detectors Surveillance Requirements

Specification 4.3.3.2.a presently requires a Channel Check within 24 hours prior to the use of the incore detector system and a Channel Check once per 7 days thereafter when required for monitoring Azimuthal Power Tilt, Radial Peaking Factors, Local Power Density or DNB margin.

Monitoring of Local Power Density margin and DNB margin is a continuous operation of the COLSS. Azimuthal Power Tilt monitoring is also a continuous operation of the COLSS, but a monthly, independent verification is required using the incore detector system. The incore detector system is used to verify the Radial Peaking Factors used in the COLSS and CPCS.

Based on this established frequency of use, it is appropriate to perform a Channel Check of the incore detector system 24 hours prior to its use, if such a check has not been performed within 7 days prior to its intended use.

Technical Specification 4.4.9.2
Pressurizer Surveillance Requirements

The pressurizer spray/water differential temperature (ΔT) of > 200°F is not a limiting condition for operation per the requirements of Technical Specification 3.4.9.2. Pressurizer spray/water ΔT is administered per the component cyclic or transient limits of Technical Specification 5.7.1. The required surveillance of pressurizer spray should not be mentioned by the bases of Technical Specification 3.4.9.2 or Surveillance Requirement 4.4.9.2.

Technical Specification 6.9.2
Special Reports

Specification 3.4.8 requires a Special Report to the Commission pursuant to Specification 6.9.2 indicating the total cumulative operating time at a primary coolant specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six month period.

Technical Specification 6.10.2.k
Records Retention

Typographical error: PSC - Plant Safety Committee
SRC - Safety Review Committee

Technical Specification 6.5.2.2
SRC Composition

This change is requested due to reorganization.

Technical Specification Figure 6.2-2
Functional Organization for Plant Operation

This change is requested due to reorganization.

Technical Specification Figure 6.2-1
Management Organization Chart

This change is requested due to a position title change.

Technical Specification 6.5.1.7
Plant Safety Committee Authority

This change is requested due to a position title change.

Technical Specification 6.6.1.b
Reportable Occurrence Action

This change is requested due to a position title change.

Technical Specification 6.7.1.b
Safety Limit Violation

This change is requested due to a position title change.

Technical Specification 6.7.1.d
Safety Limit Violation

This change is requested due to a position title change.

ATTACHMENT 2

TECHNICAL SPECIFICATION CHANGES

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The SRC shall be composed of the:

Chairman: Director, Technical and Environmental Services
Member: Vice-President, Nuclear Operations
Member: Director, Generation Technology
Member: Manager, Quality Assurance
Member: Arkansas Nuclear One Plant Safety Committee Chairman
Member: Manager, Technical Analysis
Member: Arkansas Nuclear One Plant Analysis Superintendent
Member: Director, Generation Engineering
Member: Manager, Instrumentation and Controls Engineering
Member: Manager of System Nuclear Operations*

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

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC.

MEETING FREQUENCY

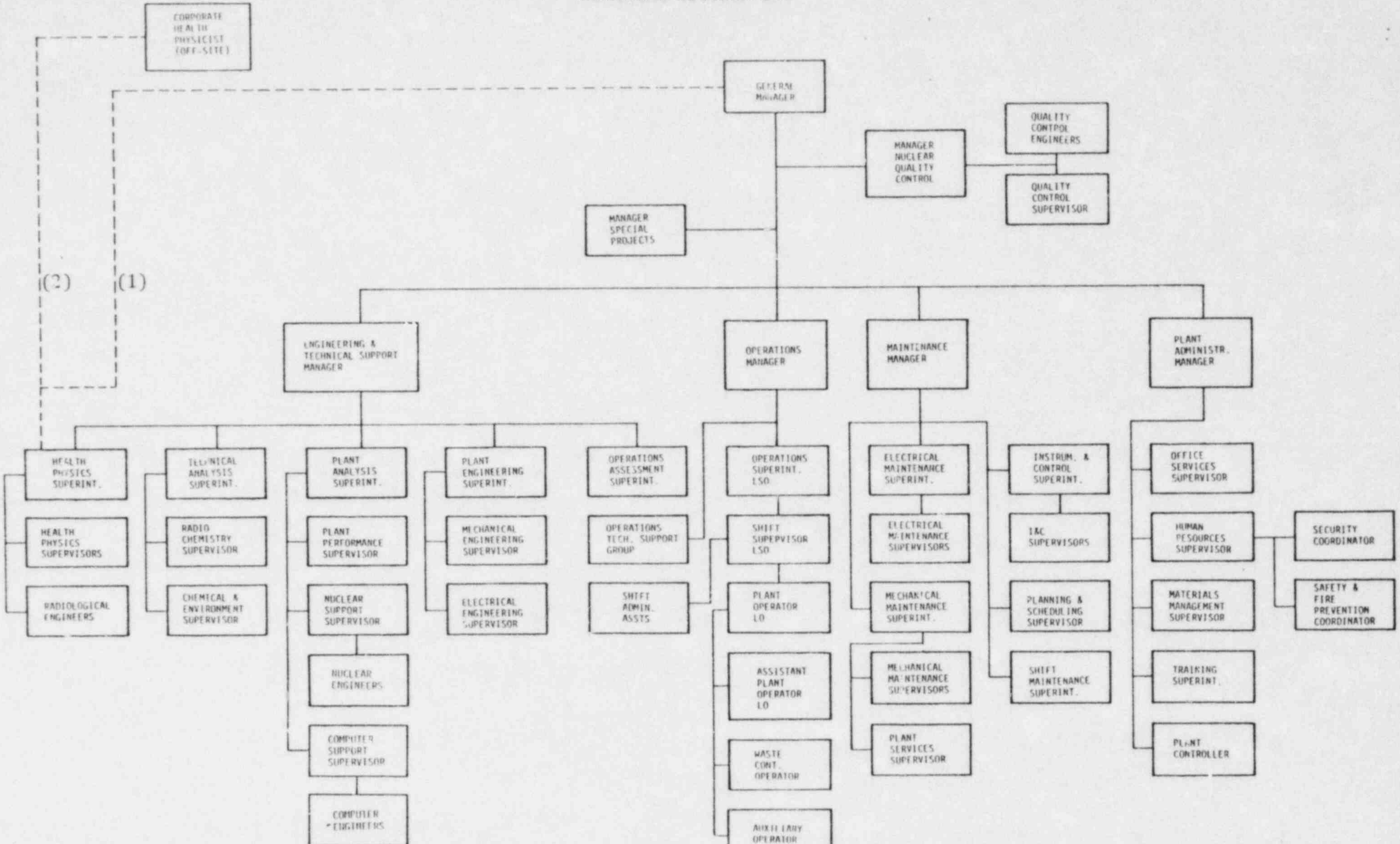
6.5.2.5 The SRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 The minimum quorum of the SRC necessary for the performance of the SRC review and audit functions of these technical specifications shall consist of the Chairman or his designated alternate and at least 4 SRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

*Middle South Services

ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS NUCLEAR ONE



CODE: LSO - SENIOR OPERATOR LICENSE REQUIRED

LS - OPERATOR LICENSE REQUIRED

*ONSITE RESPONSIBILITY FOR FIRE PROTECTION PROGRAM

(1) The Health Physics Superintendent reports to the Manager, Engineering and Technical Support in administrative matters and routine health physics concerns and he reports to the General Manager in matters of radiological health, safety and policy.

(2) The Health Physics Superintendent has direct interface with the Corporate Health Physicist in matters of radiological health and safety. The Corporate Health Physicist reports to the Manager, Nuclear Services. He will help formulate Corporate Health Physics Policy and ensure that it is properly implemented.

FIGURE 6.2-2 FUNCTIONAL ORGANIZATION FOR PLANT OPERATION

ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS NUCLEAR ONE

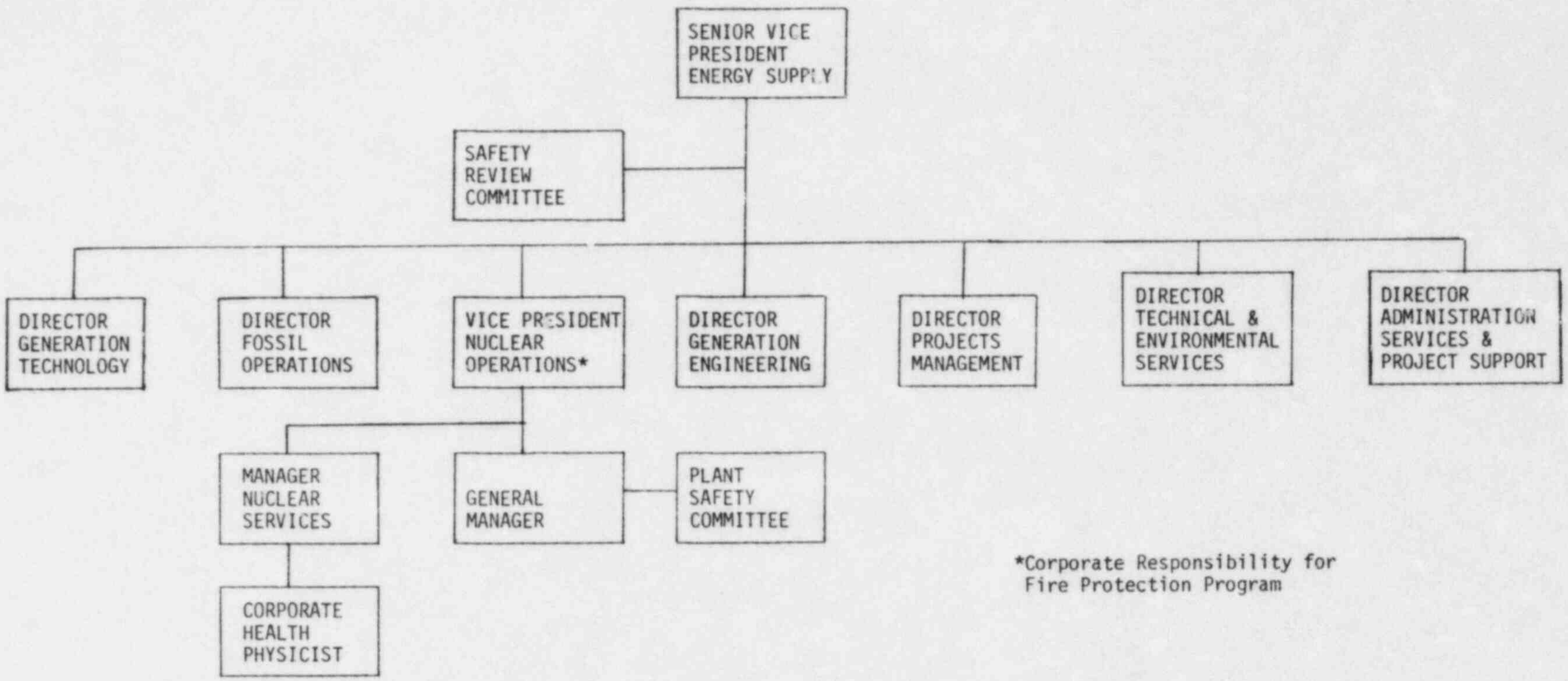


Figure 6.2-1 Management Organization Chart

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

h. Performance of special reviews, investigations and reports thereon as requested by the General Manager.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the General Manager.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the General Manager.

AUTHORITY

6.5.1.7.1. The Plant Safety Committee shall:

a. Recommend to the General Manager written 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, Nuclear Operations and the Safety Review Committee of disagreement between the PSC and the General Manager; however, the General Manager 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 General Manager 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

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for Reportable Occurrences:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.12.

b. Each Reportable Occurrence requiring 24-hour notification to the Commission shall be reviewed by the PSC and submitted to the SRC and the Vice-President, Nuclear Operations by the General Manager.

6.7 SAFETY LIMIT VIOLATION

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

BASES

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-44(A)P, "Core Protection Calculator Functional Description," January 7, 1977, Supplement 1P, May 13, 1977, Supplement 2P, May 19, 1977, Supplement 3P, September 2, 1977; CEN-45(A)P, "Control Element Assembly Calculator Functional Description," January 7, 1977; CEN-53(A)P, "ANO-2 Cycle 1 CPC and CEAC Data Base Document," May 20, 1977, Amendment 1P, June 28, 1977, Supplement 2P, September 2, 1977.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of $\leq$110.712% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of $\leq$ 0.819% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. The trip's setpoint is at $\leq$ 2370.887 psia which is below the nominal lift setting (2500 pisa) of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at $\geq$ 1712.757 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at $\leq$ 200 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at $\leq$ 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

a.	RCS Cold Leg Temperature-Low	$\geq$ 465°F
b.	RCS Cold Leg Temperature-High	$\leq$ 605°F
c.	Axial Shape Index-Positive	Not more positive than +0.6
d.	Axial Shape Index-Negative	Not more negative than -0.6
e.	Pressurizer Pressure-Low	$\geq$ 1750 psia
f.	Pressurizer Pressure-High	$\leq$ 2400 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

2.2.2 CPC Addressable Constants

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1.1 and 6.8.1) ensures that inadvertent misloading is unlikely. The methodology for determination of CPC addressable constant values is described in MSS-NA2-P, "Arkansas Nuclear One-Unit 2 Core Protection Calculator Addressable Constant Determination Methodology" dated August, 1981.

POWER DISTRIBUTION LIMITS

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-3 or 3.2-4, as applicable.

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

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-4.

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

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENT

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator Channel.

*See Special Test Exception 3.10.2

POWER DISTRIBUTION LIMITS

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2225 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to $\geq$ 11% of RATED THERMAL POWER.

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 the inoperable status.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 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 during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 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 its 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.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing either a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use or at least once per 7 days when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 200°F in any one hour period, and

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Apppendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature Limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the funcitonal requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement 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, Specification 3.3.3.8.

f. Inoperable Fire Suppression Systems, Specifications 3.7.10.1 and 3.7.10.2.

g. Primary coolant specific activity, Specification 3.4.8

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSC and the SRC.

l. Records of changes to the Core Protection Calculator System (CPCS) SOFTWARE. Changes to the CPCS SOFTWARE shall be made in accordance with methods approved by the NRC. These records shall include the following:

   1. Purpose of change.

   2. Detailed description of change including algorithms, changes to the assembly listings, checksums and disk identification numbers.

   3. Summary of validation test results.

m. Records of Environmental Qualification which are covered under the provisions of paragraph 6.12.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 ENVIRONMENTAL QUALIFICATION

6.12.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License NPF-6 dated October 24, 1980.

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 SRC shall be composed of the:

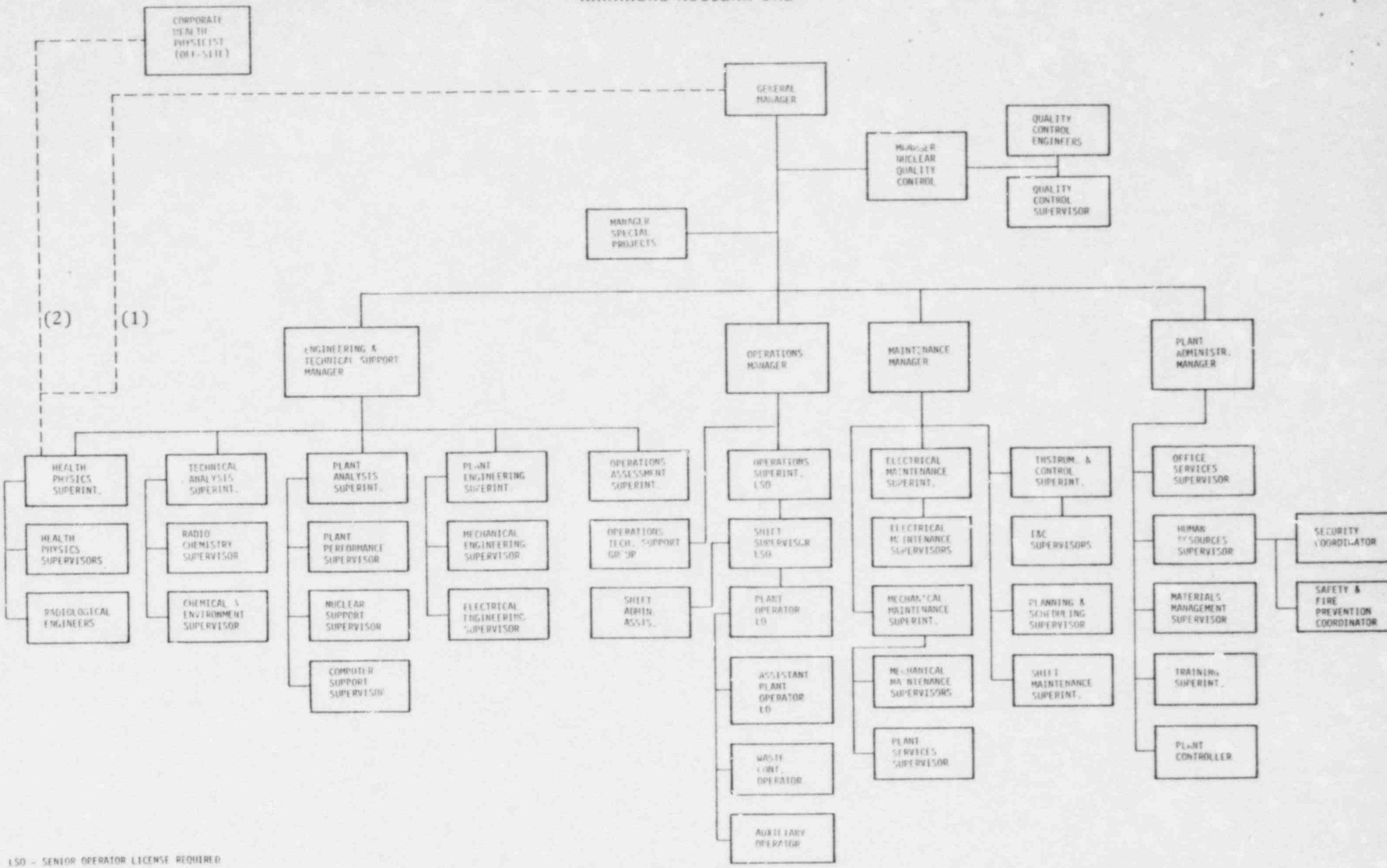
Chairman: Director, Technical and Environmental Services
Member: Vice-President, Nuclear Operations
Member: Director, Generation Technology
Member: Manager, Quality Assurance
Member: Arkansas Nuclear One Plant Safety Committee Chairman
Member: Manager, Technical Analysis
Member: Arkansas Nuclear One Plant Analysis Superintendent
Member: Director, Generation Engineering
Member: Manager, Instrumentation and Controls Engineering
Member: Manager of System Nuclear Operations*

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

*Middle South Services

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE



CODE: LSO - SENIOR OPERATOR LICENSE REQUIRED

LS - OPERATOR LICENSE REQUIRED

*ONSITE RESPONSIBILITY FOR FIRE PROTECTION PROGRAM

(1) The Health Physics Superintendent reports to the Manager, Engineering and Technical Support in administrative matters and routine health physics concerns and he reports to the General Manager in matters of radiological health, safety and policy.

(2) The Health Physics Superintendent has direct interface with the Corporate Health Physicist in matters of radiological health and safety. The Corporate Health Physicist reports to the Manager, Nuclear Services. He will help formulate Corporate Health Physics Policy and ensure that it is properly implemented.

FIGURE 6.2-2 FUNCTIONAL ORGANIZATION FOR PLANT OPERATION

ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS NUCLEAR ONE

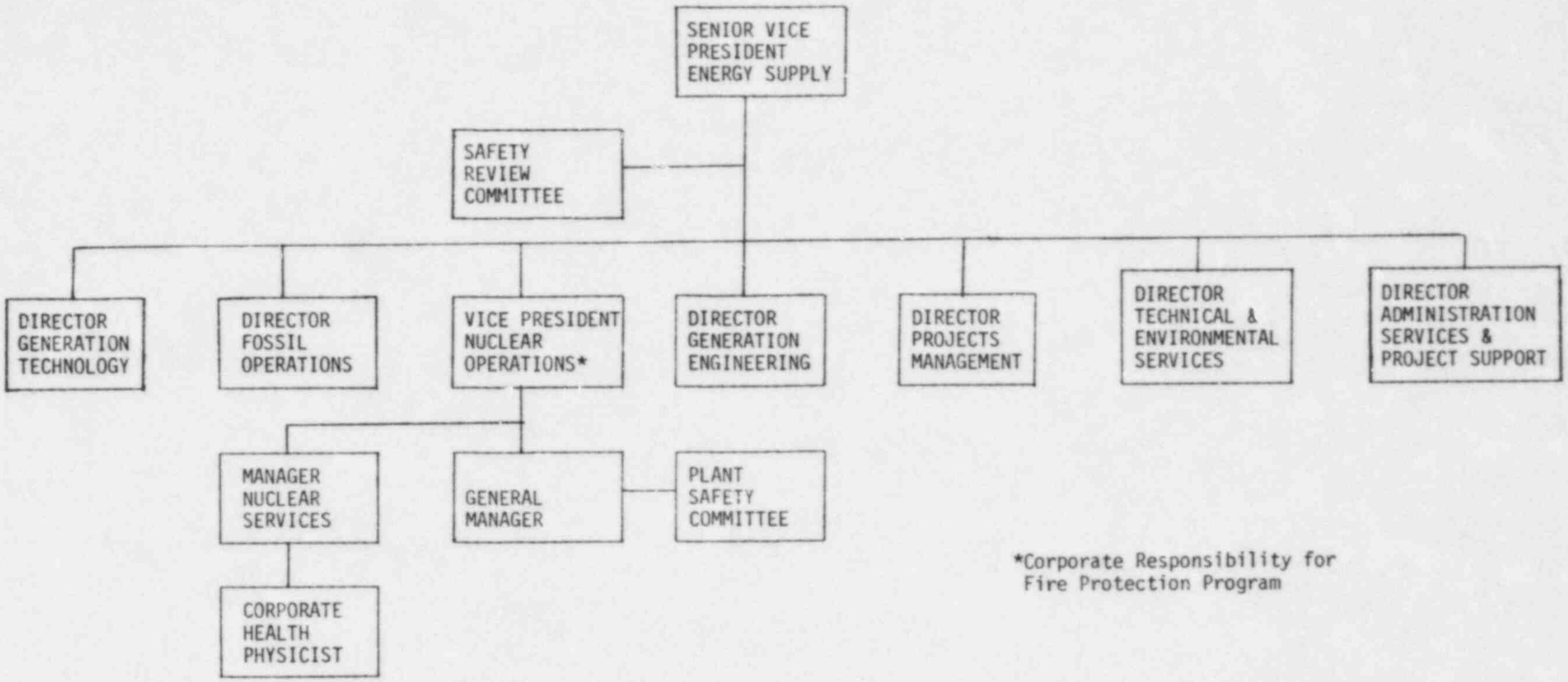


Figure 6.2-1 Management Organization Chart

f. Review of events requiring 24 hours written notification to the Commission.

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 General Manager of the Safety Review Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the General Manager and the Safety Review Committee.

j. Review of the Emergency Plan and inplementing procedures and shall submit recommended changes to the General Manager and the Safety Review Committee.

AUTHORITY

6.5.1.7 The Plant Safety Committee shall:

a. Recommend in writing to the General Manager 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, Nuclear Operations and the Safety Review Committee of disagreement between the PSC and General Manager; however, the General Manager 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 General Manager and Chairman of the Safety Review Committee.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved and forwarded to the Senior Vice President, Energy Supply (SRVP,ES) within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President, Energy Supply (SRVP,ES) within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President, Energy Supply (SRVP,ES) and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PSC and submitted to the SRC and the Vice-President, Nuclear Operations.

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 Safety Limit violation shall be reported to the Commission, the Vice-President, Nuclear Operations and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Vice-President, Nuclear Operations within 14 days of the violation.

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.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administratvie procedures.