ATTACHMENT 1

SERIAL: NLS-84-204

BRUNSWICK STEAM ELECTRIC PLANT
PROPOSED TECHNICAL SPECIFICATION PAGES - UNIT 1

(84TSB13)

(9974MAT/cfr)

SUMMARY LIST OF REVISIONS
BRUNSWICK UNIT 1

Page	Comments
II	"Reportable Occurrence" changed to "Reportable Event"
XV	Incorporates change to reporting requirements
XVI	Revised to reflect repagination
1-6	Reportable Event change incorporated
3/4 3-62	Specification 6.9.1.14.b changed to 6.6.1
3/4 3-68	Specification 6.9.1.14.b changed to 6.6.1
3/4 4-7	Reference to Specification 6.9.1.12 deleted
3/4 4-10	Mathematical Symbols have been put into words
	Revised to reflect Special Report changes
	Typographical errors corrected ("of the" added)
3/4 6-6	Revised to reflect reporting of abnormal primary containment degradation pursuant to Specification 6.9.2
3/4 11-22	Specification 6.9.1.14.b changed to 6.6.1
6-3	Revised to reflect organizational changes
6-4	Revised to reflect organizational changes
6-5	Revised to reflect organizational changes
6-8	"Manager - Operations" changed to "Director - Training"
6-11	"Manager - Plant Operations" eliminated from Section 6.5.3.3, PNSC membership
6-12	Reportable Events change incorporated
6-13	"Reportable Occurrence" changed to "Reportable Event"
6-14	Comma added in Section 6.5.4.6
6-15	Revised to reflect title changes
	"Bi-monthly" changed to "once every two months"
6-17	"Principal QA Specialist - Performance Evaluation Unit" changed to "Manager - Quality Assurance Service Section"
6-18	Reportable Events change incorporated
6-20	Reportable Events change incorporated
6-25	Sections 6.9.1.12, 6.9.1.13, and 6.9.1.14 deleted
	Item 6.9.2.c added
	Item 6.9.2.q added
	Section 6.9.2 re-ordered
	Repaginated

(9974MAT/cfr)

Page	Comments
6-26	"Reportable Occurrences" changed to "Reportable Events"
	Repaginated
6-27	Reference to snubber Table 3.7.5-1 revised to reflect table deletion per our submittal dated May 7, 1984
	Repaginated
6-28	"Dose" added to "Offsite Calculation Manual"
	Typographical error corrected ("toally" changed to "totally")
	Repaginated
6-29	Repaginated
6-30	Repaginated

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DEFINITIONS

SECTION

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ADMINISTRATIVE CONTROLS

DEFINITIONS

PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the containment.

RATED THERMAL POWER

RATED THERMAL POWER shall be total reactor core heat transfer rate to the reactor coolant of 2436 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All automatic reactor building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position.

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the reactor building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.5.8-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.8-1.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than one radioactive liquid effluent monitoring instrumentation channel in each release pathway OPERABLE, take the ACTION shown in Table 3.3.5.8-1. Return the instruments to OPERABLE status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.6.1 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.5.8-1.

NOTE: See Bases 3/4.3.5.8.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.5.9-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology as described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.9-1.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than one radioactive gaseous effluent monitoring instrumentation channel OPERABLE, take the ACTION shown in Table 3.3.5.9-1. Return the instruments to OPERABLE status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.6.1 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.5.9-1.

NOTE: See Bases 3/4.3.5.9.

REACTOR COOLANT SYSTEM

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, and 3:

   1. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1, but less than 10 µmho/cm at 25°C and less than 0.5 ppm, respectively, operation may continue for up to 24 hours and this condition need not be reported to the Commission provided that operation under these conditions shall not exceed 336 hours per year. The provisions of Specification 3.0.4 are not applicable.

   2. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1 for more than 24 hours during one continuous time interval or with the conductivity exceeding 10 µmho/cm at 25°C or chloride exceeding 0.5 ppm, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. At all other times with the conductivity and/or chloride concentration of the reactor coolant in excess of the limit specified in Table 3.4.4-1, restore the conductivity and/or chloride concentration to within the limit within 48 hours.

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the reactor coolant shall be limited to:

a. less than or equal to 0.2 µCi/gram DOSE EQUIVALENT I-131, and

b. less than or equal to $100/\overline{E}$ µCi/gram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, and 3, with the specific activity of the reactor coolant;

1. Greater than 0.2 µCi/gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 µCi/gram, operation may continue for up to 48 hours provided that operation under these conditions shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 µCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. Greater than $100/\overline{E}$ µCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 1, 2, 3, or 4,

1. With the specific activity of the primary coolant greater than 0.2 µCi/gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ µCi/gram, perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 at least once per 4 hours until the specific activity of the primary coolant is restored to within its limits. In lieu of a License Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 µCi/gram DOSE EQUIVALENT I-131 together with the below additional information.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.4.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2. This Special Report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3, 3.0.4, and 6.6.1 are not applicable.

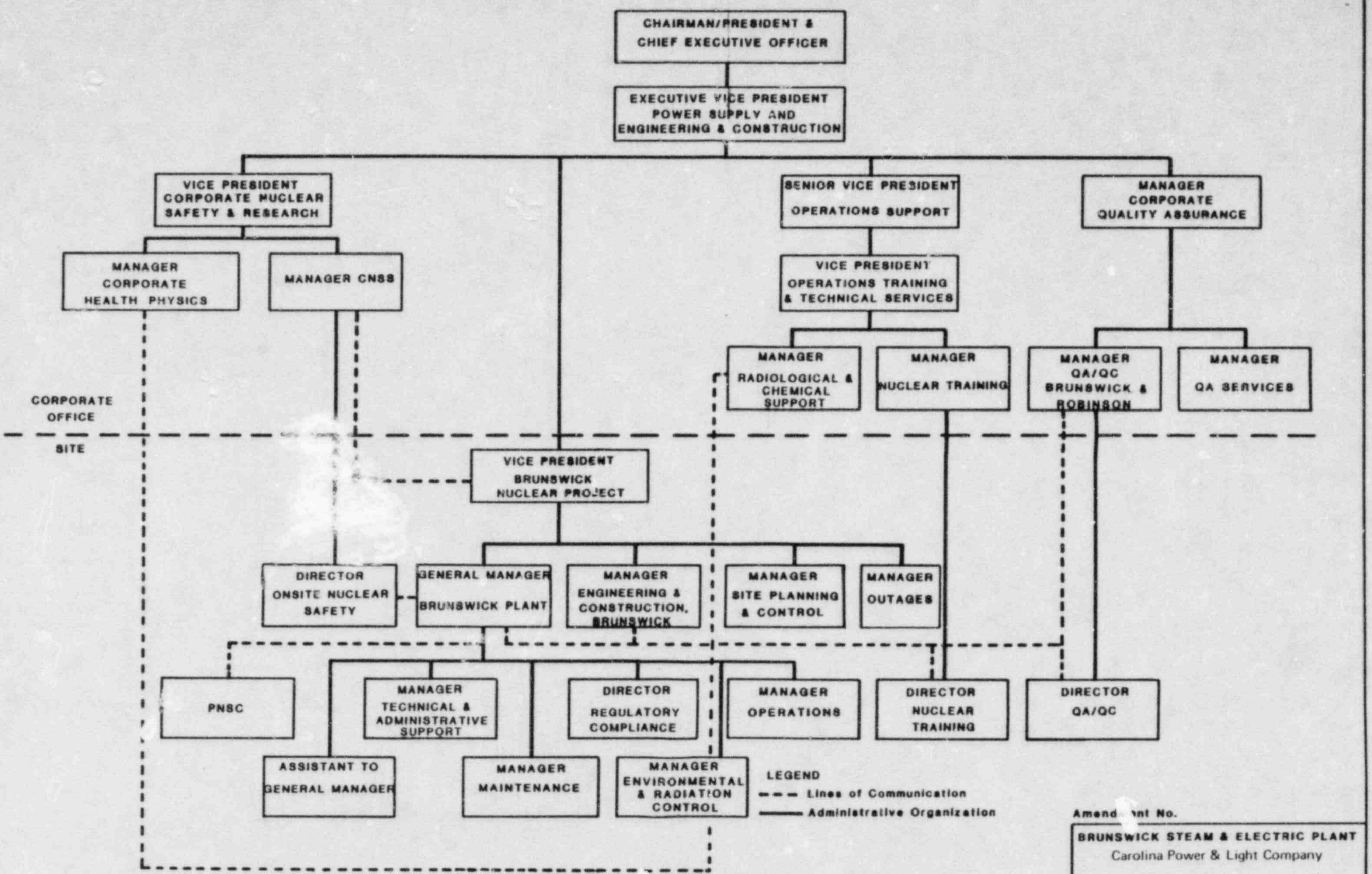
SURVEILLANCE REQUIREMENTS

4.11.3 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection of testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.14, to assure SOLIDIFICATION of subsequent batches of waste.

NOTE: See Bases 3/4.11.3

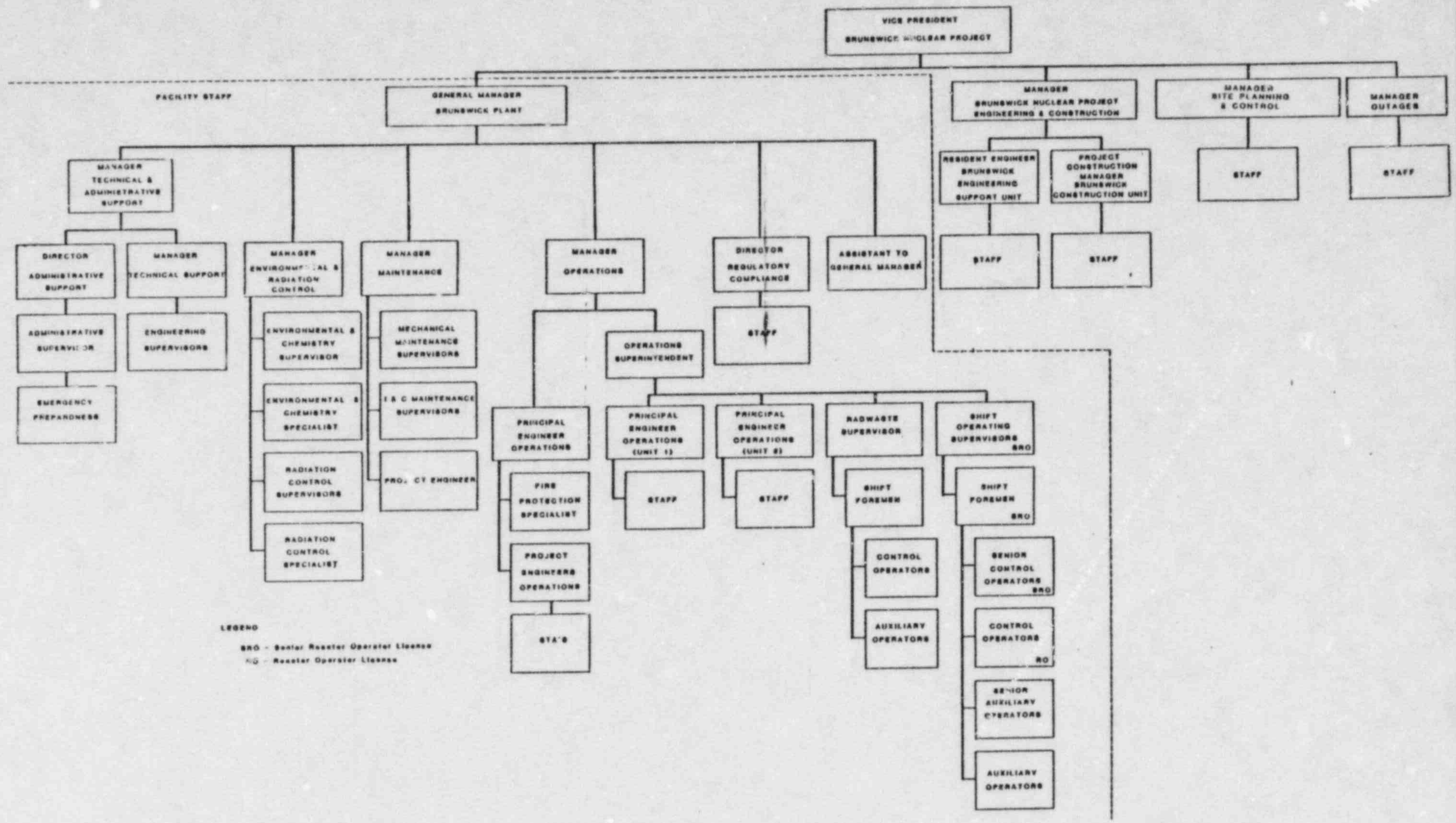


Amendment No.

BRUNSWICK STEAM & ELECTRIC PLANT
Carolina Power & Light Company

MANAGEMENT ORGANIZATION

Figure 6.2.1-1



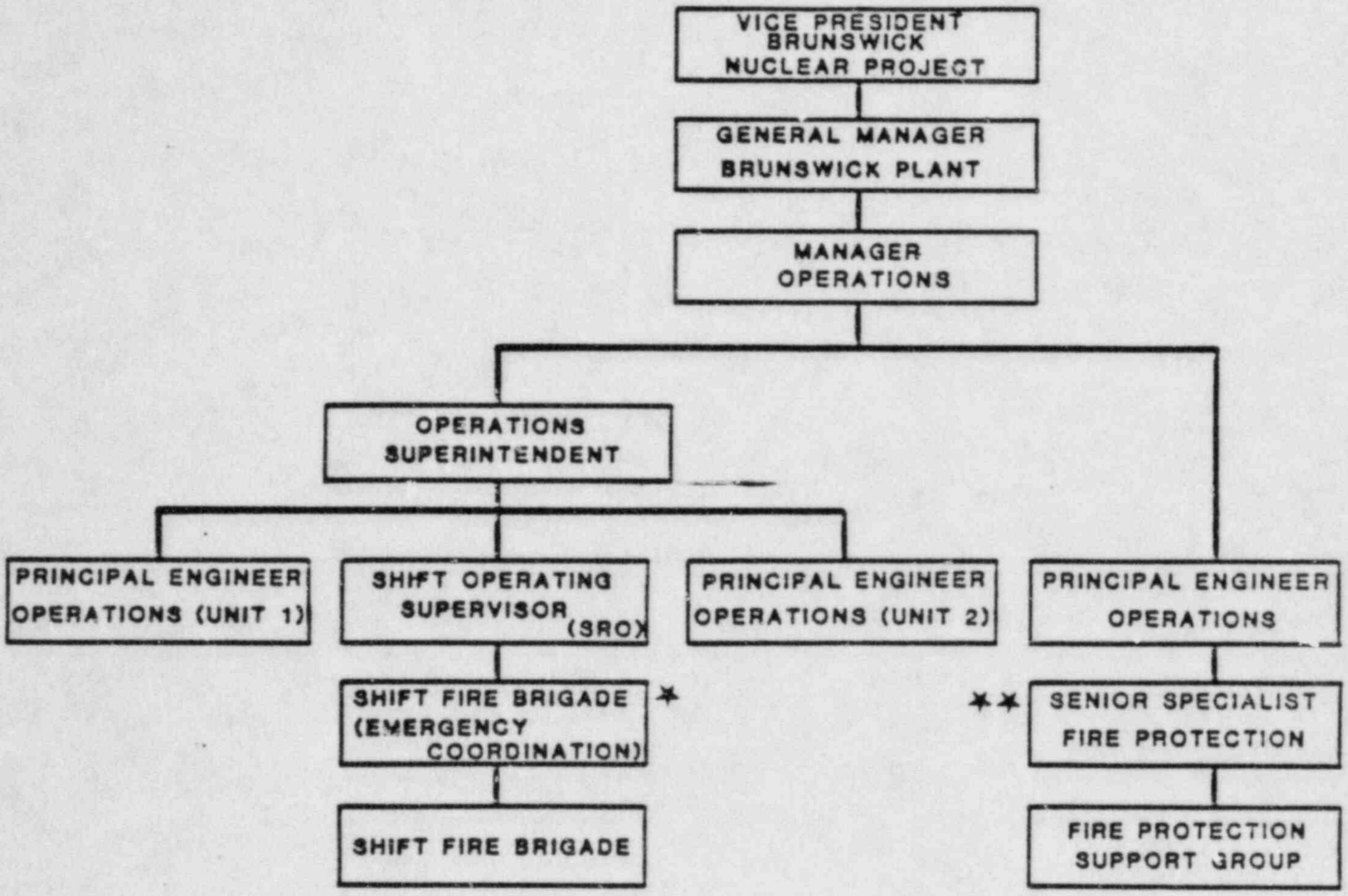
Amendment No.

BRUNSWICK STEAM & ELECTRIC PLANT

Carolina Power & Light Company

FACILITY ORGANIZATION

Figure 6.2.2-1



LEGEND

* Number of Brigade Fire Chiefs varies with Shift Organization

** One Specialist is Assigned The Duties of The Plant Fire Chief

Amendment No.

BRUNSWICK STEAM & ELECTRIC PLANT Carolina Power & Light Company	PLANT FIRE PROTECTION ORGANIZATION Figure 6.2.2-2	

6.2.3 ONSITE NUCLEAR SAFETY (ONS)

FUNCTION

6.2.3.1 The ONS Unit shall function to examine facility operating characterisitics, NRC issues, industry advisories, and other sources which may indicate areas for improving facility safety.

RESPONSIBILITIES

6.2.3.2 The ONS Unit shall be responsible for maintaining surveillance of facility activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.3 The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving facility safety to the Manager-Corporate Nuclear Safety Section.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Operating Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.3 FACILITY STAFF QUALIFICATION

6.3.1 Each member of the facility staff defined in Figure 6.2.2-1 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Manager - Environmental & Radiation Control who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant during transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Director - Training 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 A training program for the Fire Brigade shall be maintained under the direction of the Director - Training and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

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* Not responsible for sign-off function.

6.5.3 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.3.1 As an effective means for the regular review, overview, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) shall be established.

6.5.3.2 The PNSC shall function through the utilization of subcommittees, audits, investigations, reports, and/or performance of reviews as a group.

COMPOSITION

6.5.3.3 The PNSC shall be composed of the:

Chairman:	General Manager - Brunswick Plant*
Member:	Manager - Technical & Administrative Support
Member:	Manager - Technical Support
Member:	Manager - Operations
Member:	Manager - Maintenance
Member:	Manager - Environmental & Radiation Control
Member:	Assistant to Plant General Manager
Member:	Director - QA/QC
Member:	Director - Regulatory Compliance
Member:	Director - Administrative Support

ALTERNATES

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

6.5.3.5 All alternates, shall as a minimum, meet equivalent qualification criteria as specified for professional-technical personnel in Section 4.4 of ANSI N18.1-1971.

MEETING FREQUENCY

6.5.3.6 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or his designated alternate.

QUORUM

6.5.3.7 The minimum quorum of the PNSC necessary for the performance of the PNSC activities of the Technical Specifications shall consist of the PNSC Chairman or his designated alternate and five members including alternates. No more than two alternates shall be counted toward meeting the minimum quorum requirement.

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* Or designated alternate.

ACTIVITIES

6.5.3.8 The PNSC activities shall include the following:

a. Review of all procedures required by Specification 6.8 and changes thereto (and any other procedures and changes thereto), any of which constitute an unreviewed safety question or involve a change to the Technical Specifications, prior to implementation.

b. Review of all proposed tests or experiments that constitute an unreviewed safety question or involve a change to the Technical Specifications, prior to implementation.

c. Review of all proposed modifications that constitute an unreviewed safety question or involve a change to the Technical Specifications, prior to implementation.

d. Review of all proposed changes to the Technical Specifications or Operating License, prior to implementation.

e. Review of reports on violations of Technical Specifications including reports covering evaluation and recommendations to prevent recurrence to the Vice President - Brunswick Nuclear Project and to the Manager - Corporate Nuclear Safety Section.

f. Performance of special reviews, investigations (or analyses), and reports thereon as requested by the Manager - Corporate Nuclear Safety Section.

g. Review of all REPORTABLE EVENTS.

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

i. Annual review of the Security Plan.

j. Annual review of the Emergency Plan.

k. Review of accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Brunswick Nuclear Project and the Manager - Corporate Nuclear Safety Section.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

AUTHORITY

6.5.3.9 If there is a disagreement between recommendations of a majority of the PNSC and the actions contemplated by the General Manager - Brunswick Plant, the PNSC shall provide written notification within 24 hours to the Vice President - Brunswick Nuclear Project and the Vice President - Corporate Nuclear Safety and Research. The course determined by the General Manager - Brunswick Plant to be the most conservative shall be followed.

RECORDS

6.5.3.10 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications. Copies shall be provided to the Vice President - Brunswick Nuclear Project and the Manager - Corporate Nuclear Safety Section.

6.5.4 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.4.1 The Corporate Nuclear Safety Section (CNSS) of the Corporate Nuclear Safety & Research Department shall function to provide independent review of significant plant changes, tests, and procedures; verify that REPORTABLE EVENTS are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer.

ORGANIZATION

6.5.4.2 The individuals assigned responsibility for independent reviews shall be specified in technical disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. non-destructive testing

f. instrumentation and control

g. radiological safety

h. mechanical and electrical engineering

i. administrative controls

ADMINISTRATIVE CONTROLS

ORGANIZATION (Continued)

j. seismic and environmental

k. quality assurance practices

l. Other appropriate fields associated with the unique characteristics of the nuclear power plant.

6.5.4.3 The Manager - Corporate Nuclear Safety Section shall have an academic degree in an engineering or related field and, in addition, shall have a minimum of ten years related experience, of which a minimum of five years shall be in the operation and/or design of nuclear power plants.

6.5.4.4 The independent safety review program reviewers shall have an academic degree in an engineering or related field or equivalent and, in addition, shall have a minimum of five years related experience.

6.5.4.5 An individual may possess competence in more than one specialty area. If sufficient expertise is not available within the Corporate Nuclear Safety Section, competent individuals from other Carolina Power & Light Company organizations or outside consultants shall be utilized in performing independent reviews and investigations.

6.5.4.6 At least three individuals, qualified as discussed in 6.5.4.4 above, shall review each item submitted under the requirements of Section 6.5.4.9.

6.5.4.7 Independent safety reviews shall be performed by individuals not directly involved with the activity under review or responsible for the activity under review.

6.5.4.8 The Corporate Nuclear Safety Section independent safety review program shall be conducted in accordance with written, approved procedures.

REVIEW

6.5.4.9 The Corporate Nuclear Safety Section shall perform reviews of the following:

a. The safety evaluations for 1) changes to procedures required by Specification 6.8, 2) modifications of equipment or systems, and 3) tests or experiments that constitute a change to the safety analysis report to verify that such actions did not constitute an unreviewed safety question or involve a change to the Technical Specifications. Implementation may proceed prior to completion of this review.

b. Proposed changes to procedures required by Specification 6.8, and proposed modifications that constitute an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications, prior to implementation.

REVIEW (Continued)

c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications, prior to implementation.

d. Proposed changes to the Technical Specifications and Operating License.

e. Violations, deviations, and events requiring 24 hour written notification to the Commission, such as:

  1. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, and internal procedures or instructions having nuclear safety significance.

  2. Significant operating abnormalities or deviations from normal and expected performance of plant safety-related structures, systems, or components.

f. Reports and minutes of the PNSC.

g. Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager - Corporate Nuclear Safety Section by the on-site operating organization or other functional organizational units within Carolina Power & Light Company.

6.5.4.10 Review of items considered under 6.5.4.9(e) through (g) above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.4.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNSS files.

b. Recommendations and concerns shall be submitted to the General Manager - Brunswick Plant and Vice President - Brunswick Nuclear Project, within 14 days of completion of the review.

c. A summation of Corporate Nuclear Safety recommendations and concerns shall be submitted to the Chairman/President and Chief Executive Officer; Executive Vice President - Power Supply and Engineering and Construction; Vice President - Corporate Nuclear Safety and Research; Vice President - Brunswick Nuclear Project; General Manager - Brunswick Plant; and others, appropriate, at least once every two months.

AUDITS (Continued)

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975, at least once per 12 months.

l. Any other area of facility operation considered appropriate by the Manager - Quality Assurance Services Section.

6.5.5.3 Personnel performing the quality assurance audits shall have access to the plant operating records.

RECORDS

6.5.5.4 Records of audits shall be prepared and retained.

6.5.5.5 Audit reports encompassed by 6.5.5.2 above shall be prepared, approved by the Manager - Quality Assurance Service Section, and forwarded to the Executive Vice President - Power Supply and Engineering and Construction; Vice President - Brunswick Nuclear Project; Vice President - Corporate Nuclear Safety and Research; General Manager - Brunswick Plant; and others, as appropriate, within 30 days after completion of the audit.

AUTHORITY

6.5.5.6 The Manager - Quality Assurance Services Section under the Manager - Corporate Quality Assurance shall be responsible for the following:

a. The administering of the Corporate Quality Assurance Audit Program.

b. The approval of the individual(s) selected to conduct quality assurance audits.

PERSONNEL

6.5.5.7 Audit personnel shall be independent of the area audited.

6.5.5.8 Selection of personnel for auditing assignments shall be based on experience or training that establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting audit personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

6.5.5.9 Qualified outside consultants or other individuals independent from those personnel directly involved in plant operation shall be used to augment the audit teams when necessary.

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.6.1 An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

6.5.6.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Plant Nuclear Safety Committee - Brunswick Plant and shall be submitted to the Manager - Corporate Nuclear Safety Section and the Vice President - Brunswick Nuclear Project.

6.7 SAFETY LIMIT VIOLATION

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

a. The facility shall be placed in at least HOT SHUTDOWN within two hours.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President - Brunswick Nuclear Project and the Manager - Corporate Nuclear Safety Section shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the General Manager - Brunswick Plant. 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 Vice President - Brunswick Nuclear Project, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

c. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

ADMINISTRATIVE CONTROLS

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional 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. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Accident Monitoring Instrumentation, Specification 3.3.5.3.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Reactor coolant specific activity analysis, Specification 3.4.5.

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

h. Fire barrier penetration, Specification 3.7.8.

i. Liquid Effluents Dose, Specification 3.11.1.2.

j. Liquid Radwaste Treatment, Specification 3.11.1.3.

k. Dose - Noble Gases, Specification 3.11.2.2.

l. Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

m. Gaseous Radwaste Treatment, Specification 3.11.2.4.

n. Ventilation Exhaust Treatment, Specification 3.11.2.5.

o. Total Dose, Specification 3.11.4.

p. Monitoring Program, Specification 3.12.1.b.

q. Primary Containment Structural Integrity, Specification 4.6.1.4.2

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

RECORDS RETENTION (Continued)

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records or radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.1-1.

g. Records of reactor tests and experiments.

h. Records of training and qualification for current members of the plant staff.

i. Records of inservice inspections performed pursuant to these Technical Specifications.

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

RECORDS RETENTION (Continued)

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

l. Records of the service lives of all hydraulic and mechanical snubbers referenced in Section 3.7.5 including the data at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

n. Records of (1) meetings of the PNSC, (2) meetings of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC), (3) the independent reviews performed by the Corporate Nuclear Safety Section, and (4) the independent reviews performed by the Corporate Quality Assurance Audit Program, Performance Evaluation Unit.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 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 HIGH RADIATION AREA

6.12.1 In lieu of the "Control Device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

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* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 above shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Operations Shift Foreman on duty and/or the Radiation Control Supervisor.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1. Sufficiently detailed information to totally support rationale without benefit of additional of supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and,

3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

PROCESS CONTROL PROGRAM (PCP) (Continued)

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2. A determination that the change did not reduce the overall conformance of the solidification waste product to existing criteria for solid wastes; and

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS[7/]

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

   2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

   4. An evaluation of the change that shows the predicted release of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   5. An evaluation of the change that shows the expected maximum exposure to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

7/ Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable to the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

ATTACHMENT 2

SERIAL: NLS-84-204

BRUNSWICK STEAM ELECTRIC PLANT
PROPOSED TECHNICAL SPECIFICATION PAGES - UNIT 2

(84TSB13)

(9974MAT/cfr)

SUMMARY LIST OF REVISIONS
BRUNSWICK UNIT 2

Page	Comments
II	"Reportable Occurrence" changed to "Reportable Event"
XV	Incorporates change to reporting requirements
	Revised to reflect repagination
XVI	Revised to reflect repagination
1-7	Reportable Event change incorporated
3/4 3-62	Specification 6.9.1.14.b changed to 6.6.1
3/4 3-68	Specification 6.9.1.14.b changed to 6.6.1
3/4 4-7	Reference to Specification 6.9.1.12 deleted
3/4 4-10	Revised to reflect Special Report changes
	Mathematical symbols have been put into words
	Typographical error corrected ("of the" added)
3/4 6-6	Revised to reflect reporting of abnormal primary containment degradation pursuant to Specification 6.9.2
3/4 11-22	Specification 6.9.1.14.b changed to 6.6.1
6-3	Revised to reflect organizational changes
6-4	Revised to reflect organizational changes
6-5	Revised to reflect organizational changes
6-6	Typographical error corrected ("Unit 2" changed to "Unit 1")
6-8	"Manager - Operations" changed to "Director - Training"
6-11	"Manager - Plant Operations" eliminated from Section 6.5.3.3, PNSC membership
6-12	Reportable Events change incorporated
6-13	"Reportable Occurrence" changed to "Reportable Events"
6-14	Comma added in Section 6.5.4.6
6-15	Revised to reflect title changes
	"Bi-monthly" changed to "once every two months"
6-17	"Principal QA Specialist - Performance Evaluation Unit" changed to "Manager - Quality Assurance Service Section"
6-18	Reportable Events change incorporated
6-20	Reportable Events change incorporated

(9974MAT/cfr)

Page	Comments
6-25	Sections 6.9.1.12, 6.9.1.13, and 6.9.1.14 deleted
	Item 6.9.2.c added
	Item 6.9.2.q added
	Section 6.9.2. re-ordered
	Repaginated
6-26	"Reportable Occurrences" changed to "Reportable Events"
	Repaginated
6-27	Reference to snubber Table 3.7.5-1 revised to reflect table deletion per our submittal of May 7, 1984
	Repaginated
6-28	"Dose" added to "Offsite Calculation Manual"
	Typographical error corrected ("toally" changed to "totally")
	Repaginated
6-29	Repaginated
6-30	Repaginated

(9974MAT/cfr)

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ADMINISTRATIVE CONTROLS

DEFINITIONS

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted are equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGERITY shall exist when:

a. All automatic Reactor Building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position.

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the Reactor Building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuing that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon-free.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee, as defined by Figure 5.1.3-1.

SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.5.8-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.8-1.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than one radioactive liquid effluent monitoring instrumentation channel in each release pathway OPERABLE, take the ACTION shown in Table 3.3.5.8-1. Return the instruments to OPERABLE status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.6.1 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.5.8-1.

NOTE: See Bases 3/4.3.5.8.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.5.9-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology as described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.9-1.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than one radioactive gaseous effluent monitoring instrumentation channel OPERABLE, take the ACTION shown in Table 3.3.5.9-1. Return the instruments to OPERABLE status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.6.1 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.5.9-1.

NOTE: See Bases 3/4.3.5.9.

REACTOR COOLANT SYSTEM

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, and 3:

   1. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1, but less than 10 µmho/cm at 25°C and less than 0.5 ppm, respectively, operation may continue for up to 24 hours and this condition need not be reported to the Commission provided that operation under these conditions shall not exceed 336 hours per year. The provisions of Specification 3.0.4 are not applicable.

   2. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1 for more than 24 hours during one continuous time interval or with the conductivity exceeding 10 µmho/cm at 25°C or chloride exceeding 0.5 ppm, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. At all other times with the conductivity and/or chloride concentration of the reactor coolant in excess of the limit specified in Table 3.4.4-1, restore the conductivity and/or chloride concentration to within the limit within 48 hours.

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the reactor coolant shall be limited to:

a. less than or equal to 0.2 μCi/gram DOSE EQUIVALENT I-131, and

b. less than or equal to $100/\overline{E}$ μCi/gram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, and 3, with the specific activity of the reactor coolant;

1. Greater than 0.2 μCi/gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 μCi/gram, operation may continue for up to 48 hours provided that operation under these conditions shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 μCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. Greater than $100/\overline{E}$ μCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 1, 2, 3, or 4,

1. With the specific activity of the primary coolant greater than 0.2 μCi/gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ μCi/gram, perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 at least once per 4 hours until the specific activity of the primary coolant is restored to within its limits. In lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 μCi/gram DOSE EQUIVALENT I-131 together with the below additional information.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.4.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2. This Special Report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3, 3.0.4, and 6.6.1 are not applicable.

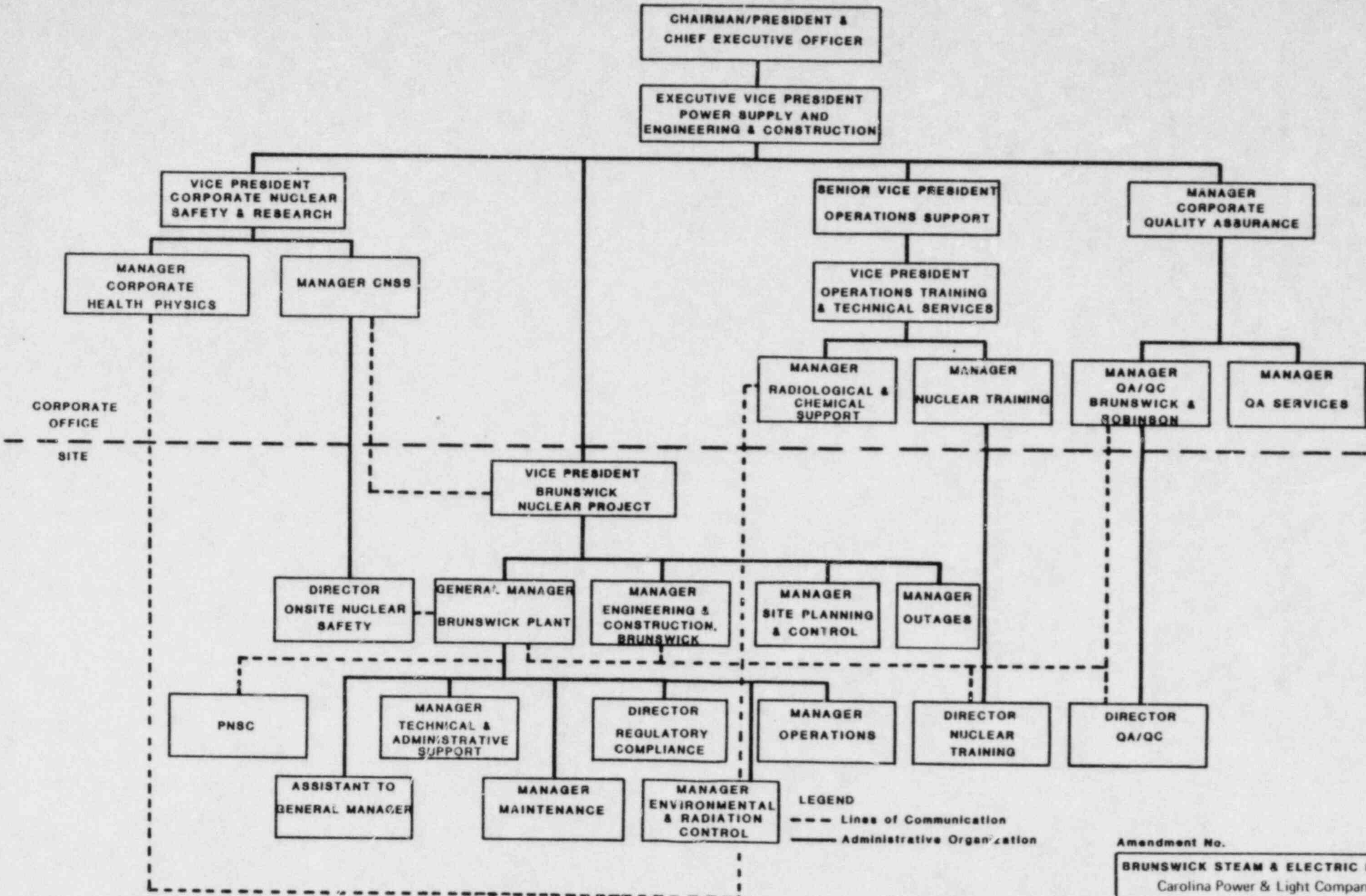
SURVEILLANCE REQUIREMENTS

4.11.3 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection of testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.14, to assure SOLIDIFICATION of subsequent batches of waste.

NOTE: See Bases 3/4.11.3

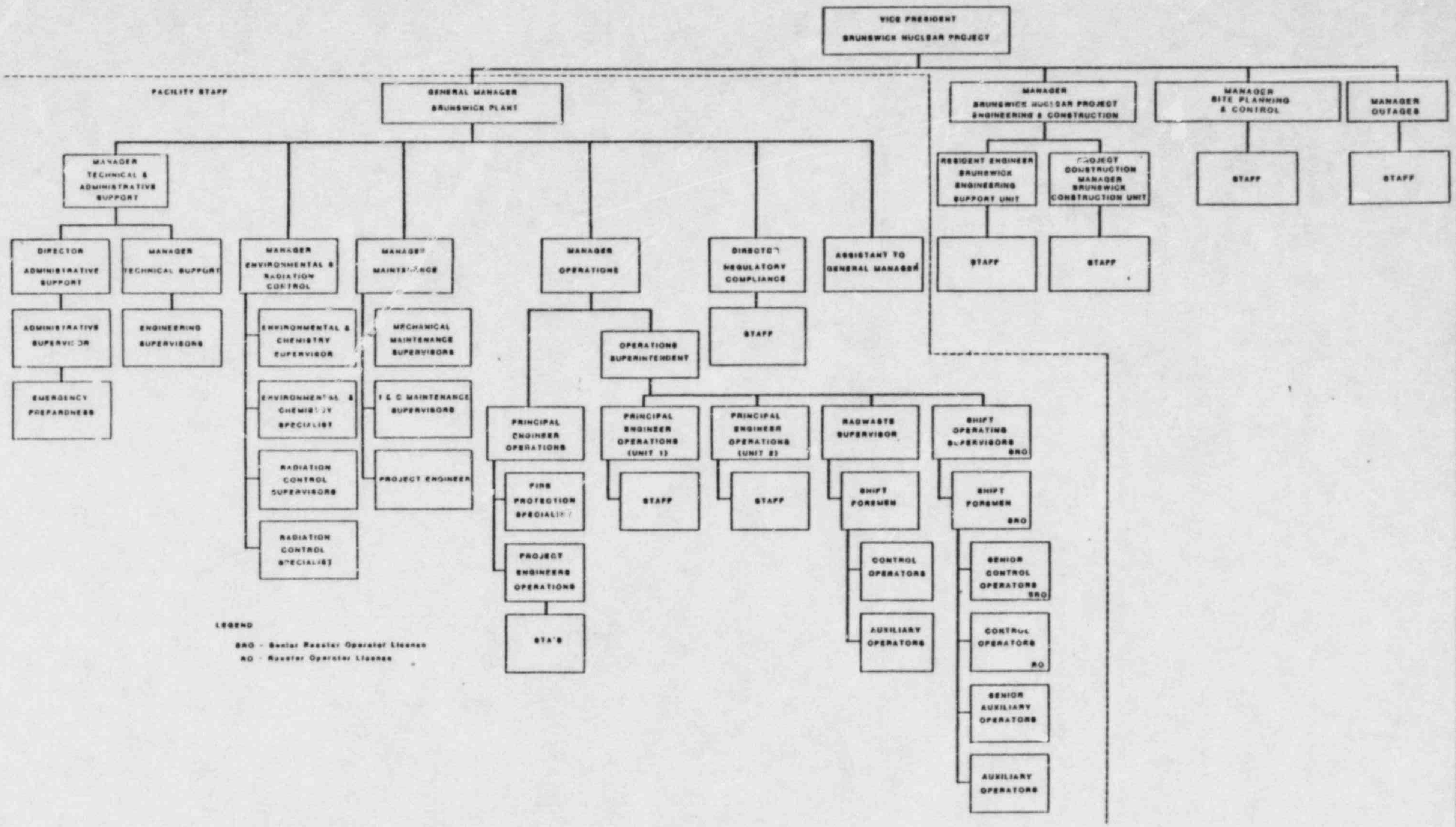


Amendment No.

BRUNSWICK STEAM & ELECTRIC PLANT
Carolina Power & Light Company

MANAGEMENT ORGANIZATION

Figure 6.2.1-1

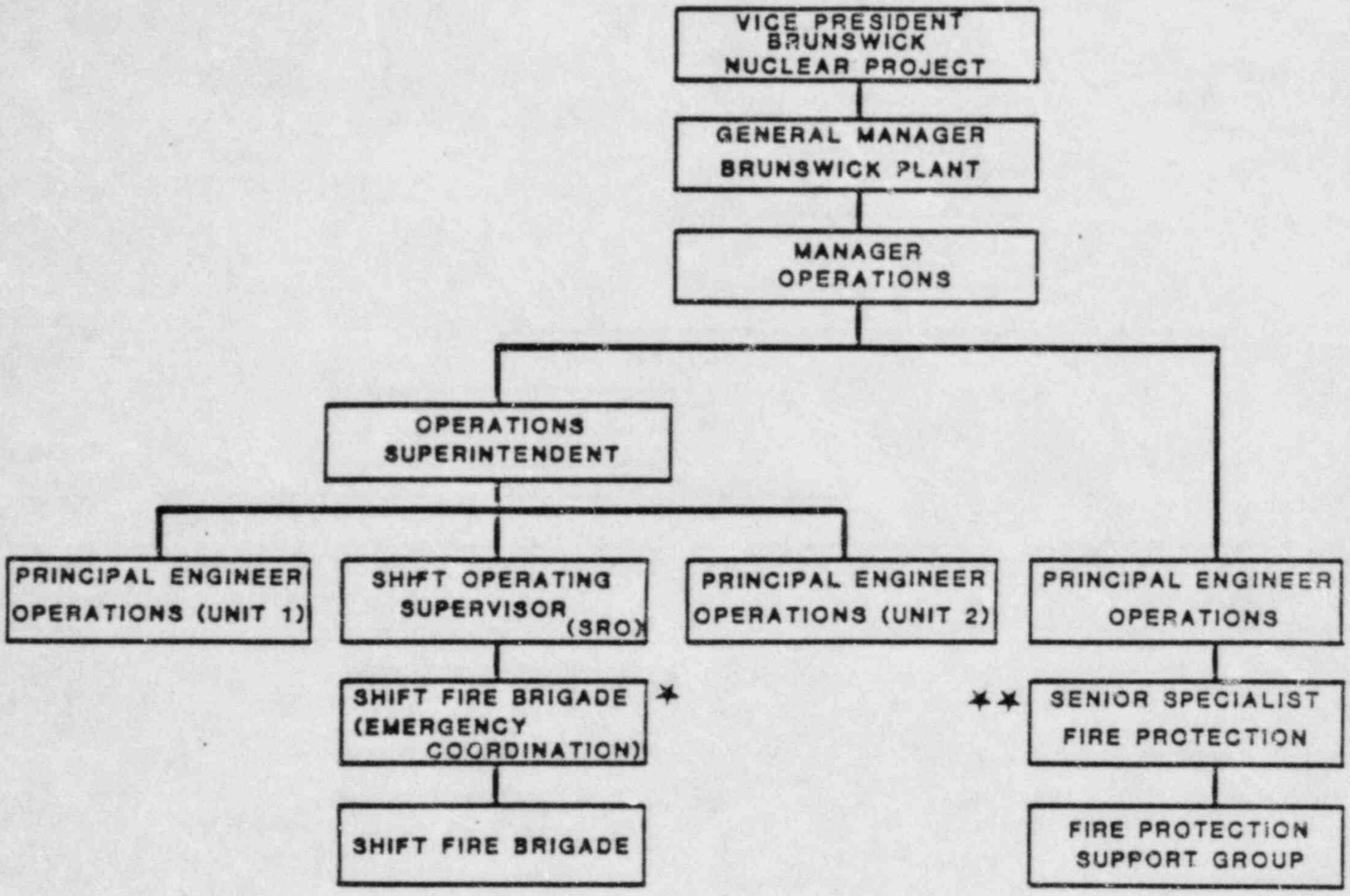


Amendment No.

BRUNSWICK STEAM & ELECTRIC PLANT
Carolina Power & Light Company

FACILITY ORGANIZATION

Figure 6.2.2-1



LEGEND

* Number of Brigade Fire Chiefs varies with Shift Organization

** One Specialist is Assigned The Duties of The Plant Fire Chief

Amendment No.

BRUNSWICK STEAM & ELECTRIC PLANT Carolina Power & Light Company	PLANT FIRE PROTECTION ORGANIZATION Figure 6.2.2-2	Page No. 6-5

TABLE 6.2.2-1

MINIMUM FACILITY SHIFT CREW COMPOSITION

WITH UNIT 1 IN CONDITION 1, 2, OR 3		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[(b)]
SRO[(a)]	1	1[(b)]
RO[(a)]	3	3
AO	3	3
STA	1	1

WITH UNIT 1 IN CONDITION 4 OR 5		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[(b)]
SRO[(a)]	1	1[(b)]
RO[(a)]	3	2
AO	3	3
STA	1	None

WITH UNIT 1 DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[(b)]
SRO[(a)]	1	1[(b)]
RO[(a)]	2	2
AO	3	3
STA	1	None

6.2.3 ONSITE NUCLEAR SAFETY (ONS)

FUNCTION

6.2.3.1 The ONS Unit shall function to examine facility operating characterisitics, NRC issues, industry advisories, and other sources which may indicate areas for improving facility safety.

RESPONSIBILITIES

6.2.3.2 The ONS Unit shall be responsible for maintaining surveillance of facility activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.3 The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving facility safety to the Manager-Corporate Nuclear Safety Section.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Operating Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.3 FACILITY STAFF QUALIFICATION

6.3.1 Each member of the facility staff defined in Figure 6.2.2-1 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Manager - Environmental & Radiation Control who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant during transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Director - Training 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 A training program for the Fire Brigade shall be maintained under the direction of the Director - Training and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

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* Not responsible for sign-off function.

6.5.3 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.3.1 As an effective means for the regular review, overview, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) shall be established.

6.5.3.2 The PNSC shall function through the utilization of subcommittees, audits, investigations, reports, and/or performance of reviews as a group.

COMPOSITION

6.5.3.3 The PNSC shall be composed of the:

Chairman:	General Manager - Brunswick Plant*
Member:	Manager - Technical & Administrative Support
Member:	Manager - Technical Support
Member:	Manager - Operations
Member:	Manager - Maintenance
Member:	Manager - Environmental & Radiation Control
Member:	Assistant to Plant General Manager
Member:	Director - QA/QC
Member:	Director - Regulatory Compliance
Member:	Director - Administrative Support

ALTERNATES

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

6.5.3.5 All alternates, shall as a minimum, meet equivalent qualification criteria as specified for professional-technical personnel in Section 4.4 of ANSI N18.1-1971.

MEETING FREQUENCY

6.5.3.6 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or his designated alternate.

QUORUM

6.5.3.7 The minimum quorum of the PNSC necessary for the performance of the PNSC activities of the Technical Specifications shall consist of the PNSC Chairman or his designated alternate and five members including alternates. No more than two alternates shall be counted toward meeting the minimum quorum requirement.

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* Or designated alternate.

ADMINISTRATIVE CONTROLS

ACTIVITIES

6.5.3.8 The PNSC activities shall include the following:

a. Review of all procedures required by Specification 6.8 and changes thereto (and any other procedures and changes thereto), any of which constitute an unreviewed safety question or involve a change to the Technical Specifications, prior to implementation.

b. Review of all proposed tests or experiments that constitute an unreviewed safety question or involve a change to the Technical Specifications, prior to implementation.

c. Review of all proposed modifications that constitute an unreviewed safety question or involve a change to the Technical Specifications, prior to implementation.

d. Review of all proposed changes to the Technical Specifications or Operating License, prior to implementation.

e. Review of reports on violations of Technical Specifications including reports covering evaluation and recommendations to prevent recurrence to the Vice President - Brunswick Nuclear Project and to the Manager - Corporate Nuclear Safety Section.

f. Performance of special reviews, investigations (or analyses), and reports thereon as requested by the Manager - Corporate Nuclear Safety Section.

g. Review of all REPORTABLE EVENTS.

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

i. Annual review of the Security Plan.

j. Annual review of the Emergency Plan.

k. Review of accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Brunswick Nuclear Project and the Manager - Corporate Nuclear Safety Section.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

ADMINISTRATIVE CONTROLS

AUTHORITY

6.5.3.9 If there is a disagreement between recommendations of a majority of the PNSC and the actions contemplated by the General Manager - Brunswick Plant, the PNSC shall provide written notification within 24 hours to the Vice President - Brunswick Nuclear Project and the Vice President - Corporate Nuclear Safety and Research. The course determined by the General Manager - Brunswick Plant to be the most conservative shall be followed.

RECORDS

6.5.3.10 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications. Copies shall be provided to the Vice President - Brunswick Nuclear Project and the Manager - Corporate Nuclear Safety Section.

6.5.4 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.4.1 The Corporate Nuclear Safety Section (CNSS) of the Corporate Nuclear Safety & Research Department shall function to provide independent review of significant plant changes, tests, and procedures; verify that REPORTABLE EVENTS are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer.

ORGANIZATION

6.5.4.2 The individuals assigned responsibility for independent reviews shall be specified in technical disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. non-destructive testing

f. instrumentation and control

g. radiological safety

h. mechanical and electrical engineering

i. administrative controls

ORGANIZATION (Continued)

j. seismic and environmental

k. quality assurance practices

l. Other appropriate fields associated with the unique characteristics of the nuclear power plant.

6.5.4.3 The Manager - Corporate Nuclear Safety Section shall have an academic degree in an engineering or related field and, in addition, shall have a minimum of ten years related experience, of which a minimum of five years shall be in the operation and/or design of nuclear power plants.

6.5.4.4 The independent safety review program reviewers shall have an academic degree in an engineering or related field or equivalent and, in addition, shall have a minimum of five years related experience.

6.5.4.5 An individual may possess competence in more than one specialty area. If sufficient expertise is not available within the Corporate Nuclear Safety Section, competent individuals from other Carolina Power & Light Company organizations or outside consultants shall be utilized in performing independent reviews and investigations.

6.5.4.6 At least three individuals, qualified as discussed in 6.5.4.4 above, shall review each item submitted under the requirements of Section 6.5.4.9.

6.5.4.7 Independent safety reviews shall be performed by individuals not directly involved with the activity under review or responsible for the activity under review.

6.5.4.8 The Corporate Nuclear Safety Section independent safety review program shall be conducted in accordance with written, approved procedures.

REVIEW

6.5.4.9 The Corporate Nuclear Safety Section shall perform reviews of the following:

a. The safety evaluations for 1) changes to procedures required by Specification 6.8, 2) modifications of equipment or systems, and 3) tests or experiments that constitute a change to the safety analysis report to verify that such actions did not constitute an unreviewed safety question or involve a change to the Technical Specifications. Implementation may proceed prior to completion of this review.

b. Proposed changes to procedures required by Specification 6.8, and proposed modifications that constitute an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications, prior to implementation.

REVIEW (Continued)

c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications, prior to implementation.

d. Proposed changes to the Technical Specifications and Operating License.

e. Violations, deviations, and events requiring 24 hour written notification to the Commission, such as:

   1. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, and internal procedures or instructions having nuclear safety significance.

   2. Significant operating abnormalities or deviations from normal and expected performance of plant safety-related structures, systems, or components.

f. Reports and minutes of the PNSC.

g. Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager - Corporate Nuclear Safety Section by the on-site operating organization or other functional organizational units within Carolina Power & Light Company.

6.5.4.10 Review of items considered under 6.5.4.9(e) through (g) above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.4.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNSS files.

b. Recommendations and concerns shall be submitted to the General Manager - Brunswick Plant and Vice President - Brunswick Nuclear Project, within 14 days of completion of the review.

c. A summation of Corporate Nuclear Safety recommendations and concerns shall be submitted to the Chairman/President and Chief Executive Officer; Executive Vice President - Power Supply and Engineering and Construction; Vice President - Corporate Nuclear Safety and Research; Vice President - Brunswick Nuclear Project; General Manager - Brunswick Plant; and others, appropriate, at least once every two months.

AUDITS (Continued)

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975, at least once per 12 months.

l. Any other area of facility operation considered appropriate by the Manager - Quality Assurance Services Section.

6.5.5.3 Personnel performing the quality assurance audits shall have access to the plant operating records.

RECORDS

6.5.5.4 Records of audits shall be prepared and retained.

6.5.5.5 Audit reports encompassed by 6.5.5.2 above shall be prepared, approved by the Manager - Quality Assurance Service Section, and forwarded to the Executive Vice President - Power Supply and Engineering and Construction; Vice President - Brunswick Nuclear Project; Vice President - Corporate Nuclear Safety and Research; General Manager - Brunswick Plant; and others, as appropriate, within 30 days after completion of the audit.

AUTHORITY

6.5.5.6 The Manager - Quality Assurance Services Section under the Manager - Corporate Quality Assurance shall be responsible for the following:

a. The administering of the Corporate Quality Assurance Audit Program.

b. The approval of the individual(s) selected to conduct quality assurance audits.

PERSONNEL

6.5.5.7 Audit personnel shall be independent of the area audited.

6.5.5.8 Selection of personnel for auditing assignments shall be based on experience or training that establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting audit personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

6.5.5.9 Qualified outside consultants or other individuals independent from those personnel directly involved in plant operation shall be used to augment the audit teams when necessary.

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.6.1 An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

6.5.6.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Plant Nuclear Safety Committee - Brunswick Plant and shall be submitted to the Manager - Corporate Nuclear Safety Section and the Vice President - Brunswick Nuclear Project.

6.7 SAFETY LIMIT VIOLATION

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

a. The facility shall be placed in at least HOT SHUTDOWN within two hours.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President - Brunswick Nuclear Project and the Manager - Corporate Nuclear Safety Section shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the General Manager - Brunswick Plant. 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 Vice President - Brunswick Nuclear Project, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

c. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional 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. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Accident Monitoring Instrumentation, Specification 3.3.5.3.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Reactor coolant specific activity analysis, Specification 3.4.5.

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

h. Fire barrier penetration, Specification 3.7.8.

i. Liquid Effluents Dose, Specification 3.11.1.2.

j. Liquid Radwaste Treatment, Specification 3.11.1.3.

k. Dose - Noble Gases, Specification 3.11.2.2.

l. Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

m. Gaseous Radwaste Treatment, Specification 3.11.2.4.

n. Ventilation Exhaust Treatment, Specification 3.11.2.5.

o. Total Dose, Specification 3.11.4.

p. Monitoring Program, Specification 3.12.1.b.

q. Primary Containment Structural Integrity, Specification 4.6.1.4.2.

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

RECORDS RETENTION (Continued)

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records or radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.1-1.

g. Records of reactor tests and experiments.

h. Records of training and qualification for current members of the plant staff.

i. Records of inservice inspections performed pursuant to these Technical Specifications.

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

RECORDS RETENTION (Continued)

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

l. Records of the service lives of all hydraulic and mechanical snubbers referenced in Section 3.7.5 including the data at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

n. Records of (1) meetings of the PNSC, (2) meetings of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC), (3) the independent reviews performed by the Corporate Nuclear Safety Section, and (4) the independent reviews performed by the Corporate Quality Assurance Audit Program, Performance Evaluation Unit.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 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 HIGH RADIATION AREA

6.12.1 In lieu of the "Control Device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

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* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 above shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Operations Shift Foreman on duty and/or the Radiation Control Supervisor.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1. Sufficiently detailed information to totally support rationale without benefit of additional of supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and,

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

PROCESS CONTROL PROGRAM (PCP) (Continued)

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2. A determination that the change did not reduce the overall conformance of the solidification waste product to existing criteria for solid wastes; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS 7/

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4. An evaluation of the change that shows the predicted release of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change that shows the expected maximum exposure to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

7/ Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

ADMINISTRATIVE CONTROLS

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable to the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.