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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

63

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company (PP&L) dated December 26, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 63, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 6, 1987

ENCLOSURE TO LICENSE AMENDMENT NO. 63

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 6-23	3/4 6-23 (overleaf)
3/4 6-24	3/4 6-24
3/4 6-25	3/4 6-25
3/4 6-26	3/4 6-26
3/4 7-9	3/4 7-9 (overleaf)
3/4 7-10	3/4 7-10
3/4 7-11	3/4 7-11
3/4 7-12	3/4 7-12 (overleaf)
3/4 7-13	3/4 7-13
3/4 7-14	3/4 7-14 (overleaf)
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4 (overleaf)
6-3	6-3
6-4	6-4 (overleaf)
6-7	6-7 (overleaf)
6-8	6-8
6-17	6-17
	6-17a
6-18	6-18 (overleaf)

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 0.2 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries per gram.

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

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant:

1. Greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

2. Greater than $100/\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Manual Isolation Valves (Continued)

RCIC Suction[b][c]

HV-149F031

RCIC Turbine Exhaust[b]

HV-149F059

RCIC Vacuum Pump Discharge[b]

HV-149F060

HPCI Injection

HV-155F006
1-55-038

RHR - Shutdown Cooling Return/
LPCI Injection

HV-151F015 A,B

RHR - Suppression Pool Suction[b][c]

HV-151F004 A,B,C,D

RHR Heat Exchanger Vent[c]

HV-151F103 A,B

CS Injection

HV-152F005 A,B
HV-152F037 A,B

CS Suction[b][c]

HV-152F001 A,B

Containment Instrument Gas

SV-12654 A,B

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Manual Isolation Valves (Continued)

SLCS[b]

HV-148F006

Demineralized Water

1-41-017
1-41-018

ILRT

1-57-193
1-57-194

HPCI Turbine Exhaust[b]

HV-155F066

RHR - Shutdown Cooling Return/
LPCI Injection - Pressure Equalizing Valve

HV-151F122 A,B

c. Other Valves

Feedwater

141F010 A,B

RHR - Shutdown Cooling Suction

PSV-151F126

RHR - Shutdown Cooling Return/
LPCI Injection

HV-151F050 A,B

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Valves (Continued)

RHR-Minimum Recirculation Flow(b)(c)

HV-151F007 A,B

RHR - Relief Valve Discharge(c)

PSV-151F055 A,B
PSV-15106 A,B
PSV-151F097

CS Injection

HV-152F006 A,B

CS Minimum Recirculation Flow(b)(c)

HV-152F031 A,B

Containment Instrument Gas

1-26-072
1-26-074
1-26-152
1-26-154
1-26-164

Recirculation Pump Seal Water

143F013 A,B
XV-143F017 A,B

TIP Shear Valves(d)

C51-J004 A,B,C,D,E

SLCS(b)

148F007

HPCI Turbine Exhaust(b)

155F049

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Valves (Continued)

HPCI Minimum Recirculation Flow(b)(c)

HV-155F012
HV-155F046

RCIC Turbine Exhaust(b)

149F040

RCIC Minimum Recirculation Flow(b)(c)

FV-149F019
HV-149F021

RCIC Vacuum Pump Discharge(b)

149F028

d. Excess Flow Check Valves

HPCI

XV-155F024 A,B,C,D

Core Spray

XV-152F018 A,B

RHR

XV-15109 A,B,C,D

RCIC

XV-149F044 A,B,C,D

RWCU

XV-14411 A,B,C,D
XV-144F046

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150, + 15, -0 psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.

4. Performing a CHANNEL CALIBRATION of the condensate transfer pump discharge low pressure alarm instrumentation and verifying the low pressure alarm setpoint to be greater than or equal to 113 psig.

d. In the event the RCIC system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) that attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of these visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Surveillance Requirements 4.7.4.d.

c. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of at least that number of snubbers which follows the expression $35\ (1 + \frac{c}{2})$ where c = 4, is the allowable number of snubbers not meeting the acceptance criteria selected by the operator, shall be functionally tested either in-place or in a bench test. For each number of snubbers above c which does not meet the functional test acceptance criteria of Specifications 4.7.4.d., an additional sample selected according to the expression $35\ (1 + \frac{c}{2})\ (\frac{2}{c + 1})^2\ (a - c)$ shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the representative sample.

Functional testing shall continue according to the expression $b\ [35\ (1 + \frac{c}{2})\ (\frac{2}{c + 1})^2]$ where b is the number of snubbers found inoperable in the previous re-sample, until no additional inoperable snubbers are found within a sample or until all snubbers have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle.
2. Each snubber within 5 feet of heavy equipment, valve, pump, turbine, motor, etc.
3. Each snubber within 10 feet of the discharge from a safety relief valve.

Functional Test (Continued)

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber, if it is repaired and installed in another position, and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For any snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. Drag force shall not have increased more than 50% since the last surveillance test.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

SURVEILLANCE REQUIREMENTS (Continued)

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination. 29

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcurie per test sample. 29

4.7.5.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3 (tritium), and 29

   2. In any form other than gas.

CHEMISTRY (Continued)

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence.

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 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting

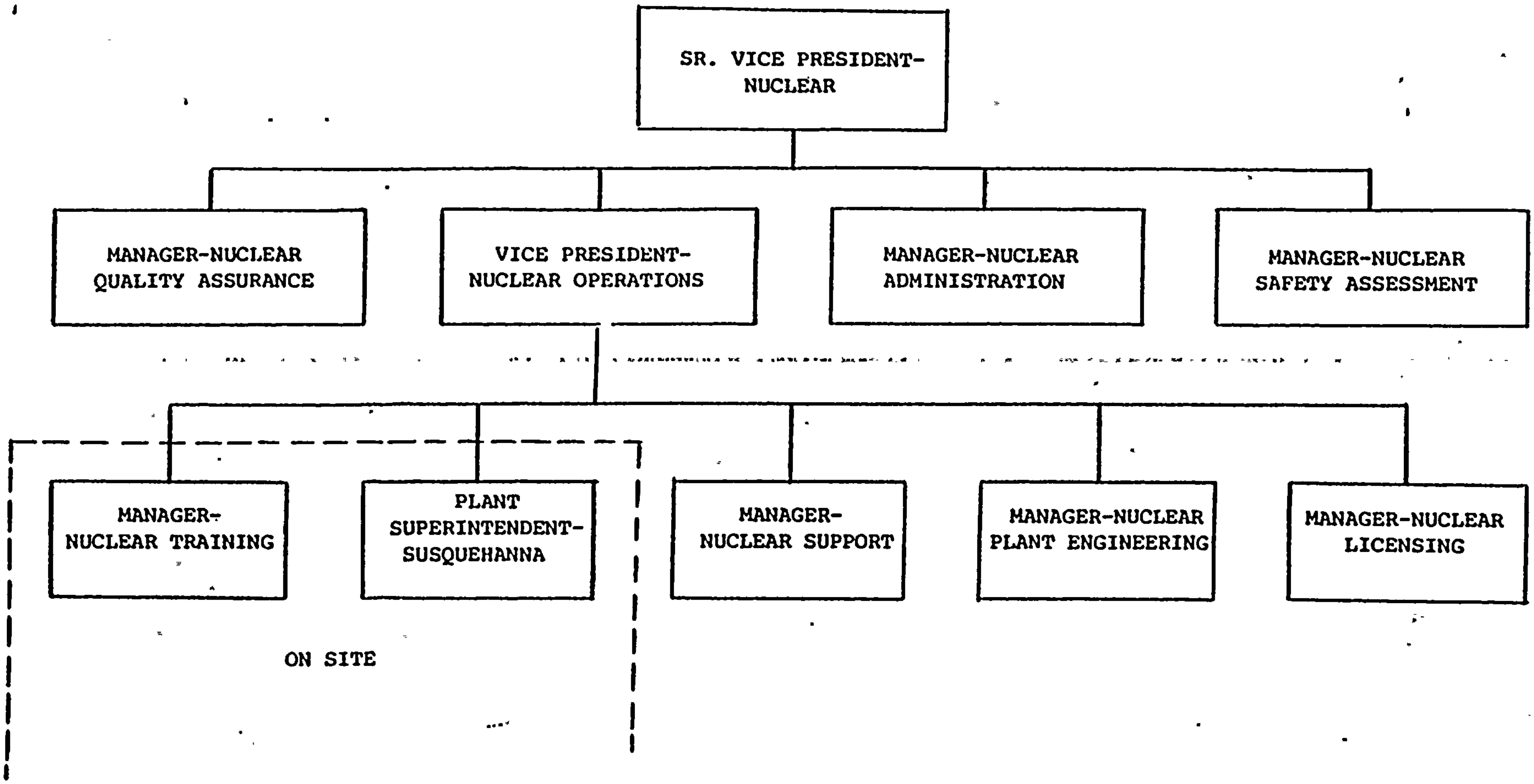

Figure 6.2.1-1

OFFSITE ORGANIZATION

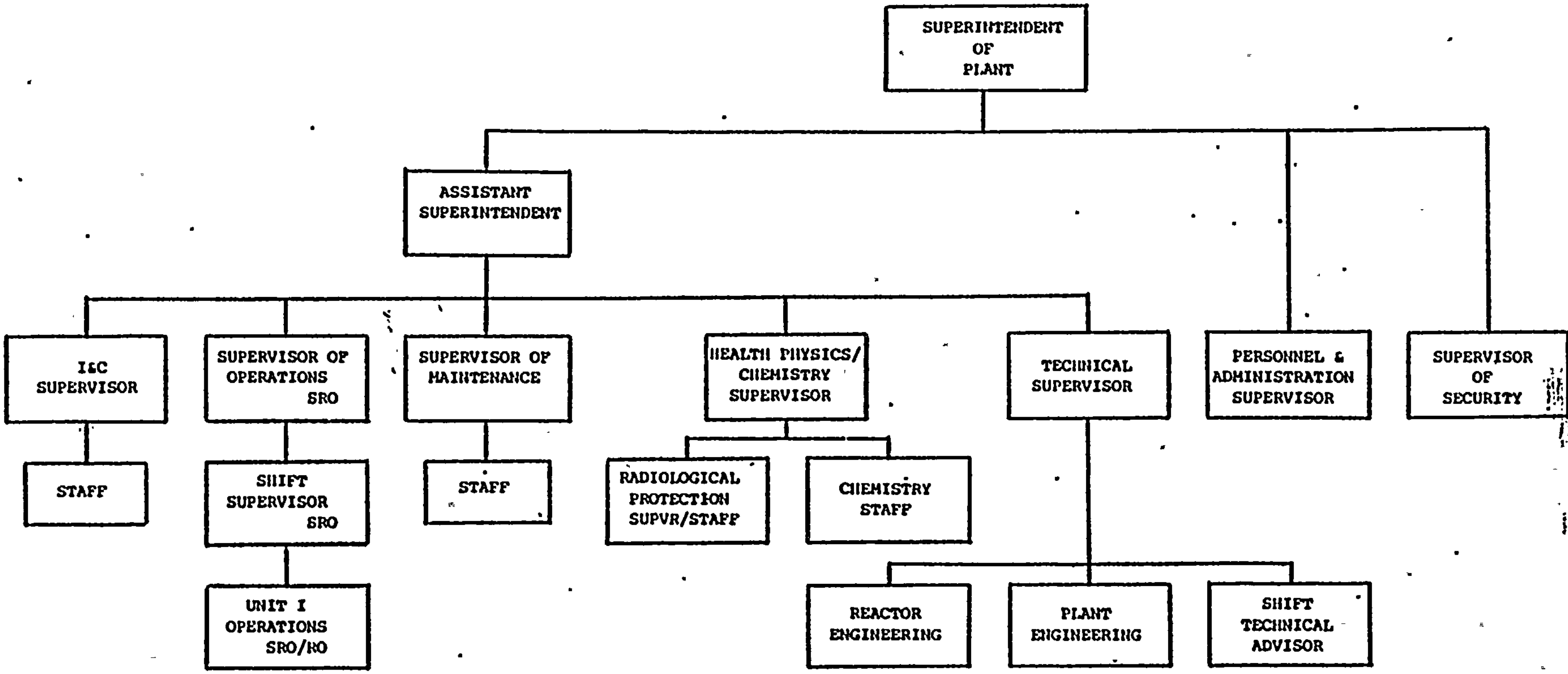

FIGURE 6.2.2-1
UNIT ORGANIZATION

6.2.3 NUCLEAR SAFETY ASSESSMENT GROUP (NSAG)

FUNCTION

6.2.3.1 The NSAG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The NSAG shall be composed of at least five dedicated, full-time engineers with at least three located onsite, each with a bachelor's degree in engineering or related science and at least two years professional level experience in his field, at least one year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The NSAG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The NSAG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Senior Vice President-Nuclear.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Radiological Protection Supervisor or Health Physics/Chemistry Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure 1 of the October 30, 1979 NRC letter to all operating nuclear power plants.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager - Nuclear Training, 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 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

*Not responsible for sign-off function.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Superintendent of Plant-Susquehanna on matters related to nuclear safety as described in Specification 6.5.1.6.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Superintendent of Plant-Susquehanna
Member:	Assistant Superintendent of Plant-Susquehanna
Member:	Assistant Superintendent - Outages
Member:	Supervisor of Operations
Member:	Technical Supervisor
Member:	Supervisor of Maintenance
Member:	I&C/Computer Supervisor
Member:	Reactor Engineering Supervisor or Unit Reactor Engineer
Member:	Health Physics/Chemistry Supervisor
Member:	Shift Supervisor or Unit Supervisor

ALTERNATES

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

MEETING FREQUENCY

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

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

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 unit.

6.9.1.2 The startup report shall address each of the startup tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation, supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL REPORTS (Continued)

The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentation and one other radioiodine isotope concentation in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 11 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days of when the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PORC.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine radiological environmental operating reports covering the 26 operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate), with preoperational studies, operational controls and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of all radiological environmental samples and of all environmental radiation measurements taken during the report period pursuant to the locations specified in the tables and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 3.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company (PP&L) dated December 26, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 34, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 6, 1987

ENCLOSURE TO LICENSE AMENDMENT NO. 34

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 6-25	3/4 6-25
3/4 6-26	3/4 6-26
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4 (overleaf)
6-3	6-3
6-4	6-4 (overleaf)
6-7	6-7 (overleaf)
6-8	6-8
6-17	6-17 (overleaf)
	6-17a
6-18	6-18 (overleaf)

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 0.2 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries per gram.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the specific activity of the primary coolant:

   1. Greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 more than 48 hours during one continuous time interval or greater than 4 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

   2. Greater than 100/$\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

b. In OPERATIONAL CONDITION 1, 2, 3, or 4, with the specific activity of the primary coolant greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

   1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

   2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

   3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Valves (Continued)

RHR - Relief Valve Discharge[c]

PSV-251F055 A,B
PSV-25106 A,B
PSV-251F097

CS Injection

HV-252F006 A,B

CS Minimum Recirculation Flow[b][c]

HV-252F031 A,B

Containment Instrument Gas

2-26-164
2-26-072
2-26-074
2-26-152
2-26-154

Recirculation Pump Seal Water

243F013 A,B
XV-243F017 A,B

TIP SHEAR VALVES[d]

C51-J004 A,B,C,D,E

SLCS[b]

248F007

HPCI Turbine Exhaust[b]

255F049

HPCI Minimum Recirculation Flow[b][c]

HV-255F012
HV-255F046

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Valves (Continued)

RCIC Turbine Exhaust[(b)]

249F040

RCIC Minimum Recirculation Flow[(b)(c)]

FV-249F019
HV-249F021

RCIC Vacuum Pump Discharge[(b)]

249F028

d. Excess Flow Check Valves

HPCI

XV-255F024 A,B,C,D

Core Spray

XV-252F018 A,B

RHR

XV-25109 A,B,C,D

RCIC

XV-249F044 A,B,C,D

RWCU

XV-24411 A,B,C,D
XV-244F046

CHEMISTRY (Continued)

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR Part 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1 includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence.

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 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 1.

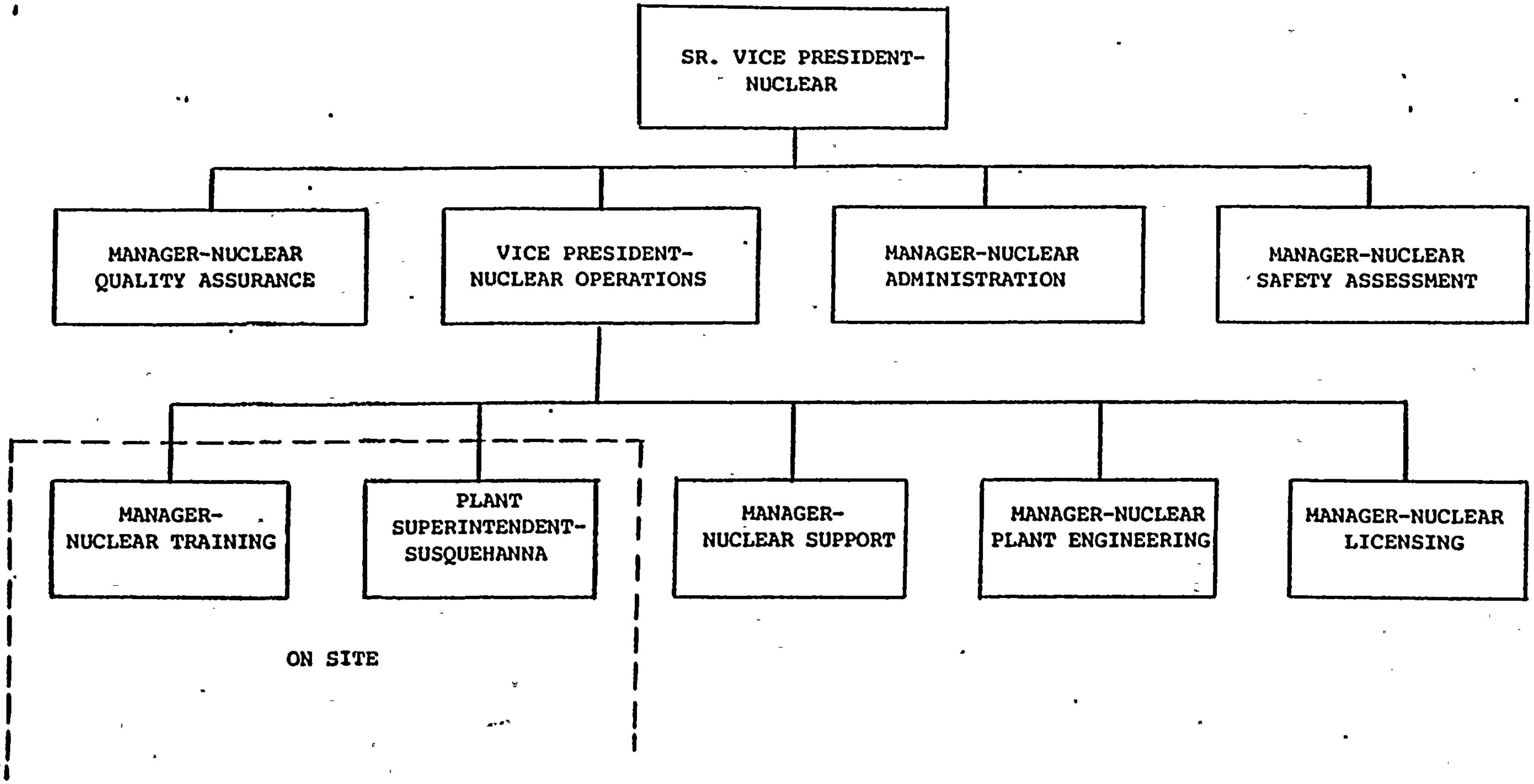

Figure 6.2.1-1

OFFSITE ORGANIZATION

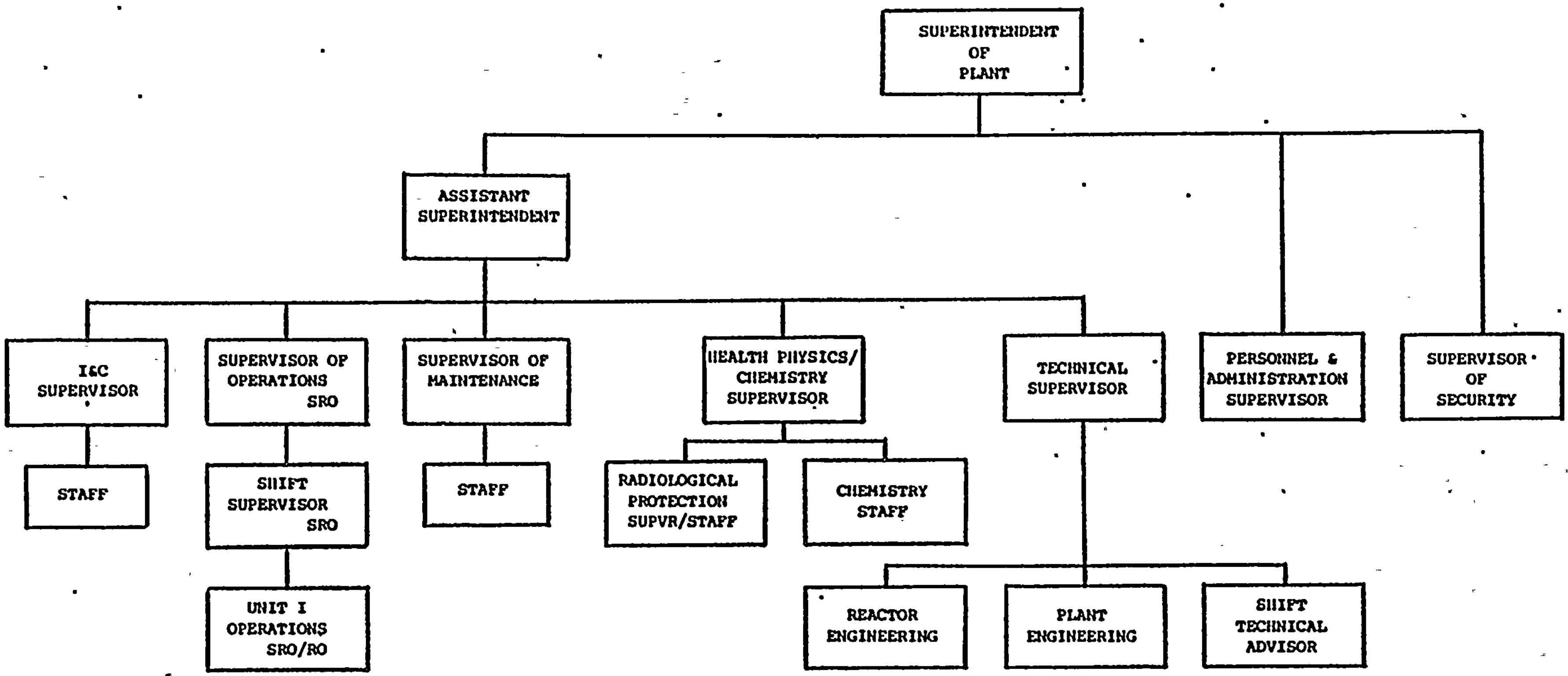

FIGURE 6.2.2-1
UNIT ORGANIZATION

6.2.3 NUCLEAR SAFETY ASSESSMENT GROUP (NSAG)

FUNCTION

6.2.3.1 The NSAG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The NSAG shall be composed of at least five dedicated, full-time engineers with at least three located onsite, each with a bachelor's degree in engineering or related science and at least two years professional level experience in his field, at least one year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The NSAG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The NSAG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Senior Vice President-Nuclear.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Radiological Protection Supervisor or Health Physics/Chemistry Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure 1 of the October 30, 1979 NRC letter to all operating nuclear power plants.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager - Nuclear Training, 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 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

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*Not responsible for sign-off function.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Superintendent of Plant-Susquehanna on matters related to nuclear safety as described in Specification 6.5.1.6.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Superintendent of Plant-Susquehanna
Member:	Assistant Superintendent of Plant-Susquehanna
Member:	Assistant Superintendent - Outages
Member:	Supervisor of Operations
Member:	Technical Supervisor
Member:	Supervisor of Maintenance
Member:	I&C/Computer Supervisor
Member:	Reactor Engineering Supervisor or Unit Reactor Engineer
Member:	Health Physics/Chemistry Supervisor
Member:	Shift Supervisor or Unit Supervisor

ALTERNATES

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

MEETING FREQUENCY

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

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

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 unit.

6.9.1.2 The startup report shall address each of the startup tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation, supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel, including contractors, receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL REPORTS (Continued)

b. The results of specific analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days of when the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PORC.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate), with preoperational studies, operational controls and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of all radiological environmental samples and of all environmental radiation measurements taken during the report period pursuant to the locations specified in the tables and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 4.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

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*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.