

3/4.11 RADIOACTIVE EFFLUENTS

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outdoor temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

- a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outdoor temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

* Tanks included in this Specification are those outdoor temporary tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times. *

ACTION:

- a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.
- b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously** monitoring the waste gases in the waste gas holdup system with the oxygen monitor. If hydrogen is not measured, the concentration of hydrogen shall be assumed to exceed 4% by volume.

* Not applicable to portions of the Waste Gas System removed from service for maintenance provided that, the portions removed for maintenance are isolated, and purged of hydrogen to less than 4% by volume.

** If the oxygen monitoring instrumentation is inoperable, operation of the waste gas holdup system may continue provided grab samples are collected in accordance with the Surveillance Frequency Control Program and analyzed within the following 4 hours.

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SALEM - UNIT 2

3/4 12-1

Amendment No. 215

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SECTION 5.0
DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE LOCATION

Salem Generating Station is located in Salem County, New Jersey along the eastern shore of the Delaware River approximately 8 miles southwest of Salem, New Jersey and 18 miles south of Wilmington, Delaware.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

- a. Nominal inside diameter = 140 feet.
- b. Nominal inside height = 210 feet.
- c. Minimum thickness of concrete walls = 4.5 feet.
- d. Minimum thickness of concrete roof = 3.5 feet.
- e. Minimum thickness of concrete floor mat = 16 feet.
- f. Nominal thickness of steel liner = 1/4 to 1/2 inch.
- g. Net free volume = 2.62×10^6 cubic feet

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DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

- 5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

- 5.3.1 The reactor core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

- 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

- 5.4.1 The reactor coolant system is designed and shall be maintained:
- In accordance with the code requirement specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
 - For a pressure of 2485 psig, and
 - For a temperature of 650°F, except for the pressurizer which is 680°F.

DESIGN FEATURES

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5.6 FUEL STORAGE

CRITICALITY

- 5.6.1.1 The new fuel storage racks are designed and shall be maintained with:
- A maximum K_{eff} equivalent of equal to 0.95 with the storage racks flooded with unborated water.
 - A nominal 21.0 inch center-to-center distance between fuel assemblies.
 - Unirradiated fuel assemblies with enrichments less than or equal to 4.25 weight percent (w/o) U-235 with no requirements for Integral Fuel Burnable Absorber (IFBA) pins.
 - Unirradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235 which contain a minimum number of Integral Fuel Burnable Absorber (IFBA) pins. This minimum number of IFBA pins shall have an equivalent reactivity hold-down which is greater than or equal to the reactivity hold down associated with N IFBA pins, at a nominal 2.35 mg B-10/linear inch loading (1.5X), determined by the equation below:

$$N = 42.67 (E - 4.25)$$

- 5.6.1.2 The spent fuel storage racks are designed and shall be maintained with:
- A maximum K_{eff} equivalent of 0.95 with the storage racks filled with unborated water.
 - A nominal 10.5 inch center-to-center distance between fuel assemblies stored in Region 1 (flux trap type) racks.
 - A nominal 9.05 inch center-to-center distance between fuel assemblies stored in Region 2 (non-flux trap) racks.

DESIGN FEATURES

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DESIGN FEATURES

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 124'8".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1632 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

<u>COMPONENT</u>	<u>CYCLIC OR TRANSIENT LIMIT</u>	<u>DESIGN CYCLE OR TRANSIENT</u>
Reactor Coolant System	200 heatup cycles at $\leq 100^\circ\text{F/hr}$ and 200 cooldown cycles at $\leq 100^\circ\text{F/hr}$ (pressurizer cooldown at $\leq 200^\circ\text{F/hr}$).	Heatup cycle - T_{avg} from $\leq 200^\circ\text{F}$ to $> 542^\circ\text{F}$. Cooldown cycle - T_{avg} from $\geq 542^\circ\text{F}$ to $\leq 200^\circ\text{F}$.
	80 loss of load cycles.	Without immediate turbine or reactor trip.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	200 large step decreases in load.	50% of RATED THERMAL POWER step load decrease with steam dump.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

<u>COMPONENT</u>	<u>CYCLIC OR TRANSIENT LIMIT</u>	<u>DESIGN CYCLE OR TRANSIENT</u>
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe > 13.5 inches equivalent diameter.
	Operating Basis Earthquake	50 cycles
	Design Basis Earthquake	10 cycles; 0.20g horizontal, 0.136g vertical.
	50 leak tests.	Pressurized to \geq 2485 psig.
Secondary System	5 hydrostatic pressure tests	Pressurized to \geq 3107 psig.
	1 steam line break	Break in a steam line > 6 inches equivalent diameter.
	5 hydrostatic pressure tests	Pressurized to \geq 1356 psig.
	10 turbine roll tests	Turbine roll on pump heat resulting in plant cooldown > 100°F/hr.

SECTION 6.0
ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Senior Nuclear Shift Supervisor or, during his absence from the Control Room, a designated individual shall be responsible for the Control Room command function. A management directive to this effect, signed by the senior corporate nuclear officer, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 ONSITE AND OFFSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include positions for activities affecting the safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Salem Updated Final Safety Analysis Report and updated in accordance with 10 CFR 50.71(e).
- b. The plant manager shall be responsible for overall facility safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plants.
- c. The senior corporate nuclear officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plants to ensure nuclear safety.
- d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 FACILITY STAFF

The Facility organization shall be subject to the following:

- a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

ADMINISTRATIVE CONTROLS

FACILITY STAFF (Continued)

- b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, at least one licensed Senior Reactor Operator shall be in the Control Room area at all times.
- c. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator who has no other concurrent responsibilities during this operation.

Figure 6.2-1 CORPORATE HEADQUARTERS AND OFF-SITE ORGANIZATION FOR
MANAGEMENT AND TECHNICAL SUPPORT

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FIGURE 6.2-2 FACILITY ORGANIZATION

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SALEM UNIT 2

WITH UNIT 1 IN MODES 5 OR 6 OR DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SNSS	1 ^a	1 ^a
SRO	1 ^b	none
STA	1 ^b	none
NCO	2	1
EO/UO	3	2 ^c
Maintenance Electrician	1	none
Rad. Pro. Technician	1 ^a	1 ^{a,e}

WITH UNIT 1 IN MODES 1, 2, 3 OR 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SNSS	1 ^a	1 ^a
SRO	1 ^b	none
STA	1 ^b	none
NCO	2	1
EO/UO	3 ^d	1
Maintenance Electrician	1 ^a	none
Rad. Pro. Technician	1 ^a	1 ^a

- a/ Individual may fill the same position on Unit 1.
- b/ Individual who fulfills the STA requirement may fill the same position on Unit 1. The STA, if a licensed SRO, may concurrently fill the SRO position on one unit; the other unit also requires a qualified SRO on shift.
- c/ One of the two required individuals may fill the position on Unit 1, such that there are a total of three EO/UO's for both units.
- d/ One of the three required individuals may fill the same position of Unit 1, such that there are a total of five EO/UO's for both units.
- e/ Not needed if both reactors are de-fueled.

TABLE 6.2-1 (Continued)

- SNSS - Senior Nuclear Shift Supervisor with a Senior Reactor Operator License on both units.
- SRO - Individual with a Senior Reactor Operator License on both units (normally, a Nuclear Shift Supervisor).
- NCO - Nuclear Control Operator with a Reactor Operator License on both units.
- STA - Shift Technical Advisor (if licensed as SRO, may be assigned duties as a Nuclear Shift Supervisor).
- EO/VO - Equipment Operator or Utility Operator.

Except for the Senior Nuclear Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate the unexpected absence of on-duty shift crew members provided that immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewperson's being late or absent.

During any absence of the Senior Nuclear Shift Supervisor from the Control Room area while the unit is in any MODE, an individual with a valid SRO License shall be designated to assume the Control Room command function.

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6.2.3 SHIFT TECHNICAL ADVISOR

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

6.2.3.2 The Shift Technical Advisor 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 for transients and accidents.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the PSEG Nuclear Quality Assurance Topical Report.

6.3.2 The Operations Manager or Assistant Operations Manager shall hold an SRO license. The Senior Nuclear Shift Supervisors and Nuclear Shift Supervisors shall each hold a senior reactor operator license. The Nuclear Control Operators shall hold reactor operator licenses.

6.3.3 The Operations Manager shall meet one of the following:

- 1) Hold an SRO license, or
- 2) Have held an SRO license for a similar unit (PWR), or
- 3) Have been certified at an appropriate simulator for equivalent senior operator knowledge.

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6.5 REVIEW AND AUDIT (THIS SECTION DELETED)

ADMINISTRATIVE CONTROLS

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ADMINISTRATIVE CONTROLS

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6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

- a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.73 to 10CFR Part 50, and
- b. Each REPORTABLE EVENT shall be reviewed by the Station Operations Review Committee (SORC) and the resultant Licensee Event Report submitted to the Nuclear Review Board and the senior corporate nuclear officer.

6.7 SAFETY LIMIT VIOLATION

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

- a. The unit shall be placed in at least HOT STANDBY within one hour.
- b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The senior corporate nuclear officer and senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight shall be notified within 24 hours.
- c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. 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 senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight and the senior corporate nuclear officer within 14 days of the violation.

ADMINISTRATIVE CONTROLS

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6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
- b. Refueling operations.
- c. Surveillance and test activities of safety related equipment.
- d. Security Plan implementation.
- e. Emergency Plan implementation.
- f. Fire Protection Program implementation.
- g. PROCESS CONTROL PROGRAM implementation.
- h. OFFSITE DOSE CALCULATION MANUAL implementation.
- i. Quality Assurance Program for effluent and environmental monitoring.

6.8.2 Each procedure and administrative policy of 6.8.1 above, except 6.8.1.d and 6.8.1.e, and changes thereto, shall be reviewed and approved in accordance with requirements in Updated Final Safety Analysis Report (UFSAR) section 17.2 for SORC or for Technical Review and Control, as appropriate, prior to implementation and reviewed periodically as set forth in administrative procedures. Procedures of 6.8.1.d and 6.8.1.e shall be reviewed and approved in accordance with the Facility's Security and Emergency Plans or requirements in Updated Final Safety Analysis Report (UFSAR) section 17.2 for Technical Review and Control, as appropriate, prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 On-the-spot changes to procedures of 6.8.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented and receives the same level of review and approval as the original procedure under within 14 days of implementation.

ADMINISTRATIVE CONTROLS

6.8.4 The following programs shall be maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include (recirculation spray, safety injection, chemical and volume control, gas stripper, recombiners, ...). The program shall include the following:

- (i) Preventative maintenance and periodic visual inspection requirements, and
- (ii) 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 areas under accident conditions. This program shall include the following:

- (i) Training of personnel,
- (ii) Procedures for monitoring, and
- (iii) Provisions for maintenance of sampling and analyses equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

- (i) Identification of a sampling schedule for the critical variables and the control points for these variables,
- (ii) Identification of the procedures used to measure the values of the critical variables,
- (iii) Identification of process sampling points, including monitoring at the discharge of the condensate pumps for evidence of condenser in-leakage.
- (iv) Procedures for the recording and management of data,
- (v) Procedures defining corrective actions for off-control-point chemistry conditions,

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- (vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

- d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

- (i) Training of personnel, and
- (ii) Procedures for monitoring

- e. Deleted

- 6.8.4.f. Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 47.0 psig.

The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

- a. Primary containment leakage rate acceptance criterion is less than or equal to $1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate

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acceptance criteria are less than or equal to 0.6 La for Type B and Type C tests and less than or equal to 0.75 La for Type A tests;

- b. Air lock testing acceptance criteria are:
- 1) Overall air lock leakage rate is less than or equal to 0.05 La when tested at greater than or equal to Pa,
 - 2) Seal leakage rate less than or equal to 0.01 La per hour when the gap between the door seals is pressurized to 10.0 psig.

Test frequencies and applicable extensions will be controlled by the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 will be applied to the Primary Containment Leakage Rate Testing Program.

6.8.4.g Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to the members of the public from radioactive effluents as low as reasonable achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,
- 2) Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR 20, Appendix B, Table II, Column 2,
- 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,
- 4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas conforming to Appendix I to 10 CFR Part 50,
- 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.
- 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 92-day period would exceed a suitable fraction of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

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- 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,
- 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,
- 9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,
- 10) Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.4.h Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

- 1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,
- 2) A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of the census, and
- 3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.4.i Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

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- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.
 1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
 2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gallon per minute per SG.
 3. The operational leakage performance criterion is specified in LCO 3.4.7.2, “Reactor Coolant System Operational Leakage.”
- c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3, below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.
 2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
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- e. Provisions for monitoring operational primary-to-secondary leakage.

ADMINISTRATIVE CONTROLS

6.8.4.j Deleted

6.8.4.k Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of the ISI Program, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

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6.8.4.l Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

6.8.4.m Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

- (i) This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.
- (ii) The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.
- (iii) The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants" in lieu of Section XI of the ASME BPV Code ISI requirements for snubbers, or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).
- (iv) The 120-month program updates shall be made in accordance with 10 CFR 50.55a (including 10 CFR 50.55a(b)(3)(v)) subject to the limitations and modifications listed therein.

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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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the USNRC Administrator, Region I, unless otherwise noted.

STARTUP REPORT

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 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.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of 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 power 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.

* 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.

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6.9.1.5 Reports required on an annual basis shall include:

- a. DELETED
- b. DELETED
- c. The results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.9. 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 the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the 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 REPORT

6.9.1.6 DELETED

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ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objective outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR 50.

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

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6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
 1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
 2. Control Bank Insertion Limits for Specification 3/4.1.3.5,
 3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,
 4. Heat Flux Hot Channel Factor, F_Q , its variation with core height, $K(z)$, and Power Factor Multiplier PF_{xy} , Specification 3/4.2.2, and
 5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.
 6. Refueling boron concentration per Specification 3.9.1

- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a.
 2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference
 3. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.
 4. WCAP-10266-P-A, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.
 5. WCAP-12472-P-A, BEACON – Core Monitoring and Operations Support System, (W Proprietary).
 6. CENPD-397-P-A, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology
 7. WCAP-10054-P-A, Addendum 2, “Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model.”

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8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, Optimized ZIRLO™, (W Proprietary)
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any mid-cycle revisions or supplements shall be provided upon issuance for each reload cycle to the NRC.

6.9.1.10 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.8.4.i, "Steam Generator (SG) Program." The report shall include:

- a. The scope of inspections performed on each SG;
- b. Then nondestructive examination techniques utilized for tubes with increased degradation susceptibility;
- c. For each degradation mechanism found:
 1. The nondestructive examination techniques utilized;
 2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

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3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;
4. The number of tubes plugged during the inspection outage; and
- d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results.
- e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;
- f. The results of any SG secondary side inspections.

6.9.1.11 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, POPS enable temperature, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
 1. Specification 3.1.2.3, "Charging Pump-Shutdown"
 2. Specification 3.4.1.3, "Reactor Coolant System Hot Shutdown"
 3. Specification 3.4.1.4, "Reactor Coolant System Cold Shutdown"
 4. Specification 3/4.4.10.1, "Reactor Coolant System Pressure/Temperature Limits"
 5. Specification 3.4.10.3, "Reactor Coolant System Overpressure Protection Systems"
 6. Specification 3/4.5.3, "Emergency Core Cooling Systems ECCS Subsystems - Tavg <350°F"
- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC.
 1. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May, 2004.
 2. WCAP-18124-NP-A, Rev 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018, may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 1, 4, 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

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6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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

- a. Records and logs of unit operation covering time interval at each power level.
- 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 submitted to the Commission.
- d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
- e. DELETED
- f. Records of changes made to Operating Procedures required by Specification 6.8.1.
- g. Records of radioactive shipments.
- h. Records of sealed source and fission detector leak tests and results.
- i. Records of annual physical inventory of all sealed source material of record.
- j. Records of reviews performed for changes made to procedures or reviews of tests and experiments, pursuant to 10CFR50.59.

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

- a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report, pursuant to 10CFR50.59.
- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

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- c. Records of radiation exposure for all individuals entering radiation control areas.
- d. Records of gaseous and liquid radioactive material released to the environs.
- e. Records of transient or operational cycles for those facility components identified in Table 5.7-1.
- f. Records of reactor tests and experiments.
- g. Records of training and qualification for current members of the plant staff.
- h. Records of in-service inspections.
- i. Records of Quality Assurance activities required by the QA Manual.
- j. DELETED
- k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff).
- l. Records for Environmental Qualification which are covered under the provisions of Paragraph 2.C(7) and 2.C(8) of Facility Operating License DPR-75.
- m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.
- n. Records of secondary water sampling and water quality.
- o. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.
- p. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

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

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6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

- 6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation
- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
 - b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
 - c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
 - d. Each individual or group entering such an area shall possess:
 1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or
 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
 - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

- (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
 - 1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.
 - 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
 - 1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
 - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
 - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.
 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.
- f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Changes to the PCP:

1. Shall be documented and records of review performed shall be retained as required by Specification 6.10.3p. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after review and acceptance by the SORC and the approval of the Plant Manager.

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6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3p. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

b) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the SORC and the approval of the Plant Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g. month/year) the change was implemented.

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

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

1. Shall be reported to the Commission in the UFSAR for the period in which the evaluation was reviewed by (SORC). The discussion of each change shall contain:

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR50.59;

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

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- c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;
 - d. An evaluation of the change, which shows the predicted releases 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;
 - e. An evaluation of the change, which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;
 - f. 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;
 - g. An estimate of the exposure to plant operating personnel as a result of the change; and
 - h. Documentation of the fact that the change was reviewed and found acceptable by the (SORC).
2. Shall become effective upon review and acceptance by the SORC.

6.16 TECHNICAL SPECIFICATION (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. PSEG may make changes to the Bases without prior NRC approval provided the changes do not require either of the following:
 1. A change in the TS incorporated in the License, or
 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. Proposed changes to the Bases that require either condition of Specification 6.16.b above shall be reviewed and approved by the NRC prior to implementation.
- d. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).
- e. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

6.17 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Conditioning System (CREACS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.
- d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the Surveillance Requirements, at a frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.
- e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
- f. The provisions of Surveillance Requirement 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

APPENDIX B

TO

FACILITY OPERATING LICENSE NO. DPR-70
SALEM GENERATING STATION UNIT 1
DOCKET NO. 50-272

AND

FACILITY OPERATING LICENSE NO. DPR-75
SALEM GENERATING STATION UNIT 2
DOCKET NO. 50-311

PSEG NUCLEAR LLC

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

SALEM GENERATING STATION
UNIT NOS. 1 AND 2

ENVIRONMENTAL PROTECTION PLAN
(NONRADIOLOGICAL)

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Amendment No. 77

AUG 21 1989

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of nonradiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

- (1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating Licensing Stage (FES-OL) and other NRC environmental impact assessments.
- (2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.
- (3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NUPDES permit.

2.0 Environmental Protection Issues

In the FES, dated April 1973, the staff considered the environmental impacts associated with the operation of Salem Generating Station Unit Nos. 1 and 2. Certain environmental issues were identified which required study or license conditions to resolve and to assure adequate protection of the environment. The Appendix B Environmental Technical Specifications (ETS) issued with the operating license included discharge restrictions and monitoring programs related to aquatic and terrestrial resources.

2.1 Aquatic Issues

Requirements for study of station intake and discharges effects were removed from the EIS by License Amendments 51 (Unit 1) and 18 Unit 2, dated March 14, 1983 and March 11, 1983, respectively. These issues now are addressed by the effluent limitations and monitoring requirements contained in the effective NPDES Permit No. NJ0005622 issued by the State of New Jersey, and by the determination of the State of New Jersey on the Section 316(a) and (b) demonstration submitted by licensee. The NRC will rely on the State for regulation of matters involving water quality and aquatic biota.

2.2 Terrestrial Issues

Requirements for study of station effects on terrapins and raptors have been met.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in station design or operation performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological environmental effects are confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

* This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) as significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0.

3.2 Reporting Related to the NJPDES Permit or the State Certification

The NRC shall be provided with a copy of the current NJPDES permit or State certification within 30 days of approval. Changes to the NJPDES permit or State certification shall be reported to the NRC within 30 days of the date the change is approved.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. If an event is reportable under 10 CFR 50.72, then a duplicate immediate report under this Subsection is not required. However, a written report is required in accordance with Section 5.4.2.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The Nuclear Regulatory Commission (NRC) will rely on the decisions made by the State of New Jersey under the authority of the Clean Water Act and, in the case of threatened or endangered species, decisions made by the National Marine Fisheries Service (NMFS) under the authority of the Endangered Species Act, for any requirements pertaining to aquatic monitoring.

In accordance with Section 7(a) of the Endangered Species Act, the National Marine Fisheries Service has issued multiple Section 7 Consultation Biological Opinions related to the operation of Salem Unit 1 and 2 Generating Stations concluding that "...continued operation is not likely to jeopardize the continued existence of listed species."

PSEG Nuclear LLC shall adhere to the specific requirements within the currently applicable Incidental Take Statement, to the Biological Opinion. Changes to the incidental take statement must be preceded by consultation between the NRC, as the authorizing agency, and NMFS.

4.2.2 Terrestrial Monitoring

Terrestrial monitoring is not required.

5.0 Administrative Procedures

5.1 Review

The licensee shall provide for review of compliance with the EPP. The review shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review function and results of the review activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

5.4 Plant Reporting Requirements

5.4.1 Deleted.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics; (b) describe the probable cause of the event; (c) indicate the action taken to correct the reported event; (d) indicate the corrective action taken to

preclude repetition of the event and to prevent similar occurrences involving similar components or systems; and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided with a copy of such report at the same time it is submitted to the other agency.

APPENDIX C
ADDITIONAL CONDITIONS
OPERATING LICENSE NO. DPR-75

PSEG Nuclear LLC and Constellation Energy Generation, LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
175	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 21, 1997.
177	The licensee is authorized to upgrade the initiation circuitry for the power operated relief valves, as described in the licensee's application dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented prior to entry into Mode 3 from the current outage for Salem Unit 2.
179	Containment Fan Cooler Units The licensee shall complete all modifications associated with the amendment request concerning Containment Fan Cooler Units (CFCU) response time dated October 25, 1996, as described in the letters supplementing the amendment request dated December 11, 1996, January 28, March 27, April 24, June 3, and June 12, 1997, prior to entry into Mode 3 following refueling outage 12. All modifications made in support of this amendment request and described in the referenced submittals shall be in conformance with the existing design basis for Salem Unit 1, and programmatic controls for tank monitoring instrumentation shall be as described in the letter dated April 24, 1997. Post modification testing and confirmatory analyses shall be as described in the letter dated March 27, 1997. Future changes to the design described in these submittals may be made in accordance with the provisions of 10 CFR 50.59. Further, the administrative controls associated with CFCU operability and containment integrity described in the letters dated March 27, and April 24, 1997 shall not be relaxed or changed without prior staff review until such time as the license has been amended to include the administrative controls as technical specification requirements.	The amendment shall be implemented prior to entry into Mode 3 from the current outage for Salem Unit 2.
181	The licensee shall perform an evaluation of the containment liner anchorage by November 30, 1997, for the loading induced on the containment liner during a Main Steam Line Break event to confirm the assumptions provided in the Preliminary Safety Analysis Report and Updated Final Safety Analysis Report.	The amendment shall be implemented within 30 days from July 17, 1997.