Docket No. 50-213
CY-97-006

Attachment 1

Haddam Neck Plant

Proposed Revision To Operating License And Technical Specifications

Defueled Operating License And Technical Specifications

Marked-Up Pages

May 1997

CONNECTICUT YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-213

HADDAM NECK PLANT

FACILITY OPERATING LICENSE

License No. DPR-61

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license, as amended, filed by Connecticut Yankee Atomic Power Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Haddam Neck Plant (facility) has been substantially completed in conformity with Construction Permit No. CPPR-14 and the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

(possession only) JUL 1 6 1996

(3) CYAPCO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) CYAPCO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) CYAPCO, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear material as may be produced by the operations of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is not authorized to operate the reactor. Fuel may not be placed in the reactor vessel.

~~The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 1825 megawatts (thermal).~~

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~190~~, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(1)

[(3) Deleted ~~per Amdt. 29, 10-24-78.]~~ by Amendment No. 29

(4) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER, dated October 3, 1978, and Supplements dated February 6, 1981, November 11, 1981, November 14, 1984, November 27, 1987, January 20, 1990, April 10, 1990, August 14, 1990, June 27, 1991, October 16, 1991, November 21, 1991, and February 1 , 1995, subject to the following provisions.

The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment which could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

~~The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

(5) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Haddam Neck Plant Physical Security Plan," with revisions submitted through January 24, 1989; "Haddam Neck Plant Guard Training and Qualification Plan," with revisions submitted through January 27, 1983; and "Haddam Neck Plant Safeguards Contingency Plan," with revisions submitted through December 9, 1983. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

(6) Integrated Implementation Schedule

a. The Connecticut Yankee Atomic Power Company shall implement and maintain in effect the Integrated Implementation Schedule Program Plan (the Program Plan) to be followed for scheduling of plant modifications and engineering studies. The Program Plan shall be followed from and after the effective date of this license condition.

b. This license condition shall be effective for three years from the date of issuance of Amendment No. 183. (Date of Issuance February 23 , 1995)

(7) Fuel Movement

The movement of special nuclear material used as reactor fuel into the containment is prohibited.

D. This license is effective as of the date of issuance and shall expire at midnight, June 29, 2007.

FOR THE ATOMIC ENERGY COMMISSION

A. Giambusso, Deputy Director for
Reactor Projects
Directorate of Licensing

Original Signed by A. Giambusso

Enclosure:
Appendices A and B - Technical Specifications

Date of Issuance: December 27, 1974

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. DPR-61 is in accordance with Appendix D to 10 CFR Part 50 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70.

2. Facility Operating License No. DPR-61 is hereby issued to the Connecticut Yankee Atomic Power Company to read as follows:

A. This license applies to the Haddam Neck Plant, a pressurized lightwater reactor and associated equipment (the facility) owned by the Connecticut Yankee Atomic Power Company. The facility is located on Connecticut Yankee's Haddam Neck site on the east bank of the Connecticut River, approximately 21 miles south-southeast of Hartford, in the Town of Haddam, Middlesex County, Connecticut, and is described in the "Facility Description and Safety Analysis Report" as supplemented and amended (Amendment 10 through Amendment 25 to the License Application) and the "Environmental Report" (as supplemented by Amendment 1).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Connecticut Yankee Atomic Power Company:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in the Town of Haddam, Middlesex County, Connecticut, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Facility Description and Safety analysis as supplemented and amended; or as described in any amendment to this license;

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DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

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DEFINITIONS

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~~DEFINITIONS~~

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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ADMINISTRATIVE CONTROLS

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ANALOG CHANNEL OPERATIONAL TEST

1.2 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

~~AXIAL OFFSET~~ DELETED

1.3 ~~AXIAL OFFSET shall be the ratio of the difference in upper and lower detector signals, to the sum of upper and lower detector signals, as measured by the excore instruments.~~

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions, and may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

~~CONTAINMENT INTEGRITY~~ DELETED

1.6 ~~CONTAINMENT INTEGRITY shall exist when:~~

~~a. All penetrations required to be closed during accident conditions are either:~~

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as noted below:

Note 1) Normally-closed isolation valves SS-SOV-150A, SS-SOV-150B, SS-SOV-150C, SS-SOV-150D, SS-SOV-151A, SS-SOV-151B, SS-SOV-151C, and SS-SOV-151D which fail closed on loss of power and are capable of being closed within 60 seconds of a containment isolation actuation signal (CIAS) by an operator utilizing normal control switches and normal position indication within the main control room may be opened for periodic testing.

Note 2) Normally-closed manual isolation valves SI-V-863A, B, C, and D, SA-V-413, and SS-V-999A may be opened for periodic surveillance and containment boundary (vent and drain) manual valves may be opened for diagnostic checks to ensure Technical Specification limits or to ensure system operability are maintained. While these valves are open, a locally stationed operator will be in direct communication with the main control room. This ensures the valves are capable of being closed within 60 seconds of a CIAS.

b. The equipment hatch is closed and sealed,

c. The air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

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

~~CONTROLLED LEAKAGE~~ DELETED

1.7 ~~CONTROLLED LEAKAGE shall be that seal water flow returned from the reactor coolant pump number 2 seals.~~

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe position.

~~DOSE EQUIVALENT I-131~~ DELETED

1.9 ~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of U. S. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluent for the Purpose of Evaluating Compliance with 10CFR50, Appendix I," Revision 1, October 1977.~~

~~$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY~~ DELETED

1.10 ~~$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for isotopes, other than iodines and tritium, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity.~~

~~END-OF-CORE LIFE~~ DELETED

1.11 ~~END-OF-CORE LIFE shall correspond to a reactor operating condition with all control rod banks fully withdrawn, essentially 0 ppm boron concentration in the reactor coolant, and the Reactor Coolant System average temperature (Tavg) no longer maintained at normal operating temperature or the normal rated thermal power (RTP) no longer maintained.~~

FREQUENCY NOTATION

1.12 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

~~IDENTIFIED LEAKAGE~~ DELETED

1.13 ~~IDENTIFIED LEAKAGE shall be:~~

~~a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or~~

~~b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or~~

~~c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.~~

~~LINEAR HEAT GENERATION RATE~~ (DELETED)

1.14 ~~LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.~~

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

~~NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR~~ (DELETED)

1.16 ~~NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR,~~ $\cancel{F^N_{\Delta H}}$, ~~shall be the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.~~

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

~~PHYSICS TESTS~~ (DELETED)

1.19 ~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core, (1) described in the FSAR, (2) authorized under the provisions of 10CFR50.59, or (3) otherwise approved by the Commission.~~

~~PRESSURE BOUNDARY LEAKAGE~~ (DELETED)

1.20 ~~PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.~~

~~PURGE-PURGING~~ (DELETED)

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

~~QUADRANT POWER TILT RATIO~~ (DELETED)

1.22 ~~QUADRANT POWER TILT RATIO shall be the ratio of the maximum quadrant power to the average quadrant power as determined by the excore detectors.~~

RADIOACTIVE WASTE TREATMENT~~S~~ SYSTEMS

1.23 RADIOACTIVE WASTE TREATMENT SYSTEMS are those liquid, gaseous and solid waste systems which are required to maintain control over radioactive material in order to meet the LCO's set forth in those specifications.

RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

1.24 A RADIOLOGICAL EFFLUENT MONITORING MANUAL (REMM) shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures to individuals from station operation. An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation Alarm/Trip Setpoints. Requirements of the REMODCM are provided in Specification 6.13.

~~RATED THERMAL POWER~~ (DELETED)

1.25 ~~RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 1825 MWt.~~

REPORTABLE EVENT

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

RHR LOOP

1.27 An RHR LOOP consists of a Residual Heat Removal (RHR) pump, a dedicated RHR heat exchanger either from the same train or from the opposite train and all other necessary piping and components required to receive and cool reactor coolant.

~~SHUTDOWN MARGIN~~ (DELETED)

1.28 ~~SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical, or would be subcritical from its present condition assuming all rod cluster assemblies are fully inserted except for the single rod cluster of highest reactivity worth which is assumed to be fully withdrawn.~~

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

~~STAGGERED TEST BASIS~~ DELETED

1.31 ~~A STAGGERED TEST BASIS shall consist of:~~

~~a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and~~

~~b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.~~

~~THERMAL POWER~~ DELETED

1.32 ~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~

~~TRIP ACTUATING DEVICE OPERATIONAL TEST~~ DELETED

1.33 ~~A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.~~

~~UNIDENTIFIED LEAKAGE~~ DELETED

1.34 ~~UNIDENTIFIED LEAKAGE shall be all leakage which is neither IDENTIFIED LEAKAGE nor CONTROLLED LEAKAGE.~~

~~VENTING~~ DELETED

1.35 ~~VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~

~~TECHNICAL REPORT SUPPORTING CYCLE OPERATION~~ DELETED

1.36 ~~The TECHNICAL REPORT SUPPORTING CYCLE OPERATION is the unit specific document that provides core operating limits for the current operating reload cycle. The cycle-specific core operating limits shall be determined for each reload cycle in accordance with specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.~~

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~~REACTOR TRIP SYSTEM RESPONSE TIME~~ DELETED

1.37 ~~The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until actuation signal output to the Reactor Trip breakers.~~

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
B	At least once per 14 days.
M	At least once per 31 days.
SW	At least once per 42 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
~~S/U~~	~~Prior to each reactor criticality.~~
P	Prior to each release.
N.A.	Not applicable.

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TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. ~~POWER OPERATION~~	~~greater than or equal to 0.99~~	~~greater than 5~~	~~greater than or equal to 350°F~~
2. ~~STARTUP~~	~~greater than or equal to 0.99~~	~~less than or equal to 5~~	~~greater than or equal to 350°F~~
3. ~~HOT STANDBY~~	~~less than 0.99~~	~~0~~	~~greater than or equal to 350°F~~
4. ~~HOT SHUTDOWN~~	~~less than 0.99~~	~~0~~	~~T_{avg} less than 350°F but greater than 200°F~~
5. ~~COLD SHUTDOWN~~	~~less than 0.99~~	~~0~~	~~less than or equal to 200°F~~
6. REFUELING**	less than or equal to 0.94	0	less than or equal to 140°F

DELETED

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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SECTION 2.0
AND
BASES
FOR
SECTION 2.0
SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

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SECTIONS 3.0 AND 4.0
LIMITING CONDITIONS FOR OPERATION
AND
SURVEILLANCE REQUIREMENTS

APR 26 1990

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the ~~OPERATIONAL MODES or other~~ conditions ~~specified therein;~~ specified applicable except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ~~Action~~ requirements is not required.

3.0.3 DELETED ~~When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:~~

~~a. AT least HOT STANDBY within the next 6 hours,~~

~~b. AT least HOT SHUTDOWN within the following 6 hours, and~~

~~c. AT least COLD SHUTDOWN within the subsequent 24 hours.~~

~~Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.~~

~~This specification is not applicable in MODE 5 or 6.~~

3.0.4 ~~Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval.~~ Entry into ~~an OPERATIONAL MODE or~~ a specified applicable condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. ~~This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.~~ Exceptions to these requirements are stated in the individual specifications.

4.0.1 Surveillance Requirements shall be met during the ~~OPERATIONAL MODES or other~~ specified applicable conditions ~~specified~~ for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

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place 4.0.1 & 4.0.2 here

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into ~~an OPERATIONAL MODE or other~~ a specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. ~~This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.~~

~~4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:~~

a. Inservice inspection of ASME Code Class 1, 2, and 3 equivalent components and inservice testing of ASME Code Class 1, 2, and 3 equivalent pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10CFR Part 50, Section 50.55a (g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

SECTIONS 4.0.5, 406;
TABLES 4.0-1, 4.0-2;
FIGURE 4.0-1; AND
PAGES 3/4 0-3 THROUGH 3/4 0-8

DELETED

SECTIONS 3.1 AND 4.1
REACTIVITY CONTROL SYSTEMS

DELETED IN THEIR ENTIRETY

SECTIONS 3.2 AND 4.2
POWER DISTRIBUTION LIMITS

DELETED IN THEIR ENTIRETY

SECTIONS 3.3.1 THROUGH 3.3.3.2;
SECTIONS 4.3.1 THROUGH 4.3.3.2;
TABLES 3.3-1 THROUGH 3.3-4;
TABLES 4.3-1 THROUGH 4.3-3; AND
PAGES 3/4 3-1 THROUGH 3/4 3-26

DELETED

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring system instrumentation shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the seismic monitoring system inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the system to OPERABLE status.

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ DELETED

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 The above seismic monitoring system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 The above required seismic monitoring system actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. If it is determined that the magnitude of the event exceeded the Operating Basis Earthquake, then a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

TABLE 3.3-5

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Servo Accelerometer (SSA-302) Basemat-Cable Vault	0 to 0.59	1
2. Digital Cassette Accelerograph (DCA-300)**	± 5 Volts	1
3. Response Spectrum Analyzer (RSA-50)**	± 5 Volts	1
4. Playback System (SMR-102)**	± 5 Volts	1
5. Seismic Warning Panel (SWP-300)**	~~N/A~~ N.A.	1

**All located in the Control Room

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Servo Accelerometer (SSA-302) Basemat-Cable Vault	M	R	SA
2. Digital Cassette Accelerograph (DCA-300)**	M	R	SA
3. Response Spectrum Analyzer (RSA-50)**	M	R	SA
4. Playback System (SMR-102)**	M	R	SA
5. Seismic Warning Panel (SWP-300)**	M	R	SA

**All located in the Control Room

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ DELETED

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the fraquencies shown in Table 4.3-5.

TABLE 3.3-6

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT/LOCATION	MINIMUM OPERABLE
1. Wind Speed	
a. Baseline Elev. 33′	1
b. Nominal Elev. 200′	1
2. Wind Direction	
a. Baseline Elev. 33′	1
b. Nominal Elev. 196′	1
3. Air Temperature	
a. Baseline Elev. 33′	1
4. Delta T*	
a. Nominal Elev. 120′	1
b. Nominal Elev. 200′	1

* Delta T is the air temperature of the nominal elevation minus the air temperature at the 33′ baseline elevation.

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT/LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Baseline Elev. 33′	D	SA
b. Nominal Elev. 200′	D	SA
2. Wind Direction		
a. Baseline Elev. 33′	D	SA
b. Nominal Elev.196′	D	SA
3. Air Temperature		
a. Baseline Elev. 33′	D	SA
4. Delta T*		
a. Nominal Elev. 120′	D	SA
b. Nominal Elev. 200′	D	SA

* Delta T is the air temperature of the nominal elevation minus the air temperature at the 33′ baseline elevation.

SECTIONS 3.3.3.5, 4.3.3.5;

TABLES 3.3-7, 3.3-8, 4.3-6; AND

PAGES 3/4 3-33 THROUGH 3/4 3-43

DELETED

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION NOV 23 1993

LIMITING CONDITION FOR OPERATION

3.3.3.7 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with applicable Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints shall be determined in accordance with methodology and parameters described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times*.

ACTION:

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

b. With the number of channels less than the minimum channels operable requirement, take the ACTION shown in Table 3.3-9. Exert best efforts to restore the inoperable monitor to OPERABLE status within 30 days, and, if unsuccessful, explain in the next Annual Effluent Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued. 1

c. ~~The provisions of Specification 3.0.3 are not applicable.~~ DELETED

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-7.

---

* At all times means that channel shall be OPERABLE and in service on a continuous, uninterrupted basis, except that outages are permitted for a maximum of 12 hours each time for the purpose of maintenance and performance of required tests, checks, calibrations or sampling.

of monitoring channels

October 4, 1993

TABLE 3.3-9

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM # OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Waste and Recycle Test Tank Discharge Line	1	46
b. ~~Steam Generator Blowdown Line*~~ DELETED	~~1~~	~~47~~
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water Effluent Line	1	48
3. FLOW RATE MEASUREMENT		
a. Waste and Recycle Test Tank Discharge Line	1	49
b. ~~Steam Generator Blowdown Line~~ DELETED	~~**~~	~~N.A.~~
c. Discharge Canal	***	N.A.

~~* Not applicable while in Mode 5 or 6.~~

~~** Flow is determined by the use of valve curves for the purpose of determining flows only.~~

*** Discharge canal flow is determined by the use of pump curves.

TABLE 3.3-9 (Continued)

ACTION STATEMENTS

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, effluent releases may continue provided that best efforts are made to repair the instrument and that prior to initiating a release:

a. At least two independent samples of the tank to be discharged are analyzed in accordance with Specification 4.11.1.1.1, and;

b. The original release rate calculations and discharge valving are independently verified by a second individual.

DELETED

ACTION 47 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 3 x 10^-7 microcuries/ml:~~

~~1. Once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gm DOSE EQUIVALENT I-131.~~

~~2. Once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gm DOSE EQUIVALENT I-131.~~

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that once per 12 hours grab samples of the service water effluent are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 3×10^{-7} microcuries/ml.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that the flow rate is estimated once per 4 hours during actual releases. Pump performance curves generated insitu may be used to estimate flow.

TABLE 4.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Waste and Recycle Test Tank Discharge Line	D(1)	P	R(2)	Q(3)
b. ~~Steam Generator Blowdown Line*~~ DELETED	~~D(1)~~	~~M~~	~~R(2)~~	~~Q(3)~~
2. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water Effluent Line	D(1)	M	R(2)	Q(3)
3. FLOW RATE MEASUREMENT				
a. Waste and Recycle Test Tank Discharge Line	D(1)	N.A.	R	N.A.
b ~~b. Steam Generator Blowdown Line~~ DELETED	~~D(5)~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
c. Discharge Canal	D(4)	N.A.	N.A.	N.A.

* ~~Not applicable while in MODE 5 or 6.~~

TABLE 4.3-7 (Continued)

TABLE NOTATION

(1) CHANNEL CHECK need only be performed daily when discharges are made from this pathway. The CHANNEL CHECK should be done when the discharge is in process.

(2) CHANNEL CALIBRATION shall be performed using a known radioactive liquid or solid source whose strength is determined by a detector which has been calibrated to an NBS source. The radioactive source shall be in a known, reproducible geometry.

(3) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint*.

2. Instrument indicates a downscale failure or circuit failure.

3. Instrument controls not set in operate mode.

(4) Pump status should be checked at least once per 24 hours for the purpose of determining flow rate.

~~(5) Blowdown throttle valve position should be checked daily when discharges are being made via this pathway.~~

---

* Automatic isolation shall also be demonstrated annually for the test tank discharge monitor line ~~and steam generator blowdown line~~.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE with applicable Alarm~~/Trip~~ Setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology and parameters as described in the ODCM.

APPLICABILITY: At all times*.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm~~/Trip~~ Setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the Alarm~~/Trip~~ Setpoint so it is acceptably conservative.

b. With the number of channels less than the minimum channels operable requirement, take the ACTION shown in Table 3.3-10. Exert best efforts to restore the inoperable monitor to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Effluent Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

c. ~~The provisions of Specification 3.0.3 are not applicable.~~ DELETED

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

* At all times means that the channel shall be OPERABLE and in service on a continuous basis, except that outages are permitted for a maximum of 12 hours each time for the purpose of maintenance and performance of required tests, checks, calibrations.

of monitoring channels

TABLE 3.3-10

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. MAIN STACK		
a. Noble Gas Activity Monitor Providing Alarm ~~and Automatic Termination of Waste Gas System Releases~~	1	50
b. ~~Iodino Sampler~~ DELETED	~~1~~	~~51~~
c. Particulate Sampler	1	51
d. Stack Flow Rate Monitor	1	52
e. Sampler Flow Rate Monitor	1	52

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 50 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases via the Waste Gas Holdup System may continue provided that best efforts are made to repair the instrument and that prior to initiating the release:

(a) For the tank to be discharged, at least two independent samples of the tank's contents are analyzed; and,

(b) The release rate calculations and discharge valve lineups are independently verified by a second individual.

Otherwise, suspend releases from the Waste Gas Holdup System.

Releases from all pathways other than the Waste Gas Holdup System may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross radioactivity within 24 hours.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that samples are continuously collected with auxiliary sampling equipment for periods of seven (7) days and analyzed for principal gamma emitters with half lives greater than 8 days within 48 hours after the end of the sampling period. Auxiliary sampling shall be established within 12 hours of declaring the channel INOPERABLE.

ACTION 52 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that the best efforts are made to repair the instrument and that the flow rate is estimated once per 4 hours.

TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. MAIN STACK				
a. Noble Gas Activity Monitor	D(1)	M	R(2)	Q(3)
b. ~~Iodine Sampler~~ DELETED	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
c. Particulate Sampler	W	N.A.	N.A.	N.A.
d. Stack Flow Rate Monitor	D(1)	N.A.	R	N.A.
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) CHANNEL CHECK daily when releases exist via this pathway.

(2) Calibration shall be performed using a known source whose strength is determined by a detector which has been calibrated to an NBS source. These sources shall be in a known, reproducible geometry.

(3) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the Alarm~~/Trip~~ Setpoint~~*~~.

b. Instrument indicates a downscale failure or circuit failure.

c. Instrument controls not set in operate mode.

~~* Automatic isolation of the waste gas releases by the noble gas activity monitor should also be demonstrated.~~

SECTIONS 3.3.3.9, 3.3.4, 4.3.3.9, 4.3.4;

TABLE 3.3-11; AND

PAGES 3/4 3-54 THROUGH 3/4 3-57

DELETED

SECTIONS 3.4 AND 4.4

REACTOR COOLANT SYSTEM

DELETED IN THEIR ENTIRETY

SECTIONS 3.5 AND 4.5
EMERGENCY CORE COOLING SYSTEMS

DELETED IN THEIR ENTIRETY

SECTIONS 3.6 AND 4.6

CONTAINMENT SYSTEMS

DELETED IN THEIR ENTIRETY

SECTIONS 3.7.1 THROUGH 3.7.4;

SECTIONS 4.7.1 THROUGH 4.7.4;

TABLES 3.7-1 THROUGH 3.7-2;

TABLES 4.7-1 THROUGH 4.7-2; AND

PAGES 3/4 7-1 THROUGH 3/4 7-13

DELETED

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.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and either:

1. Decontaminate and repair the sealed source, or

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

b. DELETED ~~The provisions of Specification 3.0.3 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.

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 6 months for all sealed sources containing radioactive materials:

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

2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested ~~within 31 days prior to being subjected to core flux or installed in the core and~~ following repair or maintenance to the source.

4.7.5.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removal contamination.

SECTIONS 3.7.6 THROUGH 3.7.12;
SECTIONS 4.7.6 THROUGH 4.7.12;
TABLES 3.7-3 THROUGH 3.7-6; AND
PAGES 3/4 7-16 THROUGH 3/4 7-36

DELETED

SECTIONS 3.8 AND 4.8
ELECTRICAL POWER SYSTEMS

DELETED IN THEIR ENTIRETY

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled unisolated portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure a K_{eff} of 0.94 or less.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing a minimum of 14000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.94.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The above reactivity condition shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any control rod in excess of 3 feet from its fully inserted position within the reactor vessel except when control rod drag testing is being performed, at which time the reactivity condition shall be determined prior to the first control rod withdrawn in excess of 3 feet and every 4 hours thereafter until completion of the testing.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

---

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

1

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2.a As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room when CORE ALTERATIONS or positive reactivity changes are taking place. When CORE ALTERATIONS or positive reactivity changes are not taking place, at least one Source Range Neutron Flux Monitor shall be OPERABLE and operating with a visual indication in the control room and audible indication in the containment.

3.9.2.b As a minimum, two Source Range High Neutron Level Alarms (Containment Evacuation) shall be OPERABLE and operating with a minimum logic to audibly alarm in both the control room and containment of one (1) of two (2).

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

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3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3.a The reactor shall be subcritical for at least 100 hours.

3.9.3.b Irradiated fuel shall remain in the reactor vessel for at least the time required in Figure 3.9-1 prior to movement to the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.*

ACTION:

a. With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

b. With the combination of reactor hold time and cooling water temperature in the unacceptable region of Figure 3.9-1, suspend all operations involving movement of irradiated fuel from the reactor vessel to the spent fuel pool until the combination of reactor hold time and cooling water temperature return to the acceptable region of Figure 3.9-1.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

4.9.3.2 Verify that the combination of reactor hold time and cooling water temperature are in the acceptable region of Figure 3.9-1 prior to and at least once per 12 hours during transfer of irradiated fuel from the reactor vessel to the spent fuel pool.

*Specification 3.9.3.b does not apply if less than 53 fuel assemblies are transferred to the spent fuel pool.

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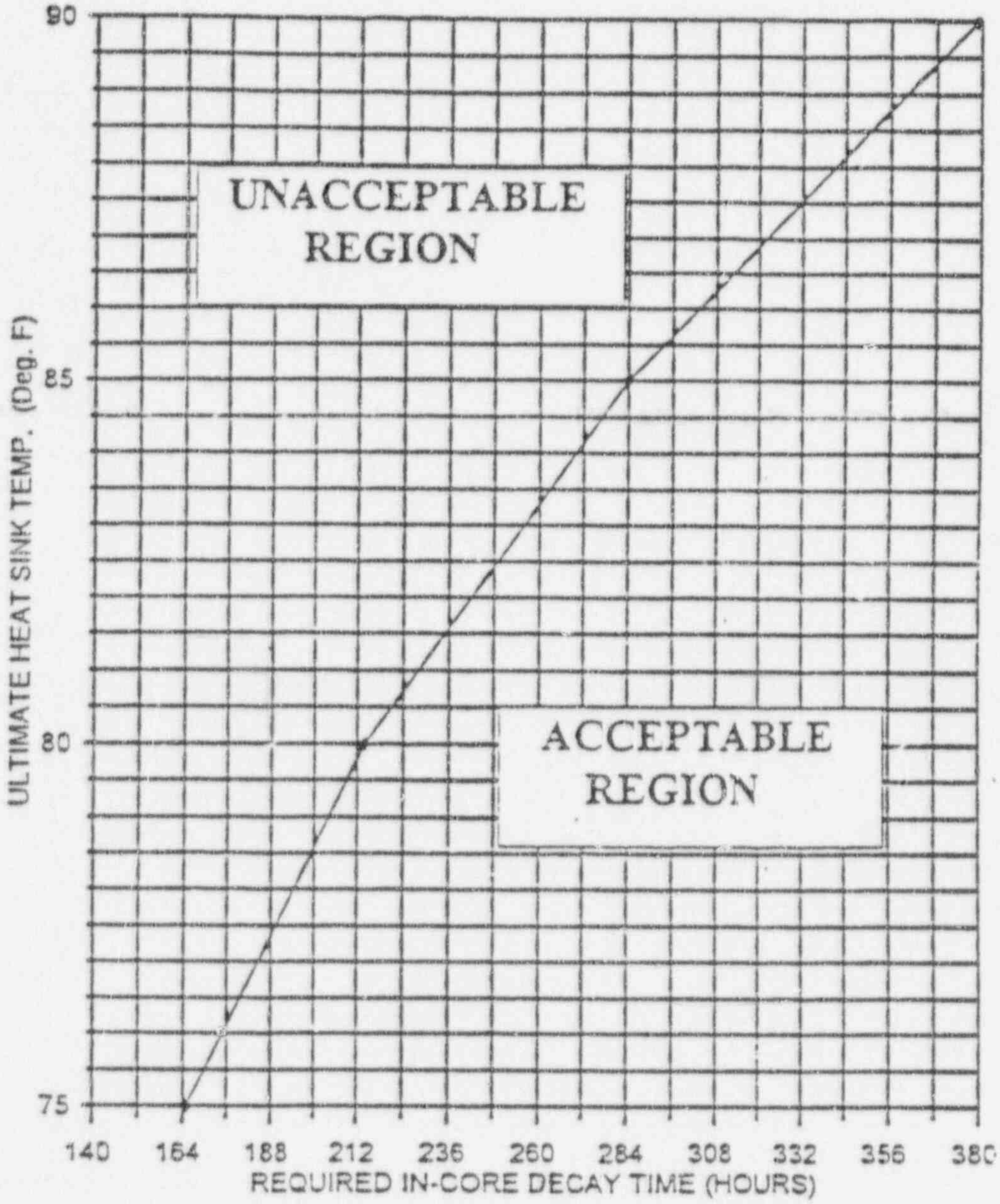

FIGURE 3.9-1 REQUIRED REACTOR HOLD TIME FOR A FULL CORE OFFLOAD

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch installed and secured in place,

b. A minimum of one door in the airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be, either:

   1) Closed by an isolation valve, blind flange, manual valve, or special device, or

   2) Be capable of being closed by an OPERABLE containment purge supply, purge exhaust, or purge exhaust bypass isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building, other than placing an irradiated fuel assembly into a safe storage location.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE containment purge supply, purge exhaust, or purge exhaust bypass isolation valve or capable of being closed by an OPERABLE airlock door within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations referred to in Specification 3.9.4c are in their closed/isolated condition,

b. Testing the containment purge supply, purge exhaust, and purge exhaust bypass isolation valves in accordance with Specification 4.9.9, and

c. Verifying that at least one of the doors in the airlock is capable of being closed.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling cavity manipulator crane area.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building other than placing an irradiated fuel assembly into a safe storage location.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling cavity manipulator crane area shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rod drive shafts* or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1) A minimum capacity of 1650 pounds, and

2) An overload cut off limit less than or equal to 175 pounds above the indicated weight for wet conditions for a stainless steel clad fuel assembly with a full length RCCA inserted.

b. The auxiliary hoist used for latching and unlatching control rod drive shafts having a minimum capacity of 1500 pounds.

APPLICABILITY: During movement of control rod drive shafts or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, place any fuel assembly or control rod in transit into a safe storage location and suspend use of the inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rod drive shafts or fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2063 pounds and demonstrating an automatic load cut off when the crane load exceeds 175 pounds above the indicated weight for wet conditions for a stainless steel clad fuel assembly with a full length RCCA inserted.

4.9.6.2 Each auxiliary hoist used for movement of control rod drive shafts within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations. If loads greater than the weight of a stainless steel clad fuel assembly with an RCCA inserted are to be lifted within the reactor vessel with the auxiliary hoist then a load test will be performed for 125% of the weight of the load to be lifted.

* The RCCA's may be attached to the control rod drive shafts during some refueling evolutions.

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1650 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Administrative controls that prevent the travel of loads in excess of 1650 pounds over fuel assemblies shall be in place prior to lifting a load in excess of 1650 pounds.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one RHR LOOP shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level is greater than or equal to 23 feet above the top of the reactor vessel flange.

ACTION:

With no RHR LOOP OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR LOOP to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least once per 12 hours verify at least one RHR LOOP is in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm.

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* The RHR LOOP may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent RHR LOOPS shall be OPERABLE, and at least one RHR LOOP shall be in operation.

APPLICABILITY: MODE 6 when the water level is less than 23 feet above the top of the reactor vessel flange.

ACTION:

a. With less than the required RHR LOOPS OPERABLE, immediately initiate corrective action to return the required RHR LOOPS to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR LOOP in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR LOOP to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2.1 At least once per 12 hours verify at least one RHR LOOP is in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm.

4.9.8.2.2 The RHR LOOP not in operation shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability.

3/4.9.9 CONTAINMENT PURGE SUPPLY, PURGE EXHAUST, AND PURGE EXHAUST BYPASS ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge supply, purge exhaust, and purge exhaust bypass isolation valves shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building, other than placing an irradiated fuel assembly into a safe storage location.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge supply, purge exhaust, and purge exhaust bypass isolation valves shall be verified OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS and prior to movement of irradiated fuel within containment by verifying that containment purge, bypass or exhaust isolation valves are accessible for manual operation or verifying that the associated penetrations are blind flanged.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, place any fuel assembly or control rod in transit into a safe storage location and suspend all further operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 20 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movements of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 FUEL STORAGE BUILDING AIR CLEANUP SYSTEM

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LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Storage Building Air Cleanup System shall be OPERABLE and in operation.

APPLICABILITY: During operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool.*

ACTION:

a. With the Fuel Storage Building Air Cleanup System inoperable, or not operating, suspend all operations involving movements of fuel within the storage pool or crane operation with loads over the storage pool.*

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The Fuel Storage Building Air Cleanup System shall be demonstrated OPERABLE and in operation:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10% at test conditions of 86°F, 95% relative humidity atmospheric pressure, and 40 feet/min face velocity in accordance with ASTM D3803; and

3) Verifying a system flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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*The above specification does not apply during movement of fuel storage rack modules into or out of the SFP during the Cycle 19 rerack, provided an approved safe loading path is established with no loads over stored fuel assemblies.

SURVEILLANCE REQUIREMENTS (Continued)

b. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10% at test conditions of 86°F, 95% relative humidty, atmospheric pressure, and 40 feet/min. face velocity in accordance with ASMT D3803.

c. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%, and

   2) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than 0 inch Water Gauge differential relative to the outside atmosphere during system operation.

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

e. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

REFUELING OPERATIONS

3/4.9.13 MOVEMENT OF FUEL IN SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.13 Prior to movement of a fuel assembly in the spent fuel pool, the boron concentration of the pool shall be maintained uniform and sufficient to maintain a boron concentration of greater than or equal to 800 ppm.

APPLICABILITY: Whenever a fuel assembly is moved in the spent fuel pool.

ACTION:

With the boron concentration less than 800 ppm, suspend the movement of all fuel in the spent fuel pool.

SURVEILLANCE REQUIREMENT

4.9.13 Verify that the boron concentration is greater than or equal to 800 ppm within 24 hours prior to any movement of a fuel assembly in the spent fuel pool and every 72 hours thereafter.

3/4.9.14 SPENT FUEL POOL--REACTIVITY CONDITION

LIMITING CONDITION FOR OPERATION

3.9.14 The Reactivity Condition of the spent fuel pool shall be such that K_{eff} is less-than-or-equal-to 0.95 at all times.

APPLICABILITY: Whenever fuel is in the spent fuel pool.

ACTION:

Immediately initiate actions to correct the loading error if the placement of fuel assemblies does not meet the requirements of Figure 3.9-2, Figure 3.9-3, and Figure 3.9-4.

SURVEILLANCE REQUIREMENT

4.9.14 Ensure that all fuel assemblies to be placed in the spent fuel pool are within the enrichment and burn-up limits of Figure 3.9-2 (Region 2) and Figure 3.9-3 (Region 3) by checking the assembly's design and burn-up documentation. Region locations are shown in Figure 3.9-4.

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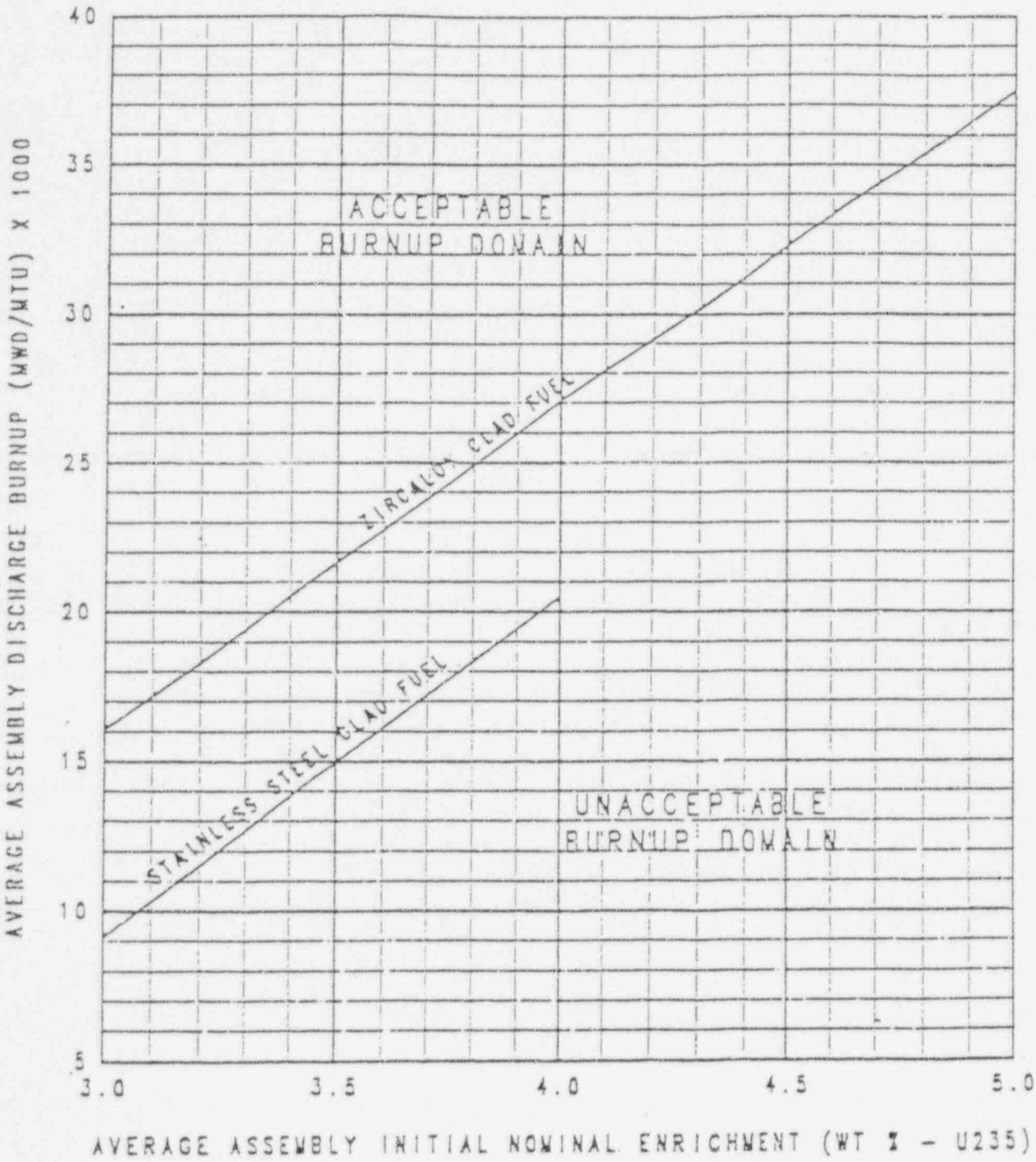

FIGURE 3.9-2 SPENT FUEL POOL RACK MINIMUM BURNUP REQUIREMENTS FOR REGION 2

JAN 22 1996

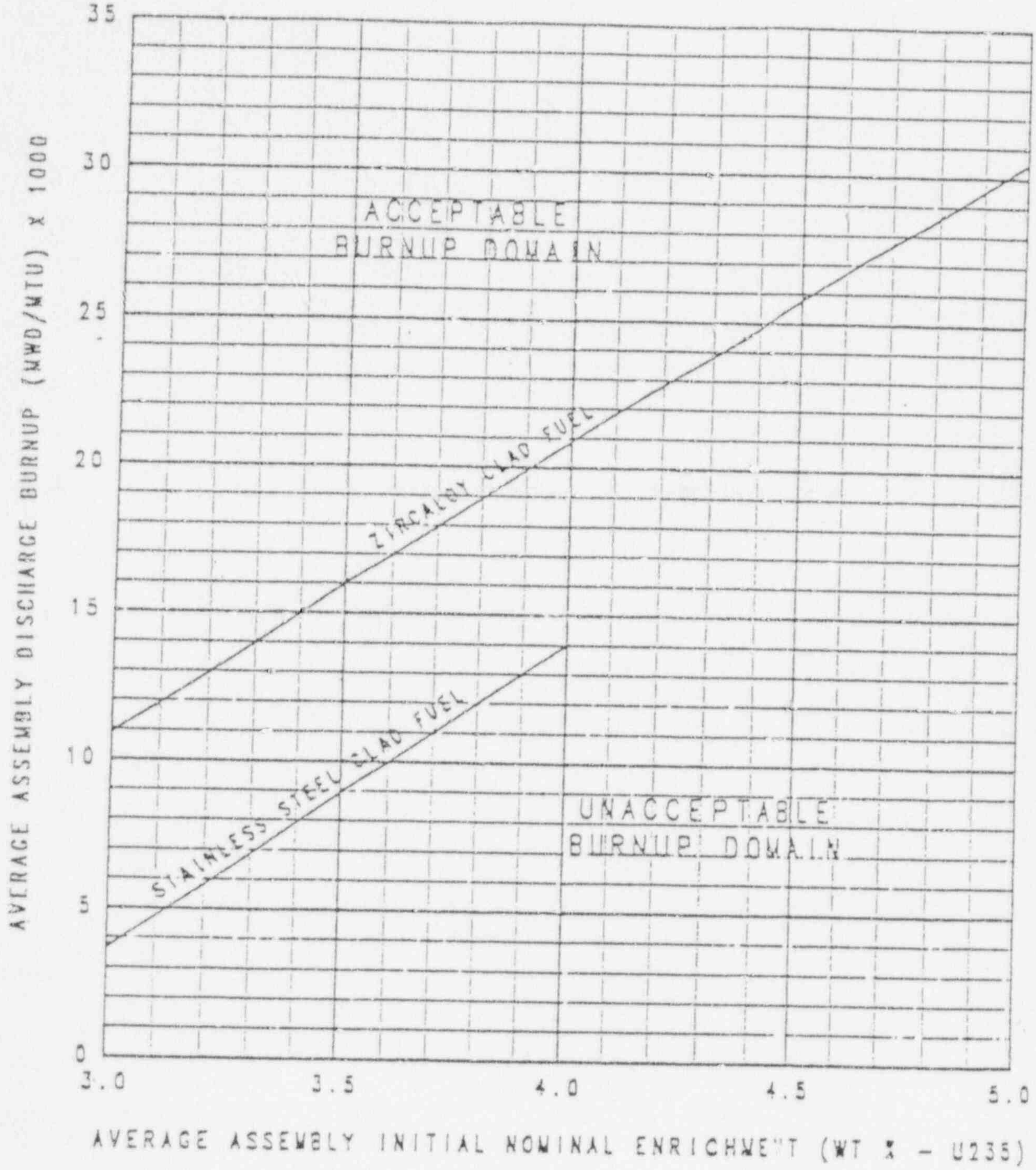

FIGURE 3.9-3 SPENT FUEL POOL RACK MINIMUM BURNUP REQUIREMENTS FOR REGION 3

JAN 22 1996

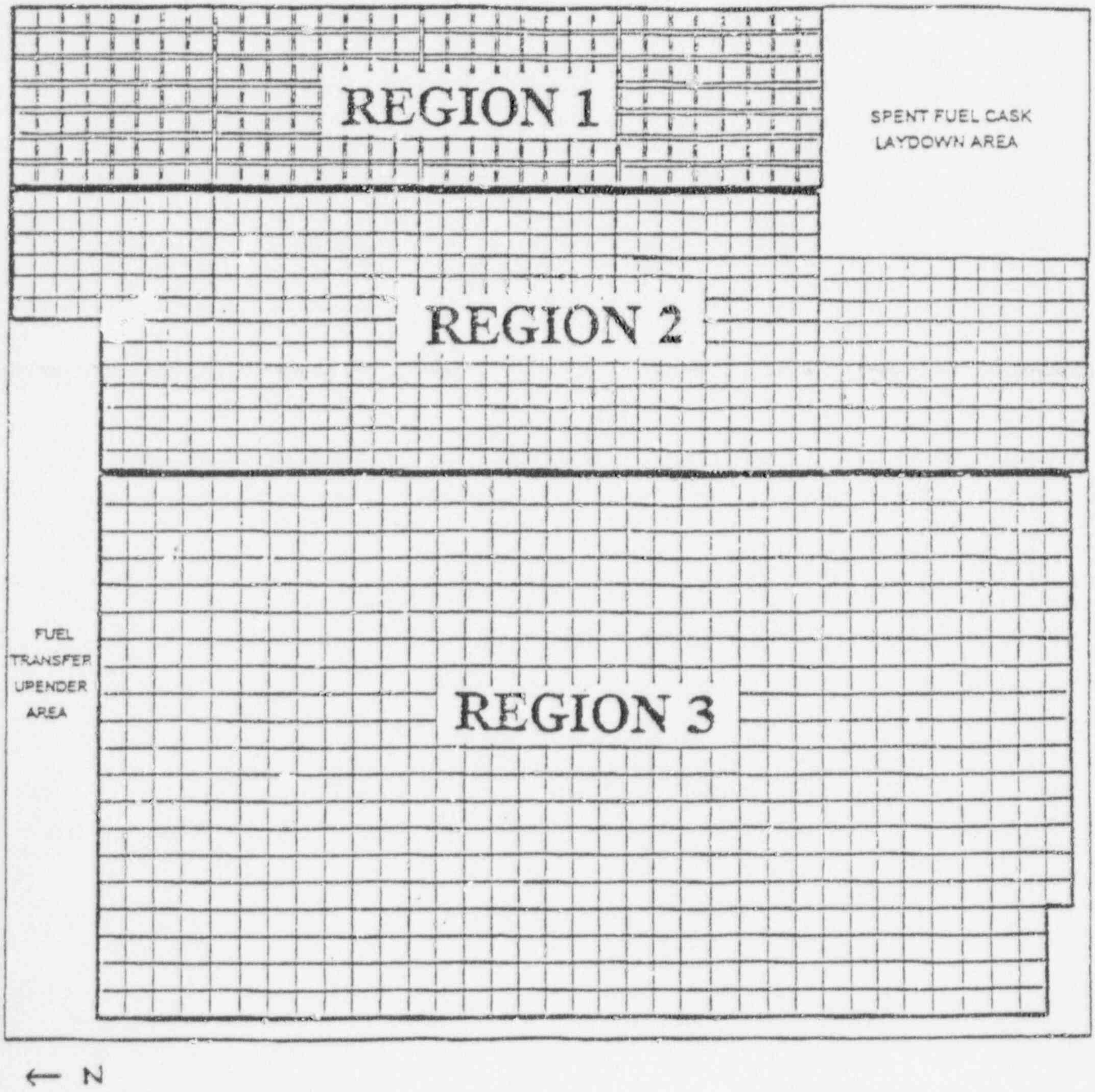

FIGURE 3.9-4 SPENT FUEL POOL RACK REGION LOCATIONS

3/4.9.15 SPENT FUEL POOL COOLING

LIMITING CONDITION FOR OPERATION

3.9.15 The spent fuel pool cooling system shall be OPERABLE with:

a. Both spent fuel pool cooling pumps OPERABLE and at least one spent fuel pool cooling pump and the plate heat exchanger in operation, and

b. Spent fuel pool temperature less than 150 degrees F.

APPLICABILITY: MODE 6, during transfer and storage of irradiated fuel from the reactor vessel to the spent fuel pool for a full core offload.*

ACTION: With less than the required equipment OPERABLE and in operation or with spent fuel pool temperature greater than 150 degrees F, suspend all operations involving the addition of irradiated fuel to the spent fuel pool and initiate corrective action to restore the spent fuel pool cooling system to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.9.15.1 Prior to movement of irradiated fuel to the spent fuel pool, verify that both spent fuel pool cooling pumps are lined up to provide flow to the plate heat exchanger.

4.9.15.2 At least once per 12 hours, verify that at least one spent fuel pool cooling pump is running.

4.9.15.3 At least once per 12 hours, verify that the spent fuel pool temperature is less than 150 degrees F.

*Specification 3.9.15 does not apply if less than 53 fuel assemblies are transferred to the spent fuel pool.

SECTIONS 3.10 AND 4.10

SPECIAL TEST EXCEPTIONS

DELETED IN THEIR ENTIRETY

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-1) shall not exceed the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall not exceed 2×10^{-4} microCi/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, restore the concentration to within the above limits within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed in accordance with the sampling and analysis program specified in Section I of the REMODCM.

4.11.1.1.2 The results of the radioactive analysis shall be used in accordance with the methods of Section II of the REMODCM to assure that the concentration of the point of release are maintained within the limits of Specification 3.11.1.1.

RADIOACTIVE EFFLUENTS

DOSE, LIQUIDS

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to any MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the site (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose or dose commitment to any MEMBER OF THE PUBLIC from such release during the calendar year is within 3 mrem to the total body and 10 mrem to any organ.

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ DELETED

SURVEILLANCE REQUIREMENTS

4.11.1.2.1 Cumulative dose contributions from liquid effluents shall be determined in accordance with Section II of the REMODCM.

4.11.1.2.2 Relative accuracy or conservatisms of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in the REMODCM.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate, at any time, offsite (see Figure 5.1-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. The dose rate limit due to inhalation for ~~Iodine 131, Iodine 133,~~ tritium and for all radioactive materials in particulate form with half-lives greater than 8 days shall be less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, decrease the release rate within 15 minutes to comply with the limit(s) given in Specification 3.11.2.1.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The release rate, at any time, of noble gases in gaseous effluents shall be controlled by the offsite dose rate as established above in Specification 3.11.2.1. The corresponding release rate shall be determined in accordance with the methodology of Section II of the REMODCM.

4.11.2.1.2 The noble gas effluent monitors of Specification 3.3.3.8 shall be used to control release rates to limit offsite doses within the values established in Specification 3.11.2.1.

4.11.2.1.3 The release rate of radioactive materials in gaseous effluents shall be determined by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Section I of the REMODCM (Table D-1). The corresponding dose rate shall be determined using the methodology and parameters given in the ODCM (Section II of the REMODCM).

RADIOACTIVE EFFLUENTS

DOSE, NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose offsite (see Figure 5.1-1) due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year to less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose during the calendar year is within 10 mrad for gamma radiation and 20 mrad for beta radiation.

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ (DELETED)

SURVEILLANCE REQUIREMENTS

4.11.2.2.1 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with Section II of the REMODCM once every 31 days.

4.11.2.2.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in Section I of the REMODCM.

DOSE, ~~RADIOIODINES,~~ RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to any MEMBER OF THE PUBLIC from ~~Iodine-131, Iodine-133,~~ tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents released offsite (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter to less than or equal to 7.5 mrem to any organ;

b. During any calendar year to less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radionuclides, radioactive materials in particulate form, or radionuclides other than noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to reduce the releases during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose or dose commitment to any MEMBER OF THE PUBLIC from such releases during the calendar year is within 15 mrem to any organ.

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ DELETED

SURVEILLANCE REQUIREMENTS

4.11.2.3.1 Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with Section II of the REMODCM once every 31 days.

4.11.2.3.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in the REMODCM.

RADIOACTIVE EFFLUENTS

3/4.11.3 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.3 The dose or dose commitment from the site to a MEMBER OF THE PUBLIC is limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) over a period of 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3, prepare and submit a Special Report to Commission pursuant to Specification 6.9.2 and limit the subsequent releases such that the dose or dose commitment from the site to any MEMBER OF THE PUBLIC is limited to less than or equal to 25 mrem to the total body or any organ (except thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures from the site to any MEMBER OF THE PUBLIC (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. If the estimated doses exceed the above limits, the special report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. ~~The provisions of Specification 3.0.3 are not applicable~~. DELETED

SURVEILLANCE REQUIREMENTS

4.11.3 Cumulative dose contributions from liquid and gaseous effluents and direct radiation shall be determined in Specifications 4.11.1.2.1, 4.11.2.2.1 and 4.11.2.3.1 and in accordance with Section II of the REMODCM once per 31 days.

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BASES FOR
SECTIONS 3.0 AND 4.0
LIMITING CONDITIONS FOR OPERATION
AND
SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

BASES

Specifications 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10CFR50.36(c)(2):

> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

applicable

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., ~~in which OPERATIONAL MODES or other~~ specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, ~~a shutdown~~ is required ~~to place~~ the facility in a ~~MODE or~~ condition in which the specification no longer applies. It is not intended that the ~~shutdown~~ ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

specified applicable

be placed

X

that

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. ~~When a shutdown is required to comply with ACTION requirements, the plant may have~~

BASES (Con't)

~~entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.~~

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 DELETED ~~establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the~~ condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than

3/4.0 APPLICABILITY

BASES (Con't)

allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may by added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

~~The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.~~

from a specified applicable condition

Specification 3.0.4 establishes limitations on ~~MODE~~ changes when a Limiting Condition for Operation is not met. ~~It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted~~. The purpose of this specification is to ensure that the facility ~~operation~~ is not changed from a specified applicable condition ~~initiated or that higher MODES of operation are not entered~~ when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant, ~~before or after a MODE change.~~ Therefore, in this case, entry into a ~~an OPERATIONAL MODE or other~~ specified applicable condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status. ~~before plant startup.~~

~~When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.~~

(4)

Specifications 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

<u>Specification 4.0.1</u> establishes the requirement that surveillances must be performed during the ~~OPERATIONAL MODES or other~~ specified applicable conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a ~~MODE or other~~ specified applicable condition for which the associated Limiting Conditions for Operation are applicable. ~~Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.~~

<u>Specification 4.0.2</u> establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. ~~It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outage.~~ The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

<u>Specification 4.0.3</u> establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable

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although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation.

If the allowable outage time limits of the ACTION requirements are less than 24 hours~~, or a shutdown is required to comply with ACTION requirements, e.g.~~, ~~Specification 3.0.3~~, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before ~~a shutdown is required to comply with ACTION requirements or before other~~ remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into ~~an OPERATIONAL MODE or other~~ a specified applicable condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a ~~MODE or~~ specified applicable condition for which these systems and components ensure safe operation of the facility. ~~This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.~~

BASES 4.0.5, 4.0.6;

REFERENCES 1, 2, 3; AND

PAGES B 3/4 0-6 THROUGH B 3/4 0-7

DELETED

BASES 3/4.1

REACTIVITY CONTROL SYSTEMS

DELETED IN THEIR ENTIRETY

BASES 3/4.2

POWER DISTRIBUTION LIMITS

DELETED IN THEIR ENTIRETY

BASES 3/4.3.1 THROUGH 3/4.3.3.2; AND

PAGES B 3/4 3-1 AND B 3/4 3-1a

DELETED

INSTRUMENTATION

BASES

3/4.3.3 MONITORING INSTRUMENTATION

~~3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS~~

~~The OPERABILITY of the Containment Atmosphere Gaseous Radioactivity Monitoring System ensures that Gaseous Radioactivity Monitoring System will monitor inside containment as a means to detect RCS leakage.~~

~~3/4.3.3.2 MOVABLE INCORE DETECTORS~~

~~The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.~~

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

~~3/4.3.3.5 ACCIDENT MONITORING INSTRUMENTATION~~

~~The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.~~

~~3/4.3.3.6 DELETED~~

BASES 3/4.3.3.5 AND 3/4.3.3.6

DELETED

3/4.3.3.7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the REMODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

~~3/4.3.3.9 BORON DILUTION ALARM~~

~~The shutdown monitors provide indication of positive reactivity insertion during operation in Modes 3, 4, 5, and 6. The indication is credited in the Boron Dilution Design basis analysis.~~

BASES 3/4.3.3.9 AND 3/4.3.4; AND

PAGE B 3/4 3-4

DELETED

BASES 3/4.4

REACTOR COOLANT SYSTEM

DELETED IN THEIR ENTIRETY

BASES 3/4.5

EMERGENCY CORE COOLING SYSTEMS

DELETED IN THEIR ENTIRETY

BASES 3/4.6

CONTAINMENT SYSTEMS

DELETED IN THEIR ENTIRETY

BASES 3/4.7.1 THROUGH 3/4.7.4; AND

PAGES B 3/4 7-1 THROUGH B 3/4 7-3a

DELETED

PLANT SYSTEMS

BASES

~~3/4.7.4 SNUBBERS (Continued)~~

The service life is evaluated via manufacturer input and information with consideration of snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, high temperature area, location of potential fluid transient loading, etc.) The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of age and operating conditions. (Due to implementation of the snubber service life monitoring program after several years of plant operation, the historical records to date may be incomplete.) Snubber service life records will provide a statistical basis for future consideration of snubber service life.

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

~~3/4.7.6 DELETED~~

BASES 3/4.7.6 THROUGH 3/4.7.12; AND

PAGES B 3/4 7-5 THROUGH B 3/4 7-7

DELETED

BASES 3/4.8

ELECTRICAL POWER SYSTEMS

DELETED IN THEIR ENTIRETY

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. A value of 0.94 or less for K_{eff} is required for this accident. For a core configuration of all rods in an additional 0.05 K_{eff} penalty is required to account for a heavy load crushing the core into a more reactive configuration.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that monitoring capability is available to detect changes in the reactivity condition of the core. Redundant monitoring capability is required to detect changes in the reactivity condition of the core during fuel movement.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum requirement for reactor hold time prior to movement of irradiated fuel from the reactor vessel to the spent fuel pool ensures that sufficient decay time has elapsed for adequate spent fuel pool cooling should a failure occur to the cooling system. The reactor hold time is a function of cooling water temperature. The required decay time assumes a conservative maximum transfer rate of six assemblies per hour. These decay times are consistent with the assumptions used in the safety analysis.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling cavity manipulator crane area personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of control rod drive shafts and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive shaft or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting forces in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analysis.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one RHR LOOP be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR LOOPS OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR LOOP will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR LOOP, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE SUPPLY, PURGE EXHAUST, AND PURGE EXHAUST BYPASS ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations can be isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL STORAGE BUILDING AIR CLEANUP SYSTEM

The limitations on the Fuel Storage Building Air Cleanup System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analysis. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.9.13 MOVEMENT OF FUEL IN SPENT FUEL POOL

The limitations of this specification ensure that, in the event of any fuel handling accident in the spent fuel pool, K_{eff} will remain ≤ 0.95.

3/4.9.14 SPENT FUEL POOL - REACTIVITY CONDITION

The limitations described by Figures 3.9-2, 3.9-3, and Figure 3.9-4 ensure that the reactivity of fuel assemblies introduced into the spent fuel racks, with no credit taken for soluble boron in the pool, are conservatively within the assumptions of the safety analysis.

3/4.9.15 SPENT FUEL POOL COOLING

The limitations on the Spent Fuel Pool Cooling System ensure there is sufficient capacity to remove decay heat produced by the stored spent fuel elements with a full core offload and maintain the bulk pool temperature below 150°F with a maximum heat load of 22.4×10^6 BTU/hr. The maximum design heat load is based on an emergency full core offload scenario conservatively evaluated after the final operating cycle.

Requiring both spent fuel pool cooling pumps to be OPERABLE provides backup capability in the event the operating pump fails. In the event of a complete loss of forced cooling during a full core offload, the time to boil is greater than 7 hours in the most severe discharge scenario. This allows sufficient time to provide an alternate power source to the SFP pumps for an electrical failure or alternate SFP inventory makeup capability for a mechanical failure. In the event of a loss of offsite power and the A train diesel is out-of-service, there is sufficient time to repower the SFP cooling system from an alternate diesel generator. Therefore, operability of the spent fuel pool cooling system does not require the A train diesel generator to be available.

Should failure to restore operation of the cooling system occur before boiling takes place, cooling of the spent fuel pool can be accomplished by allowing the SFP to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

BASES 3/4.10

SPECIAL TEST EXCEPTIONS

DELETED IN THEIR ENTIRETY

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

2
1

3/4.11.1.2 DOSE, LIQUIDS

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". The dose calculation methodology and parameters in the REMODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the REMODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose rate at anytime from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for all areas offsite. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual offsite to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of that individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid or other organ dose rate above background to a child to less than or equal to 1500 mrem/year from inhalation.

3/4.11.2.2 DOSE, NOBLE GASES

This specification is provided to implement the requirements of Sections II.B., III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculational of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977.

The ODCM equations provided for determining the air doses at the site boundary are based upon utilizing successively more realistic dose calculational methodologies. More realistic dose calculational methods are used whenever simplified calculations indicate a dose approaching a substantial portion of the regulatory limits. The methods used, in order are, previously determined air dose per released activity ratio, historical meteorological data and actual radionuclide mix released, or real time meteorology and actual radionuclides released.

3/4.11.2.3 DOSE, ~~RADIOIODINES,~~ & RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides for Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials will to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision I, July 1977. The release rate specifications for ~~radioiodines,~~ radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.3 TOTAL DOSE

This specification is provided to meet the reporting requirements of 40 CFR Part 190. For the purposes of the Special Report, it may be assumed that the dose commitment to any MEMBER OF THE PUBLIC from other fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

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SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 shall be 1740 feet.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

5.2 ~~CONTAINMENT~~ DELETED

~~CONFIGURATION~~

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

a. Nominal inside diameter of cylinder = 134 feet - 11.25 inches.

b. Nominal inside height of cylinder = 119.5 feet (Not including dome).

c. Nominal inside height of containment = 188 feet - 11.5 inches (From containment floor to inside top of dome).

d. Minimum thickness of concrete walls = 4.5 feet.

e. Minimum thickness of concrete dome = 2.5 feet.

f. Minimum thickness of concrete bottom mat = 9 feet.

g. Nominal thickness of steel liner = 1/4 to 1/2 inch (bottom = 1/4 inch, spherical dome = 1/2 inch and side wall = 3/8 inch).

h. Net free volume = 2.232×10^6 cubic feet (nominal).

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 40 psig and a temperature of 260°F.~~

5.3 DELETED

5.4 DELETED

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

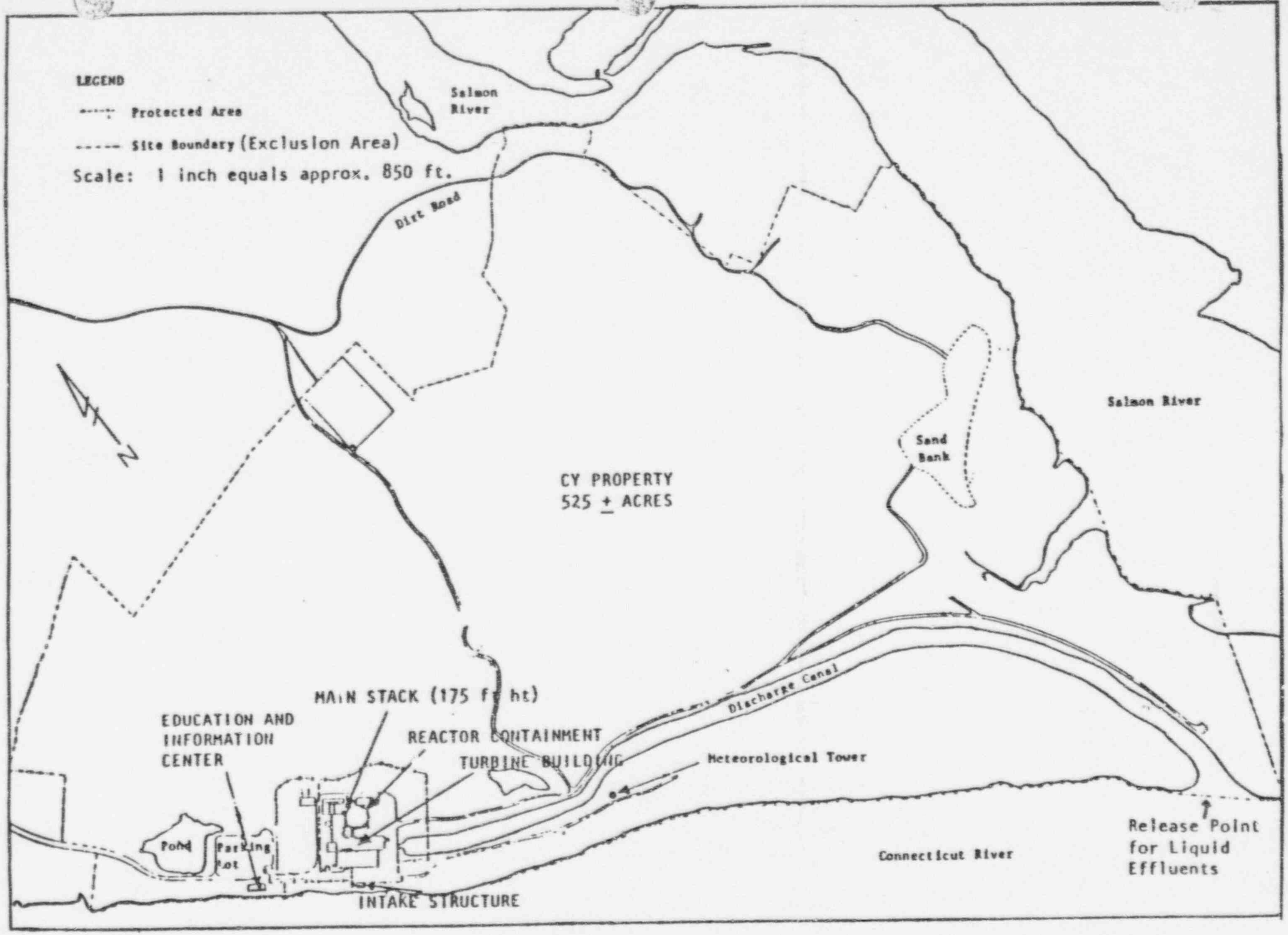

FIGURE 5.1-1

EXCLUSION AREA BOUNDARY AND SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

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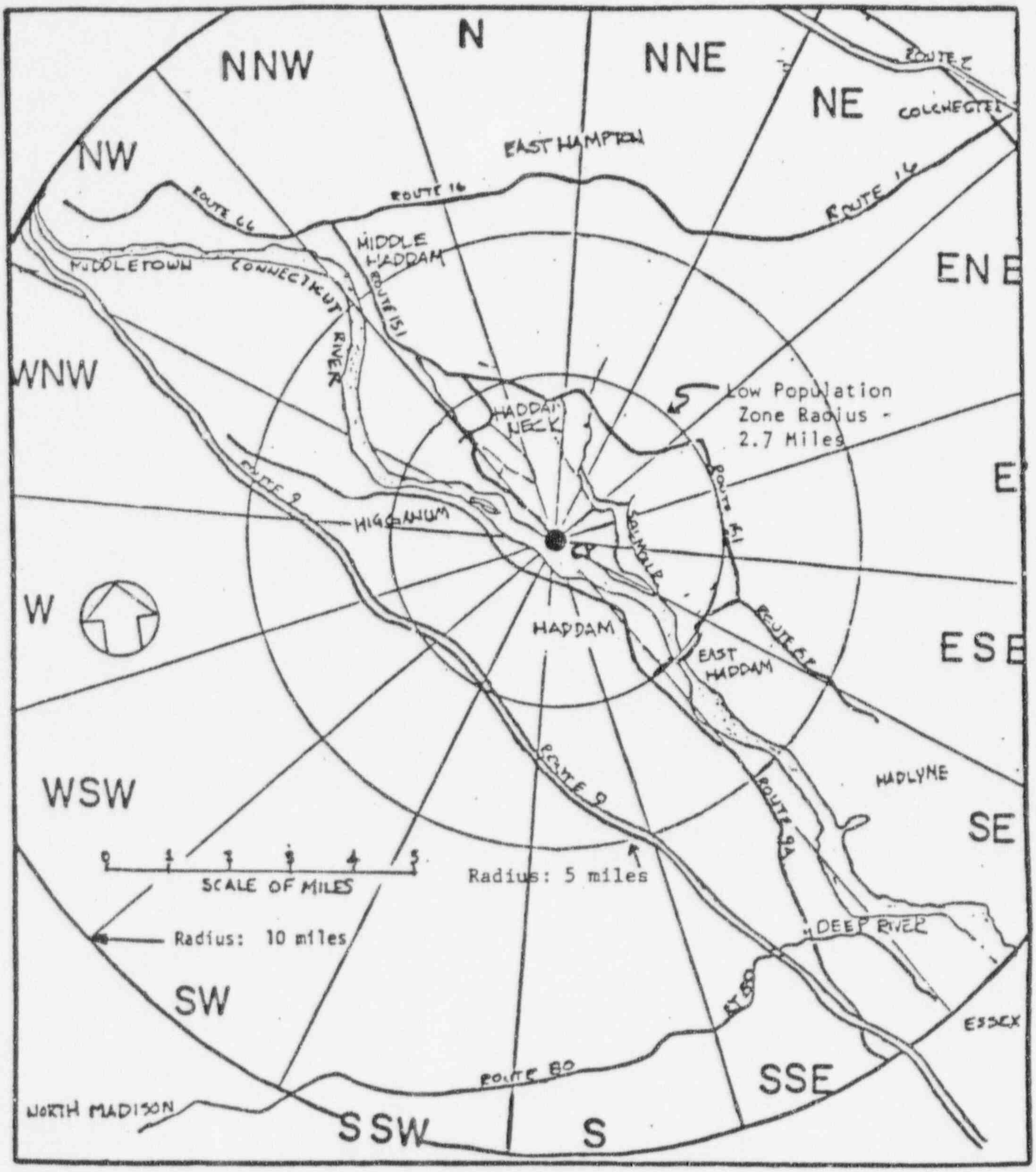

FIGURE 5.1-2

LOW POPULATION ZONE

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4 or Type 304 stainless steel. Each stainless steel clad fuel assembly shall have a nominal active fuel length of 120.5 inches and contain a typical weight of 411.5 kilograms uranium. Each Zircaloy-4 clad B&W fuel assembly shall have a nominal active fuel length of 118.6 inches and contain a typical weight of 363.8 kilograms uranium. Each Zircaloy-4 clad Westinghouse fuel assembly shall have a nominal active fuel length of 120.3 inches and contain a typical weight of 386.2 kilograms uranium. The core loading shall have a maximum enrichment of 5.00 weight percent U-235 nominal.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 120.25 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 8780 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 SPENT FUEL STORAGE

~~5.6.1~~ CRITICALITY

~~SPENT FUEL~~

5.6.1.~~1~~ 2 The spent fuel storage racks are made up of 3 regions which are designed and shall be maintained with:

a. For Region 1, a nominal 10.98 inch (East/West) and a nominal 10.142 inch (North/South) center-to-center distance including neutron absorber surrounding each assembly to ensure a K_{eff} less than or equal to .95 when flooded with unborated water. Fuel with a maximum nominal enrichment of 5 weight percent U-235, without regard for fuel burnup, may be stored in this region. The location of Region 1 within the Spent Fuel Pool is shown in Figure 3.9-4.

b. For Region 2, a nominal 9.00 inch center-to-center distance including neutron absorber surrounding each assembly to ensure a K_{eff} less than or equal to .95 when flooded with unborated water. Storage of fuel in Region 2 must be in accordance with the requirements of Specification 3.9.14. The location of Region 2 within the Spent Fuel Pool is shown in Figure 3.9-4.

c. For Region 3, a nominal 10.75 inch center-to-center distance to ensure a K_{eff} less than or equal to .95 when flooded with unborated water. No credit is taken for the neutron absorber contained within the racks of this region. Storage of fuel in Region 3 must be in accordance with the requirements of Specification 3.9.14. The location of Region 3 within the Spent Fuel Pool is shown in Figure 3.9-4.

~~NEW FUEL~~

~~5.6.1.2 The new fuel storage racks are designed and shall be maintained with:~~

a. A nominal 18.625-inch center-to-center distance with a full-length polyvinyl chloride liner to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 assuming optimum moderating conditions, and

b. The maximum fuel assembly enrichment in the new fuel storage racks will be 5.0 weight percent U-235 (nominal). New fuel assemblies with an average enrichment greater than 4.60 weight percent U-235 (nominal) must contain Integral Fuel Burnable Absorbers (IFBA) rods in accordance with the requirements of Figure 5.6-1. New ~~fuel assemblies will be stored in accordance with the requirements of Figure 5.6-2.~~

Move to page 5-4

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 27.0 feet MSL.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 224 storage locations in Region 1, 560 storage locations in Region 2, and 696 storage locations in Region 3 for a total of 1480 storage locations.

~~5.7 REACTOR VESSEL DESIGN TRANSIENTS~~

~~5.7.1 The reactor vessel design transients are as identified in Table 5.7-1. The transient description followed by the number of design cycles is provided.~~

SECTION 5.7;
TABLE 5.7-1;
FIGURES 5.6-1 AND 5.6-2; AND
PAGES 5-5 THROUGH 5-8

DELETED

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

designated CYAPCO corporate officer

6.1.1 The ~~Executive Vice President and Chief Nuclear Officer~~ shall be responsible for overall facility operation and shall delegate, in writing, the succession to this responsibility during his absence. 1

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 the positions for activities affecting the safety of the nuclear power plant.

the Connecticut Yankee Quality Assurance Program (CYQAP)

a. Lines of authority, responsibility, and communication shall be established and defined for 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 Quality Assurance Topical Report.~~

and resources

b. ~~The Unit Director~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. 1

An individual management position in the onsite organization

designated CYAPCO corporate officer,

c. The ~~Executive Vice President and Chief 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 plant to ensure nuclear safety. 1

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

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

b. At least one licensed Operator shall be in the control room when ~~fuel is in the reactor. In addition, while the facility is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the control room;~~

irradiated fuel is in the spent fuel pool;

during fuel handling operations;

c. An individual qualified in radiation protection procedures* shall be on site ~~when fuel is in the reactor;~~

d. All ~~CORE ALTERATIONS~~ fuel handling operations shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. DELETED

f. Administrative procedures shall be developed and implemented to limit the working hours of facility staff who perform safety-related functions. These procedures should follow the general guidance of the NRC Policy Statement on working hours (Generic Letter No. 82-12).

* The individual qualified in radiation protection procedures may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	~~MODE 1, 2, 3, or 4~~	~~MODE 5 or 6~~
SOP	~~2~~	1
ROP	~~2~~	1
AOP	~~2~~	2
~~STA~~	~~1*~~	~~None~~

Abbreviations:

SOP - Licensed Senior Reactor Operator
ROP - Licensed Reactor Operator
AOP - Additional Operator
~~STA - Shift Technical Advisor~~

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 unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

~~During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function~~. During any absence of the Shift ~~Supervisor~~ Manager from the control room, ~~while the unit is in MODE 5 or 6,~~ an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

~~* The STA position can be filled by either of the two Senior Reactor Operators (a dual-role individual), if he meets the requirements of Specification 6.3.1.2.1.~~

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

6.3.1.1 The position of Health Physics Manager shall meet the following minimum qualifications:

a. Academic degree in an engineering or science field or equivalent as per Section 6.3.1.1.c.

b. Minimum of five years professional technical experience in the area of radiological safety, three years of which shall be in applied radiation work in a nuclear facility dealing with problems similar to those encountered in a nuclear power reactor.

c. Technical experience in the area of radiological safety beyond the five year minimum may be substituted on a one-for-one basis towards the academic degree requirement (four years of technical experience being equivalent to a four year academic degree).

d. Academic and technical experience must total a minimum of nine years.

DELETED

6.3.1.2 ~~The position of Shift Technical Advisor (STA) shall meet the requirements of Specification 6.3.1.2.1 or 6.3.1.2.2.~~

1. Dual-role individual: Must hold a senior reactor operator's license at The Haddam Neck Plant, meet the STA training criteria of NUREG-0737, Item I.A.1.1, and meet one of the following educational alternatives:

   a. Bachelor's degree in engineering from an accredited institution;

   b. Professional Engineer's license obtained by the successful completion of the PE examination;

   c. Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences;

   d. Bachelor's degree in a physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences;

   e. Successful completion of the Memphis State University (MSU) STA program. (Note: This alternative is only acceptable for individuals who have completed the program prior to December 31, 1986); or

~~f. Successful completion of the Thames Valley State Technical College associate's degree in Nuclear Engineering Technology program, provided that the individual was enrolled in the program by October 1, 1987.~~

~~2. Dedicated STA: Must meet the STA training criteria of NUREG-0737, Item I.A.1.1, and have received specific training in plant design, and response and analysis of the plant for transients and accidents.~~

6.4 TRAINING

A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Unit Director and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10CFR55.59. The Director-Nuclear Training has the overall responsibility for the implementation of the Training Program.

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 Unit Director on all matters related to nuclear safety.

COMPOSITION

seven

6.5.1.2 The PORC shall be composed of at least ~~eleven~~ members. Members shall collectively have experience and expertise in the following areas:

~~Plant Operations~~
~~Engineering~~
~~Reactor Engineering~~
~~Mechanical Maintenance~~
~~Electrical Maintenance~~
~~Instrumentation and Controls~~
~~Health Physics~~
~~Work Planning and Control~~
~~Quality Services~~
~~Security~~
~~Chemistry~~

Plant Operations,
Decommissioning,*
Engineering,
Maintenance,
Health Physics,
Chemistry / Radiochemistry,
Quality Assurance, and
Security.*

The minimum qualifications of PORC members shall be that all members have an academic degree in an engineering or physical science field, or hold a management position, and have a minimum of five years technical experience in their respective field of expertise. The members of PORC shall be appointed in writing by the Unit Director, who is the PORC Chairperson. The alternate Chairpersons of the PORC shall be drawn from the PORC members and be appointed in writing by the Unit Director.

*These areas are exempt from the 5-year experience requirement.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairperson 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 Chairperson or his/her designated alternate.

QUORUM

6.5.1.5 The quorum of the PORC shall consist of the Chairperson or his/her designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of: (1) all procedures required by Specification 6.8 and changes thereto, and 2) any other proposed procedures or changes thereto as determined by the Unit Director to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to the Technical Specifications;

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

designated CYAPCO corporate officer

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the ~~Executive Vice President and Chief Nuclear Officer~~ and to the ~~Chairperson of the~~ Nuclear Safety Assessment Board;

f. Review of all REPORTABLE EVENTS;

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

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the ~~Chairperson of the~~ Nuclear Safety Assessment Board or the Unit Director.

i. Not used.

j. Not used.

k. Review of the Fire Protection Program and Implementing Procedures.

AUTHORITY

6.5.1.7 The PORC shall:

a. Report to and be advisory to the Unit Director on those areas of responsibility specified in Section 6.5.1.6(a) through (k);

b. Render determinations in writing to the Unit Director if any item considered under Specification 6.5.1.6a. through d., above, as appropriate and as provided by 10CFR50.59 or 10CFR50.92 constitutes an unreviewed safety question or requires a significant hazards consideration determination.

designated CYAPCO corporate officer

c. Provide written notification, meeting minutes may be used for this purpose, to the ~~Executive Vice President and Chief Nuclear Officer~~ and the Chairperson of the Nuclear Safety Assessment Board of disagreement between the PORC and the Unit Director; however, the ~~Executive Vice President and Chief Nuclear Officer~~ shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these Technical Specifications. A Copy shall be provided to the ~~Chairperson of the~~ Nuclear Safety Assessment Board.

6.5.2 NUCLEAR SAFETY ASSESSMENT BOARD (NSAB)

FUNCTION

6.5.2.1 The minimum qualifications of NSAB members are as follows:

a. The Chairperson and NSAB members shall have:

1. An academic degree in an engineering or physical science field, or hold a senior management position, and

2. A minimum of five years technical experience in their respective field of expertise.

b. The NSAB shall have experience in and shall function to provide independent oversight review and audit of designated activities in the areas of:

~~1. Nuclear power plant operations;~~

~~2. Nuclear engineering;~~

~~3. Chemistry and radiochemistry;~~

~~4. Metallurgy;~~

~~5. Instrumentation and control;~~

~~6. Radiological safety;~~

~~7. Mechanical and electrical engineering; and~~

~~8. Quality assurance practices.~~

Plant Operations,
Decommissioning,*
Engineering,
Radiological Safety,
Chemistry / Radiochemistry,
Quality Assurance, and
Environmental Protection.*

designated CYAPCO corporate officer

The NSAB serves to advise the ~~Executive Vice President and Chief Nuclear Officer~~ on matters related to nuclear safety and notify the ~~Executive Vice President and Chief Nuclear Officer~~ within 24 hours of a safety significant disagreement between the NSAB and the organization or function being reviewed.

COMPOSITION

6.5.2.2 The ~~Executive Vice President and Chief Nuclear Officer~~ shall appoint, in writing, a minimum of ~~seven~~ five members to the NSAB and shall designate from this membership, in writing, a Chairperson and a Vice Chairperson. The membership shall function to provide independent review and audit in the areas listed in Specification 6.5.2.1.

ALTERNATES

6.5.2.3 All alternate members shall be appointed, in writing, by the ~~Executive Vice President and Chief Nuclear Officer~~; however, no more than two alternates shall participate as members in NSAB activities at any one time.

MEETING FREQUENCY

6.5.2.4 The NSAB shall meet at least once per calendar quarter.

QUORUM

Haddam Neck Plant.

6.5.2.5 The quorum of the NSAB shall consist of a majority of NSAB members including the Chairperson or Vice Chairperson. No more than a minority of the quorum shall have line responsibility for operation of the ~~same Northeast Utilities' nuclear unit~~. No more than two alternates shall be appointed as members at any meeting in fulfillment of the quorum requirements.

*These areas are exempt from the 5-year experience requirement.

REVIEW RESPONSIBILITIES

6.5.2.6 The NSAB shall be responsible for the review of:

a. The safety evaluations for changes to procedures, equipment, or systems, and tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question as defined in 10 CFR 50.59;

b. Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR 50.59;

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

e. Violations of applicable codes, regulations, orders, license requirements, or internal procedures having nuclear safety significance;

f. All Licensee Event Reports required by 10 CFR 50.73;

g. Indications of significant unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect nuclear safety;

h. Significant accidental, unplanned, or uncontrolled radioactive releases, including corrective actions to prevent recurrence;

i. Significant operating abnormalities or deviations from normal and expected performance of equipment that could affect nuclear safety;

j. The performance of the corrective action program; and

k. Audits and ~~audit plans~~ the Annual Audit Plan.

Reports or records of these reviews shall be forwarded to the designated CYAPCO corporate officer ~~Executive Vice President and Chief Nuclear Officer~~ within 30 days following completion of the review.

AUDIT PROGRAM RESPONSIBILITIES

6.5.2.7 The NSAB audit program shall be the responsibility of Nuclear Oversight organization. NSAB audits shall be performed at least once per 24 months in accordance with appropriate administrative procedures ~~Nuclear Group Procedures~~ and shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The training and qualifications of the unit staff;

c. The implementation of all programs required by Specification 6.8;

d. The Fire Protection Program and implementing procedures;

e. The fire protection equipment and program implementation utilizing either a qualified offsite license fire protection engineer or an outside independent fire protection consultant.

designated CYAPCO corporate officer

f. Actions taken to correct deficiencies occurring in equipment, structures, systems, components, or method of operation that affect nuclear safety; and

g. Other activities and documents as requested by the ~~Executive Vice President and Chief Nuclear Officer~~.

RECORDS

6.5.2.8 Written records of reviews and audits shall be maintained. As a minimum these records shall include: results

a. Results of the activities conducted under the provisions of Section 6.5.2;.

b. ~~Recommended approval or disapproval of items considered under Specifications 6.5.1.6.a through 6.5.1.6.d, and 6.8; and~~

c. ~~Determination as to whether each item considered under Specification 6.5.1.6.a, 6.5.1.6.b, 6.5.1.6.d, and 6.8 constitute an unreviewed safety question as defined in 10 CFR 50.59.~~

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

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

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of this review shall be submitted to the ~~Chairperson of the~~ NSAB and the ~~Executive Vice President and Chief Nuclear Officer~~.

6.7 ~~SAFETY LIMIT VIOLATION~~ DELETED

~~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 1 hour. The Executive Vice President and Chief Nuclear Officer and the Chairperson of the NSAB shall be notified within 24 hours;~~

~~c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. 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 Chairperson of the NSAB and the Executive Vice President and Chief Nuclear Officer within 14 days of the violation;~~

~~e. Operation will not be resumed until authorized by the Commission.~~

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures and/or administrative policies 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. The requirements and recommendations of Sections 5.1 and 5.3 of ANSI N 18.7-1976.

c. Fire Protection Program implementation.

d. Quality controls for effluent monitoring, using the guidance in Regulatory Guide 1.21 Rev. 1, June 1974.

e. RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM) implementation except for Section I.E, Radiological Environmental Monitoring.

f. ~~PROCESS CONTROL PROGRAM~~ implementation.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the PORC and shall be approved by the Unit Director prior to implementation and reviewed periodically as set forth in each document or in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 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 Operator license on the unit affected; and

c. The change is documented, reviewed by the PORC and approved by the Unit Director within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained covering Section I.E., Radiological Environmental Monitoring, of the REMODCM.

6.8.5 All procedures and procedure changes required for the Radiological Environmental Monitoring Program of Specification 6.8.4 above shall be reviewed by an individual (other than the author) from the Radiological Assessment Branch or the Production Operation Services Laboratory (POSL) and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the Radiological Assessment Branch or the POSL, within 14 days of implementation.

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, one copy to the appropriate Regional Office of the NRC, and one copy to the appropriate NRC Resident Inspector, unless otherwise noted.

~~STARTUP REPORT~~

DELETED

6.9.1.1 ~~A summary report of plant startup and power escalation testing shall be submitted following: (1) amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) 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 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 power 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 facility as described below for the previous calendar year shall be submitted prior to March 1 of each year.

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 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, thermoluminescent 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 shall be assigned to specific major work functions;

DELETED

b. ~~Documentation of all challenges to the pressurizer power-operated relief valves (PORVs) and safety valves; and~~

c. ~~The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. 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.~~

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine Annual Radiological Environmental Operating Reports covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include that information delineated in the REMODCM.

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* This tabulation supplements the requirements of Section ~~20.407~~ 20.2206 of 10 CFR Part 20.

ANNUAL RADIOACTIVE EFFLUENT REPORT

6.9.1.7 A routine Annual Radioactive Effluent Report covering the operation of the facility during the previous calendar year of operation shall be submitted by May 1 of each year.

The report shall include that information delineated in the REMODCM.

Any changes to the REMODCM shall be submitted in the Annual Radioactive Effluent Report.

~~MONTHLY OPERATING REPORTS~~

DELETED

6.9.1.8 ~~Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with one copy to the appropriate Regional Office of the NRC, and one copy to the appropriate NRC Resident Inspector, no later than the 15th of each month following the calendar month covered by the report.~~

~~TECHNICAL REPORT SUPPORTING CYCLE OPERATION~~

6.9.1.9 a ~~Core operating limits shall be established and documented in the TECHNICAL REPORT SUPPORTING CYCLE OPERATION before each reload cycle or any remaining part of a reload cycle for the following:~~

1. ~~Moderator Temperature Coefficient for Specification 3.1.1.5.~~
2. ~~Moveable Control Assemblies--Bank Height for Specification 3.1.3.1.~~
3. ~~Control Group Insertion Limits--Four Loops Operating, control banks for Specification 3.1.3.6.1.~~
4. ~~Control Group Insertion Limits--Three Loops Operating, control banks for Specification 3.1.3.6.2.~~
5. ~~Axial Offset--Four Loops Operating for Specifications 3/4.2.1.1.~~
6. ~~Axial Offset--Three Loops Operating for Specifications 3/4.2.1.2.~~
7. ~~Linear Heat Generation Rate--Four Loops Operating for Specifications 3/4.2.2.1.~~
8. ~~Linear Heat Generation Rate--Three Loops Operating for Specifications 3/4.2.2.2.~~

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6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^{N}$--Specification 3/4.2.3.1.

6.9.1.9.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. F. M. Akstulewicz to E. J. Mroczka, "Review of NUSCO Topical Report on Physics Methodology for PWR Reload Design (NUSCO-152)," August 3, 1987.

2. A. B. Wang to E. J. Mroczka, "Safety Evaluation for Northeast Utilities Topical Report 140-1, NUSCO Thermal Hydraulic Qualification, Volume I (RETRAN)," July 26, 1988.

3. F. M. Akstulewicz to J. F. Opeka, "NUSCO Thermal Hydraulic Model Qualification, Volume II (VIPRE), Topical Report NUSCO 140-2," October 16, 1986.

4. A. B. Wang to E. J. Mroczka, "Safety Evaluation of Northeast Utilities Topical Report 151, Haddam Neck Non-LOCA Transient Analysis," October 18, 1988.

5. Supplement to the Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission Docket No. 50-213, Connecticut Yankee Atomic Power Company, Haddam Neck Plant, December 27, 1974.

6. A. B. Wang to J. F. Opeka, "Issuance of Amendment (TAC Nos. M80864, M82284, M66958, M71769, and M72083)," Part 3 of Safety Evaluation--Large-Break Loss-of-Coolant Accident Analysis Zircaloy Clad Fuel Evaluation, dated January 17, 1992.[1]

7. A. B. Wang to J. F. Opeka, "Issuance of Amendment (TAC Nos. M80864, M82284, M66958, M71769, and M72083)," Part 4 of Safety Evaluation--Small-Break Loss-of-Coolant Accident Analysis Zircaloy Clad Fuel Evaluation, dated January 17, 1992.

8. A. B. Wang to J. F. Opeka, "Review of NUSCO Topical Report on Physics Methodology for PWR Reload Design (NUSCO-152) Addendum 3," March 10, 1995.

9. A. B. Wang to J. F. Opeka, Haddam Neck Plant - Safety Evaluation for Topical Report, "VIPRE/WRB-1 DNBR Thermal Limit for Westinghouse Fuel Types," March 3, 1995.

6.9.1.9.c The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin and transient and accident limits) of the safety analysis are met.

6.9.1.9.d The TECHNICAL REPORT SUPPORTING CYCLE OPERATION, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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(1) Use of the peak clad temperature margin for conformance with SECY 83-472 is described in the Cycle 19 Technical Report Supporting Cycle Operation, Revision 1.

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 one copy to the appropriate Regional Office of the NRC, and one copy to the appropriate NRC Resident Inspector, within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of facility 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;

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

e. Records of ~~reactor~~ tests and experiments;

f. Records of changes made to the 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; and

6.10 RECORD RETENTION (Cont)

6.10.3 The following records shall be retained for the duration of the facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the FSAR;

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

c. Records of facility radiation and contamination surveys;

d. Records of radiation ~~expsure~~ for all individuals entering radiation control areas; exposure

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

of the reactor vessel;

f. Records of transient or operational cycles ~~for those facility components identified in Table 5.7-1;~~

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

under previous amendments

h. Records of inservice inspections performed ~~pursuant~~ to these Technical Specifications;

i. Records of quality assurance activities required by the Quality Assurance Manual not listed in Specification 6.10.2;

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

k. Records of meetings of the PORC and the NSAB; 1

l. Records for Environmental Qualification.

m. Records of reviews performed for changes made to the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) and the Process Control Program.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

20.1601(c) equal to or 20.1601

6.12.1 Pursuant to paragraph ~~20.203(c)(5)~~ 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph ~~20.203(c)~~ 20.1601, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mR/h but equal to or less than 1000 mR/h at ~~45 cm (18 in.)~~ 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physics Manager in the RWP.

1

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at ~~45 cm (18 in.)~~ 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device or continually guarded.

6.13 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

(REMM)

Section I, ~~Radiological~~ ~~Effluents~~ ~~Monitoring~~ ~~Manual~~ shall outline the sampling and analysis programs to determine the concentration of radioactive materials released offsite as well as dose commitments to individuals in those exposure pathways and for those radionuclides released as a result of facility operation. It shall also specify operating guidelines for RADIOACTIVE WASTE TREATMENT SYSTEMS and report content.

(ODCM)

Section II, the ~~Offsite~~ ~~Dose~~ ~~Calculation~~ ~~Manual~~, shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculations of gaseous and liquid effluent monitoring instrumentation Alarm/Trip Setpoints consistent with the applicable LCO's contained in these Technical Specifications.

Changes to the REMODCM:

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

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

   2) 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 10CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by PORC and the approval of the Unit Director. 1

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire REMM or ODCM, as appropriate, as a part of or concurrent with the Annual Radioactive Effluent Report for the period of the report in which any change 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.

6.14 RADIOACTIVE WASTE TREATMENT

Procedures for liquid and gaseous radioactive effluent discharges from the facility shall be prepared, approved, maintained and adhered to for all operations involving offsite releases of radioactive effluents. These procedures shall specify the use of appropriate RADIOACTIVE WASTE TREATMENT SYSTEMS utilizing the guidance provided in the REMODCM.

The Solid RADIOACTIVE WASTE TREATMENT SYSTEM shall be operated in accordance with the ~~PROCESS CONTROL PROGRAM~~ to process wet radioactive wastes to meet shipping and burial ground requirements.

6.15 ~~SYSTEMS INTEGRITY~~ DELETED

~~The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:~~

~~a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and~~

~~b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.~~

6.16 ~~PASS/SAMPLING AND ANALYSIS OF PLANT EFFLUENTS~~ DELETED

1

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

~~a. Training of personnel~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.~~

Docket No. 50-213
CY-97-006

Attachment 2

Haddam Neck Plant

Proposed Revision To Operating License And Technical Specifications

Defueled Operating License And Technical Specifications

Retyped Pages

May 1997

CONNECTICUT YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-213

HADDAM NECK PLANT

FACILITY OPERATING LICENSE

License No. DPR-61

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license, as amended, filed by Connecticut Yankee Atomic Power Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Haddam Neck Plant (facility) has been substantially completed in conformity with Construction Permit No. CPPR-14 and the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. DPR-16 is in accordance with Appendix D to 10 CFR Part 50 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70.

2. Facility Operating License No. DPR-61 is hereby issued to the Connecticut Yankee Atomic Power Company to read as follows:

A. This license applies to the Haddam Neck Plant, a pressurized lightwater reactor and associated equipment (the facility) owned by the Connecticut Yankee Atomic Power Company. The facility is located on Connecticut Yankee's Haddam Neck site on the east bank of the Connecticut River, approximately 21 miles south-southeast of Hartford, in the Town of Haddam, Middlesex County, Connecticut, and is described in the "Facility Description and Safety Analysis Report" as supplemented and amended (Amendment 10 through Amendment 25 to the License Application) and the "Environmental Report" (as supplemented by Amendment 1).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Connecticut Yankee Atomic Power Company:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in the Town of Haddam, Middlesex County, Connecticut, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Facility Description and Safety analysis as supplemented and amended; or as described in any amendment to this license;

(3) CYAPCO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup (possession only), sealed sources for reactor instrumentation (possession only) and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) CYAPCO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) CYAPCO, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear material as may be produced by the operations of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is not authorized to operate the reactor. Fuel may not be placed in the reactor vessel.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.   , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Deleted by Amendment No. 29.

(4) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER, dated October 3, 1978, and Supplements dated February 6, 1981, November 11, 1981, November 14, 1984, November 27, 1987, January 20, 1990, April 10, 1990, August 14, 1990, June 27, 1991, October 16, 1991, November 21, 1991, and February 1, 1995, subject to the following provisions.

The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment which could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

(5) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Haddam Neck Plant Physical Security Plan," with revisions submitted through January 24, 1989; "Haddam Neck Plant Guard Training and Qualification Plan," with revisions submitted through January 27, 1983; and "Haddam Neck Plant Safeguards Contingency Plan," with revisions submitted through December 9, 1983. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

(6) Integrated Implementation Schedule

a. The Connecticut Yankee Atomic Power Company shall implement and maintain in effect the Integrated Implementation Schedule Program Plan (the Program Plan) to be followed for scheduling of plant modifications and engineering studies. The Program Plan shall be followed from and after the effective date of this license condition.

b. This license condition shall be effective for three years from the date of issuance of Amendment No. 183. (Date of Issuance February 23, 1995)

(7) Fuel Movement

The movement of special nuclear material used as reactor fuel into the containment is prohibited.

D. This license is effective as of the date of issuance and shall expire at midnight, June 29, 2007.

FOR THE ATOMIC ENERGY COMMISSION

A. Giambusso, Deputy Director for
Reactor Projects
Directorate of Licensing

Original Signed by A. Giambusso

Enclosure:
Appendices A and B - Technical Specifications
Date of Issuance: December 27, 1974

DEFINITIONS

DEFINITIONS

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SECTION PAGE

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

DESIGN FEATURES

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SECTION 1.0
DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ANALOG CHANNEL OPERATIONAL TEST

1.2 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

1.3 DELETED

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions, and may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

1.6 DELETED

1.7 DELETED

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel, with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe position.

1.9 DELETED

1.10 DELETED

1.11 DELETED

FREQUENCY NOTATION

1.12 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

1.13 DELETED

1.14 DELETED

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

1.16 DELETED

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

1.19 DELETED

1.20 DELETED

1.21 DELETED

1.22 DELETED

RADIOACTIVE WASTE TREATMENT SYSTEMS

1.23 RADIOACTIVE WASTE TREATMENT SYSTEMS are those liquid, gaseous and solid waste systems which are required to maintain control over radioactive material in order to meet the LCO's set forth in those specifications.

RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

1.24 A RADIOLOGICAL EFFLUENT MONITORING MANUAL (REMM) shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures to individuals from station operation. An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation Alarm/Trip Setpoints. Requirements of the REMODCM are provided in Specification 6.13.

1.25 DELETED

REPORTABLE EVENT

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

RHR LOOP

1.27 An RHR LOOP consists of a Residual Heat Removal (RHR) pump, a dedicated RHR heat exchanger either from the same train or from the opposite train and all other necessary piping and components required to receive and cool reactor coolant.

1.28 DELETED

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.31 DELETED

1.32 DELETED

1.33 DELETED

1.34 DELETED

1.35 DELETED

1.36 DELETED

1.37 DELETED

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
B	At least once per 14 days.
M	At least once per 31 days.
SW	At least once per 42 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
P	Prior to each release.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. DELETED			
2. DELETED			
3. DELETED			
4. DELETED			
5. DELETED			
6. REFUELING**	less than or equal to 0.94	0	less than or equal to 140°F

---

* Excluding decay heat.
** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SECTION 2.0

SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

SECTION 2.0
AND
BASES
FOR
SECTION 2.0
SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

DELETED IN THEIR ENTIRETY

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the specified applicable conditions; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 DELETED

3.0.4 Entry into a specified applicable condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. Exceptions to these requirements are stated in the individual specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the specified applicable conditions for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into a specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

SECTIONS 4.0.5, 4.0.6;

TABLES 4.0-1, 4.0-2;

FIGURE 4.0-1; AND

PAGES 3/4 0-3 THROUGH 3/4 0-8

DELETED

SECTIONS 3.1 AND 4.1

REACTIVITY CONTROL SYSTEMS

DELETED IN THEIR ENTIRETY

SECTIONS 3.2 AND 4.2

POWER DISTRIBUTION LIMITS

DELETED IN THEIR ENTIRETY

SECTIONS 3.3.1 THROUGH 3.3.3.2;

SECTIONS 4.3.1 THROUGH 4.3.3.2;

TABLES 3.3-1 THROUGH 3.3-4;

TABLES 4.3-1 THROUGH 4.3-3; AND

PAGES 3/4 3-1 THROUGH 3/4 3-26

DELETED

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring system instrumentation shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the seismic monitoring system inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the system to OPERABLE status.

b. DELETED

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 The above seismic monitoring system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 The above required seismic monitoring system actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. If it is determined that the magnitude of the event exceeded the Operating Basis Earthquake, then a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

TABLE 3.3-5

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Servo Accelerometer (SSA-302) Basemat-Cable Vault	0 to 0.5g	1
2. Digital Cassette Accelerograph (DCA-300)**	± 5 Volts	1
3. Response Spectrum Analyzer (RSA-50)**	± 5 Volts	1
4. Playback System (SMR-102)**	± 5 Volts	1
5. Seismic Warning Panel (SWP-300)**	N.A.	1

**All located in the Control Room

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Servo Accelerometer (SSA-302) Basemat-Cable Vault	M	R	SA
2. Digital Cassette Accelerograph (DCA-300)**	M	R	SA
3. Response Spectrum Analyzer (RSA-50)**	M	R	SA
4. Playback System (SMR-102)**	M	R	SA
5. Seismic Warning Panel (SWP-300)**	M	R	SA

**All located in the Control Room

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. DELETED |

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

TABLE 3.3-6

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT/LOCATION	MINIMUM OPERABLE
1. Wind Speed	
a. Baseline Elev. 33′	1
b. Nominal Elev. 200′	1
2. Wind Direction	
a. Baseline Elev. 33′	1
b. Nominal Elev. 196′	1
3. Air Temperature	
a. Baseline Elev. 33′	1
4. Delta T*	
a. Nominal Elev. 120′	1
b. Nominal Elev. 200′	1

* Delta T is the air temperature of the nominal elevation minus the air temperature at the 33′ baseline elevation.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT/LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Baseline Elev. 33′	D	SA
b. Nominal Elev. 200′	D	SA
2. Wind Direction		
a. Baseline Elev. 33′	D	SA
b. Nominal Elev.196′	D	SA
3. Air Temperature		
a. Baseline Elev. 33′	D	SA
4. Delta T*		
a. Nominal Elev. 120′	D	SA
b. Nominal Elev. 200′	D	SA

* Delta T is the air temperature of the nominal elevation minus the air temperature at the 33′ baseline elevation.

SECTIONS 3.3.3.5, 4.3.3.5;

TABLES 3.3-7, 3.3-8; 4.3-6;

AND

PAGES 3/4 3-33 THROUGH 3/4 3-43

DELETED

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with applicable Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints shall be determined in accordance with methodology and parameters described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times*.

ACTION:

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

b. With the number of channels less than the minimum channels operable requirement, take the ACTION shown in Table 3.3-9. Exert best efforts to restore the inoperable monitor to OPERABLE status within 30 days, and, if unsuccessful, explain in the next Annual Effluent Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

c. DELETED

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-7.

---

* At all times means that channel shall be OPERABLE and in service on a continuous, uninterrupted basis, except that outages of monitoring channels are permitted for a maximum of 12 hours each time for the purposse of maintenance and performance of requwed tests, checks, calibrations or sampling.

TABLE 3.3-9

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM # OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Waste and Recycle Test Tank Discharge Line	1	46
b. DELETED		
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water Effluent Line	1	48
3. FLOW RATE MEASUREMENT		
a. Waste and Recycle Test Tank Discharge Line	1	49
b. DELETED		
c. Discharge Canal	*	N.A.

* Discharge canal flow is determined by the use of pump curves.

TABLE 3.3-9 (Continued)

ACTION STATEMENTS

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, effluent releases may continue provided that best efforts are made to repair the instrument and that prior to initiating a release:

a. At least two independent samples of the tank to be discharged are analyzed in accordance with Specification 4.11.1.1.1, and;

b. The original release rate calculations and discharge valving are independently verified by a second individual.

ACTION 47 - DELETED

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that once per 12 hours grab samples of the service water effluent are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 3×10^{-7} microcuries/ml.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that the flow rate is estimated once per 4 hours during actual releases. Pump performance curves generated insitu may be used to estimate flow.

TABLE 4.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Waste and Recycle Test Tank Discharge Line	D(1)	P	R(2)	Q(3)
b. DELETED				
2. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water Effluent Line	D(1)	M	R(2)	Q(3)
3. FLOW RATE MEASUREMENT				
a. Waste and Recycle Test Tank Discharge Line	D(1)	N.A.	R	N.A.
b. DELETED				
c. Discharge Canal	D(4)	N.A.	N.A.	N.A.

TABLE NOTATION

(1) CHANNEL CHECK need only be performed daily when discharges are made from this pathway. The CHANNEL CHECK should be done when the discharge is in process.

(2) CHANNEL CALIBRATION shall be performed using a known radioactive liquid or solid source whose strength is determined by a detector which has been calibrated to an NBS source. The radioactive source shall be in a known, reproducible geometry.

(3) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.*

2. Instrument indicates a downscale failure or circuit failure.

3. Instrument controls not set in operate mode.

(4) Pump status should be checked at least once per 24 hours for the purpose of determining flow rate.

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* Automatic isolation shall also be demonstrated annually for the test tank discharge monitor line.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE with applicable Alarm Setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology and parameters as described in the ODCM.

APPLICABILITY: At all times*.

ACTION:

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

b. With the number of channels less than the minimum channels operable requirement, take the ACTION shown in Table 3.3-10. Exert best efforts to restore the inoperable monitor to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Effluent Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

c. DELETED

SURVEILLANCE REQUIREMENT

4.3.3.8.1 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

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* At all times means that the channel shall be OPERABLE and in service on a continuous basis, except that outages of monitoring channels are permitted for a maximum of 12 hours each time for the purpose of maintenance and performance of required tests, checks, calibrations.

TABLE 3.3-10

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. MAIN STACK		
a. Noble Gas Activity Monitor Providing Alarm	1	50
b. DELETED		
c. Particulate Sampler	1	51
d. Stack Flow Rate Monitor	1	52
e. Sampler Flow Rate Monitor	1	52

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 50 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases via the Waste Gas Holdup System may continue provided that best efforts are made to repair the instrument and that prior to initiating the release:

(a) For the tank to be discharged, at least two independent samples of the tank's contents are analyzed; and,

(b) The release rate calculations and discharge valve lineups are independently verified by a second individual.

Otherwise, suspend releases from the Waste Gas Holdup System.

Releases from all pathways other than the Waste Gas Holdup System may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross radioactivity within 24 hours.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that samples are continuously collected with auxiliary sampling equipment for periods of seven (7) days and analyzed for principal gamma emitters with half lives greater than 8 days within 48 hours after the end of the sampling period. Auxiliary sampling shall be established within 12 hours of declaring the channel INOPERABLE.

ACTION 52 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that the best efforts are made to repair the instrument and that the flow rate is estimated once per 4 hours.

TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. MAIN STACK				
a. Noble Gas Activity Monitor	D(1)	M	R(2)	Q(3)
b. DELETED				
c. Particulate Sampler	W	N.A.	N.A.	N.A.
d. Stack Flow Rate Monitor	D(1)	N.A.	R	N.A.
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) CHANNEL CHECK daily when releases exist via this pathway.

(2) Calibration shall be performed using a known source whose strength is determined by a detector which has been calibrated to an NBS source. These sources shall be in a known, reproducible geometry.

(3) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the Alarm Setpoint.

b. Instrument indicates a downscale failure or circuit failure.

c. Instrument controls not set in operate mode.

SECTIONS 3.3.3.9, 3.3.4, 4.3.3.9.1, 4.3.4;

TABLE 3.3-11; AND

PAGES 3/4 3-54 THROUGH 3/4 3-57

DELETED

SECTIONS 3.4 AND 4.4

REACTOR COOLANT SYSTEM

DELETED IN THEIR ENTIRETY

SECTIONS 3.5 AND 4.5

EMERGENCY CORE COOLING SYSTEMS

DELETED IN THEIR ENTIRETY

SECTIONS 3.6 AND 4.6

CONTAINMENT SYSTEMS

DELETED IN THEIR ENTIRETY

SECTIONS 3.7.1 THROUGH 3.7.4;
SECTIONS 4.7.1 THROUGH 4.7.4;
TABLES 3.7-1 THROUGH 3.7-2;
TABLES 4.7-1 THROUGH 4.7-2; AND
PAGES 3/4 7-1 THROUGH 3/4 7-13

DELETED

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.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and either:

   1. Decontaminate and repair the sealed source, or

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

b. DELETED

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.

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 6 months for all sealed sources containing radioactive materials:

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

   2) In any form other than gas.

3/4.7.5 SEALED SOURCE CONTAMINATION

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested following repair or maintenance to the source.

4.7.5.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removal contamination.

SECTIONS 3.7.6 THROUGH 3.7.12;

SECTIONS 4.7.6 THROUGH 4.7.12;

TABLES 3.7-3 THROUGH 3.7-6; AND

PAGES 3/4 7-16 THROUGH 3/4 7-36

DELETED

SECTIONS 3.8 AND 4.8

ELECTRICAL POWER SYSTEMS

DELETED IN THEIR ENTIRETY

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled unisolated portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure a K_{eff} of 0.94 or less.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing a minimum of 14000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.94.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The above reactivity condition shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any control rod in excess of 3 feet from its fully inserted position within the reactor vessel except when control rod drag testing is being performed, at which time the reactivity condition shall be determined prior to the first control rod withdrawn in excess of 3 feet and every 4 hours thereafter until completion of the testing.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2.a As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room when CORE ALTERATIONS or positive reactivity changes are taking place. When CORE ALTERATIONS or positive reactivity changes are not taking place, at least one Source Range Neutron Flux Monitor shall be OPERABLE and operating with a visual indication in the control room and audible indication in the containment.

3.9.2.b As a minimum, two Source Range High Neutron Level Alarms (Containment Evacuation) shall be OPERABLE and operating with a minimum logic to audibly alarm in both the control room and containment of one (1) of two (2).

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3.a The reactor shall be subcritical for at least 100 hours.

3.9.3.b Irradiated fuel shall remain in the reactor vessel for at least the time required in Figure 3.9-1 prior to movement to the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.*

ACTION:

a. With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

b. With the combination of reactor hold time and cooling water temperature in the unacceptable region of Figure 3.9-1, suspend all operations involving movement of irradiated fuel from the reactor vessel to the spent fuel pool until the combination of reactor hold time and cooling water temperature return to the acceptable region of Figure 3.9-1.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

4.9.3.2 Verify that the combination of reactor hold time and cooling water temperature are in the acceptable region of Figure 3.9-1 prior to and at least once per 12 hours during transfer of irradiated fuel from the reactor vessel to the spent fuel pool.

*Specification 3.9.3.b does not apply if less than 53 fuel assemblies are transferred to the spent fuel pool.

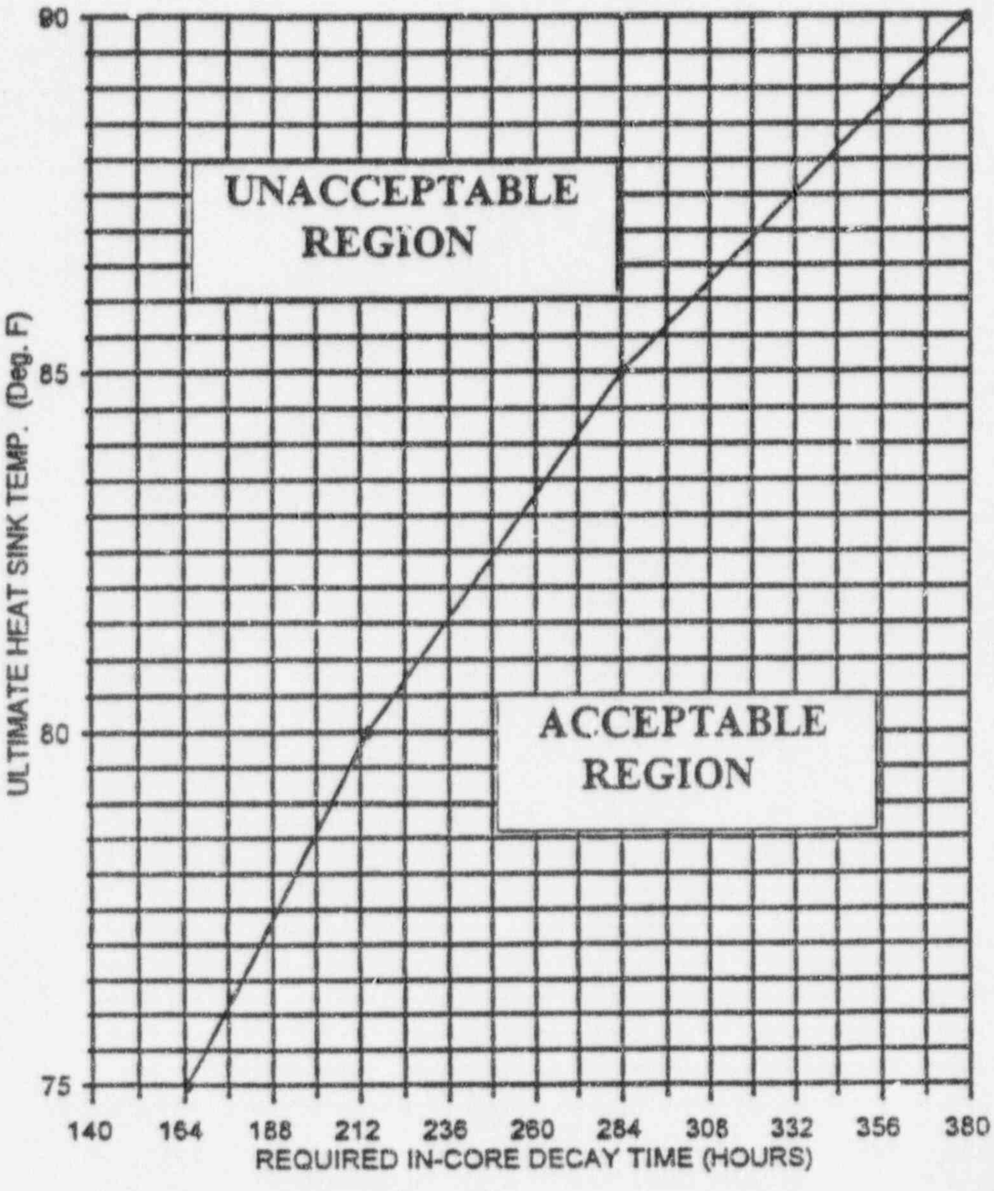

FIGURE 3.9-1 REQUIRED REACTOR HOLD TIME FOR A FULL CORE OFFLOAD

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch installed and secured in place,

b. A minimum of one door in the airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be, either:

   1) Closed by an isolation valve, blind flange, manual valve, or special device, or

   2) Be capable of being closed by an OPERABLE containment purge supply, purge exhaust, or purge exhaust bypass isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building, other than placing an irradiated fuel assembly into a safe storage location.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE containment purge supply, purge exhaust, or purge exhaust bypass isolation valve or capable of being closed by an OPERABLE airlock door within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations referred to in Specification 3.9.4c are in their closed/isolated condition,

b. Testing the containment purge supply, purge exhaust, and purge exhaust bypass isolation valves in accordance with Specification 4.9.9, and

c. Verifying that at least one of the doors in the airlock is capable of being closed.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling cavity manipulator crane area.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building other than placing an irradiated fuel assembly into a safe storage location.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling cavity manipulator crane area shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rod drive shafts* or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1) A minimum capacity of 1650 pounds, and

2) An overload cut off limit less than or equal to 175 pounds above the indicated weight for wet conditions for a stainless steel clad fuel assembly with a full length RCCA inserted.

b. The auxiliary hoist used for latching and unlatching control rod drive shafts having a minimum capacity of 1500 pounds.

APPLICABILITY: During movement of control rod drive shafts or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, place any fuel assembly or control rod in transit into a safe storage location and suspend use of the inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rod drive shafts or fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2063

exceeds 175 pounds above the indicated weight for wet conditions for a stainless steel clad fuel assembly with a full length RCCA inserted.

4.9.6.2 Each auxiliary hoist used for movement of control rod drive shafts within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations. If loads greater than the weight of a stainless steel clad fuel assembly with an RCCA inserted are to be lifted within the reactor vessel with the auxiliary hoist then a load test will be performed for 125% of the weight of the load to be lifted.

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* The RCCA's may be attached to the control rod drive shafts during some refueling evolutions.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1650 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Administrative controls that prevent the travel of loads in excess of 1650 pounds over fuel assemblies shall be in place prior to lifting a load in excess of 1650 pounds.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one RHR LOOP shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level is greater than or equal to 23 feet above the top of the reactor vessel flange.

ACTION:

With no RHR LOOP OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR LOOP to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least once per 12 hours verify at least one RHR LOOP is in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm.

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* The RHR LOOP may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent RHR LOOPS shall be OPERABLE, and at least one RHR LOOP shall be in operation.

APPLICABILITY: MODE 6 when the water level is less than 23 feet above the top of the reactor vessel flange.

ACTION:

a. With less than the required RHR LOOPS OPERABLE, immediately initiate corrective action to return the required RHR LOOPS to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR LOOP in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR LOOP to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2.1 At least once per 12 hours verify at least one RHR LOOP is in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm.

4.9.8.2.2 The RHR LOOP not in operation shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability.

3/4.9.9 CONTAINMENT PURGE SUPPLY, PURGE EXHAUST, AND PURGE EXHAUST BYPASS ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge supply, purge exhaust, and purge exhaust bypass isolation valves shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the requirements of the above specification not safistied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building, other than placing an irradiated fuel assembly into a safe storage location.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge supply, purge exhaust, and purge exhaust bypass isolation valves shall be verified OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS and prior to movement of irradiated fuel within containment by verifying that containment purge, bypass or exhaust isolation valves are accessible for manual operation or verifying that the associated penetrations are blind flanged.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, place any fuel assembly or control rod in transit into a safe storage location and suspend all further operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 20 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movements of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 FUEL STORAGE BUILDING AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Storage Building Air Cleanup System shall be OPERABLE and in operation.

APPLICABILITY: During operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool.*

ACTION:

a. With the Fuel Storage Building Air Cleanup System inoperable, or not operating, suspend all operations involving movements of fuel within the storage pool or crane operation with loads over the storage pool.*

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The Fuel Storage Building Air Cleanup System shall be demonstrated OPERABLE and in operation:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm $\pm$ 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10% at test conditions of 86°F, 95% relative humidity atmospheric pressure, and 40 feet/min face velocity in accordance with ASTM D3803; and

   3) Verifying a system flow rate of 4000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1980.

*The above specification does not apply during movement of fuel storage rack modules into or out of the SFP during the Cycle 19 rerack, provided an approved safe loading path is established with no loads over stored fuel assemblies.

SURVEILLANCE REQUIREMENTS (Continued)

b. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10% at test conditions of 86°F, 95% relative humidty, atmospheric pressure, and 40 feet/min. face velocity in accordance with ASMT D3803.

c. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%, and

   2) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than 0 inch Water Gauge differential relative to the outside atmosphere during system operation.

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

e. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

3/4.9.13 MOVEMENT OF FUEL IN SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.13 Prior to movement of a fuel assembly in the spent fuel pool, the boron concentration of the pool shall be maintained uniform and sufficient to maintain a boron concentration of greater than or equal to 800 ppm.

APPLICABILITY: Whenever a fuel assembly is moved in the spent fuel pool.

ACTION:

With the boron concentration less than 800 ppm, suspend the movement of all fuel in the spent fuel pool.

SURVEILLANCE REQUIREMENT

4.9.13 Verify that the boron concentration is greater than or equal to 800 ppm within 24 hours prior to any movement of a fuel assembly in the spent fuel pool and every 72 hours thereafter.

3/4.9.14 SPENT FUEL POOL--REACTIVITY CONDITION

LIMITING CONDITION FOR OPERATION

3.9.14 The Reactivity Condition of the spent fuel pool shall be such that K_{eff} is less-than-or-equal-to 0.95 at all times.

APPLICABILITY: Whenever fuel is in the spent fuel pool.

ACTION:

Immediately initiate actions to correct the loading error if the placement of fuel assemblies does not meet the requirements of Figure 3.9-2, Figure 3.9-3, and Figure 3.9-4.

SURVEILLANCE REQUIREMENT

4.9.14 Ensure that all fuel assemblies to be placed in the spent fuel pool are within the enrichment and burn-up limits of Figure 3.9-2 (Region 2) and Figure 3.9-3 (Region 3) by checking the assembly's design and burn-up documentation. Region locations are shown in Figure 3.9-4.

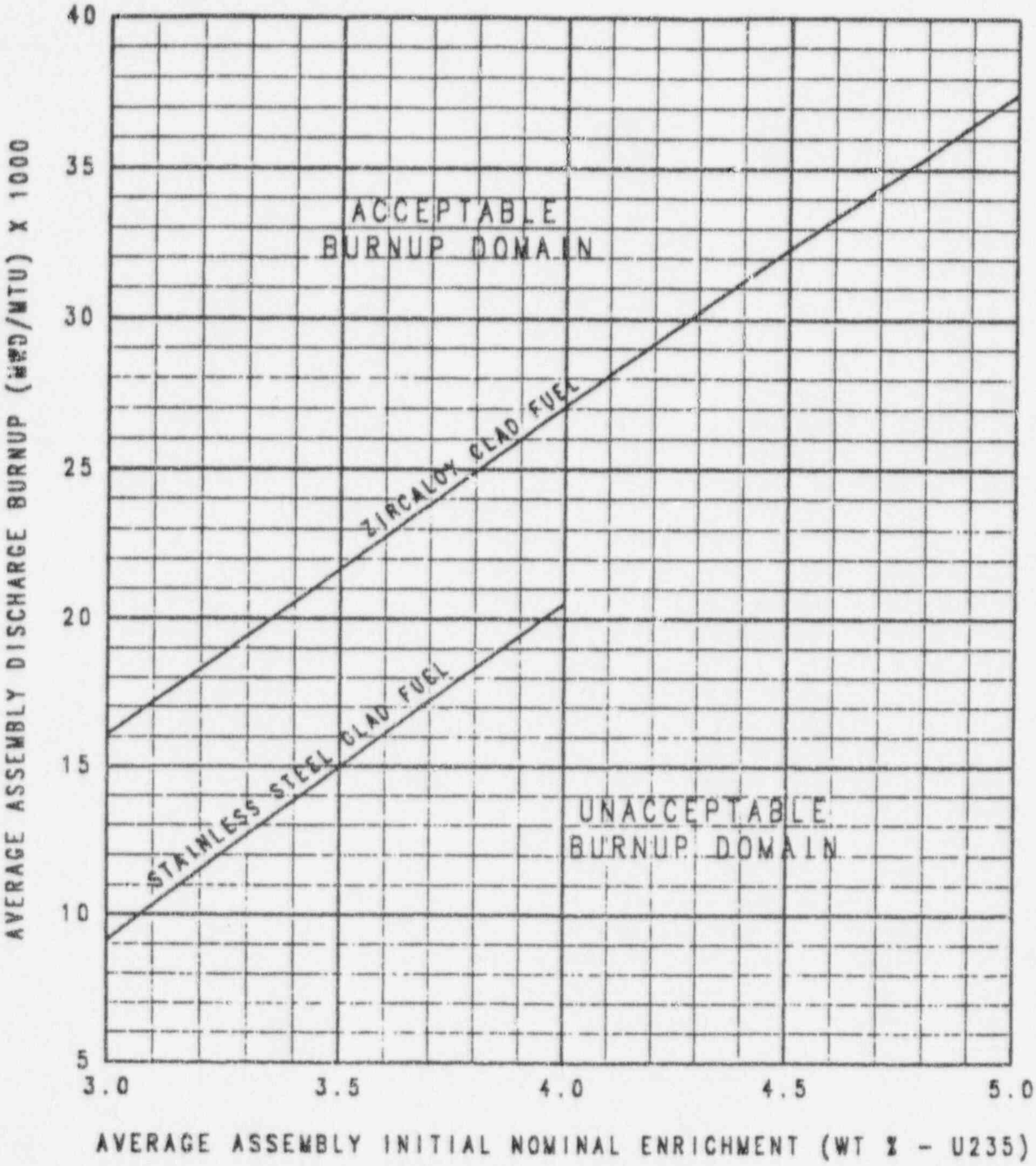

FIGURE 3.9-2 SPENT FUEL POOL RACK MINIMUM BURNUP REQUIREMENTS FOR REGION 2

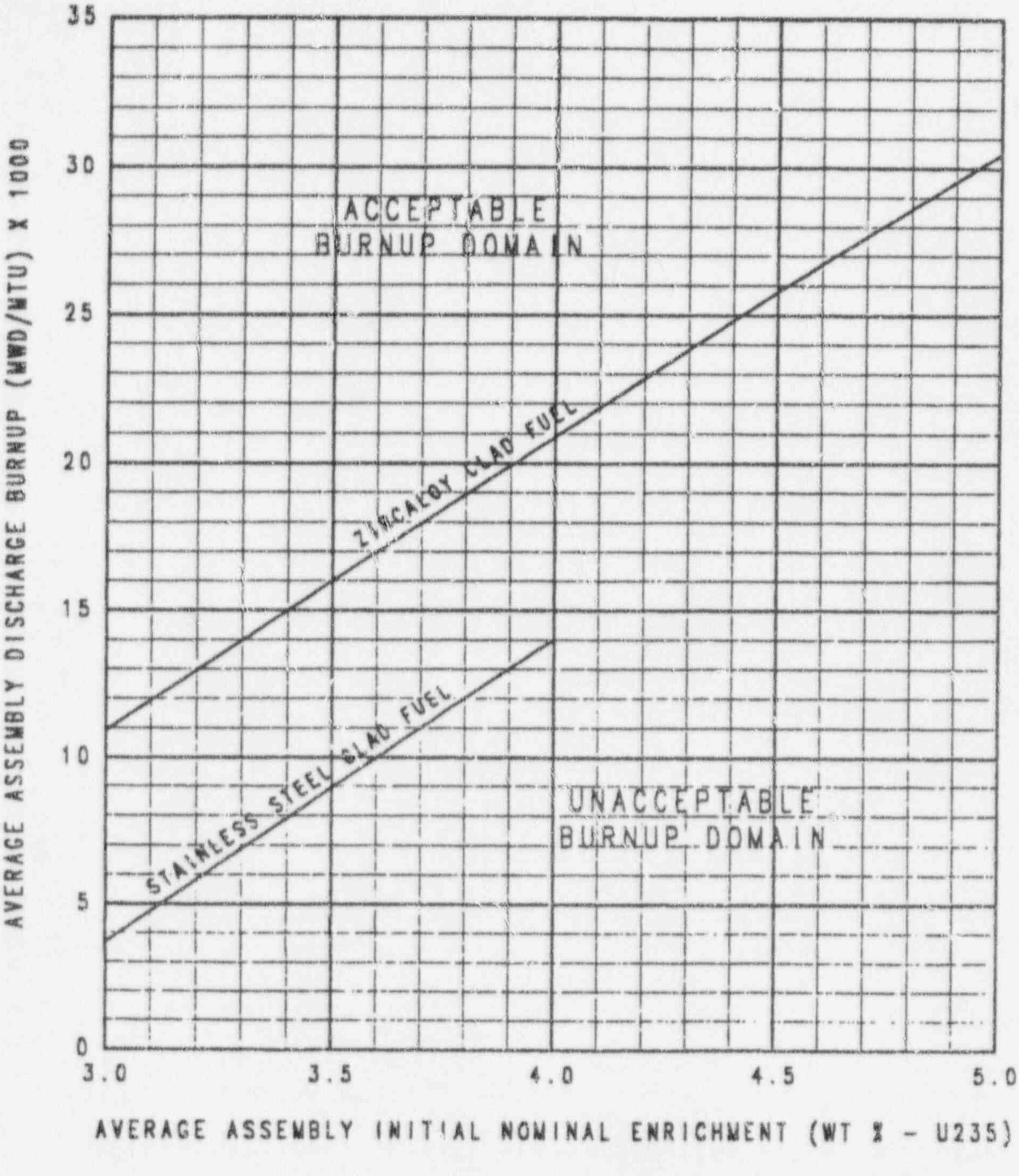

FIGURE 3.9-3 SPENT FUEL POOL RACK MINIMUM BURNUP REQUIREMENTS FOR REGION 3

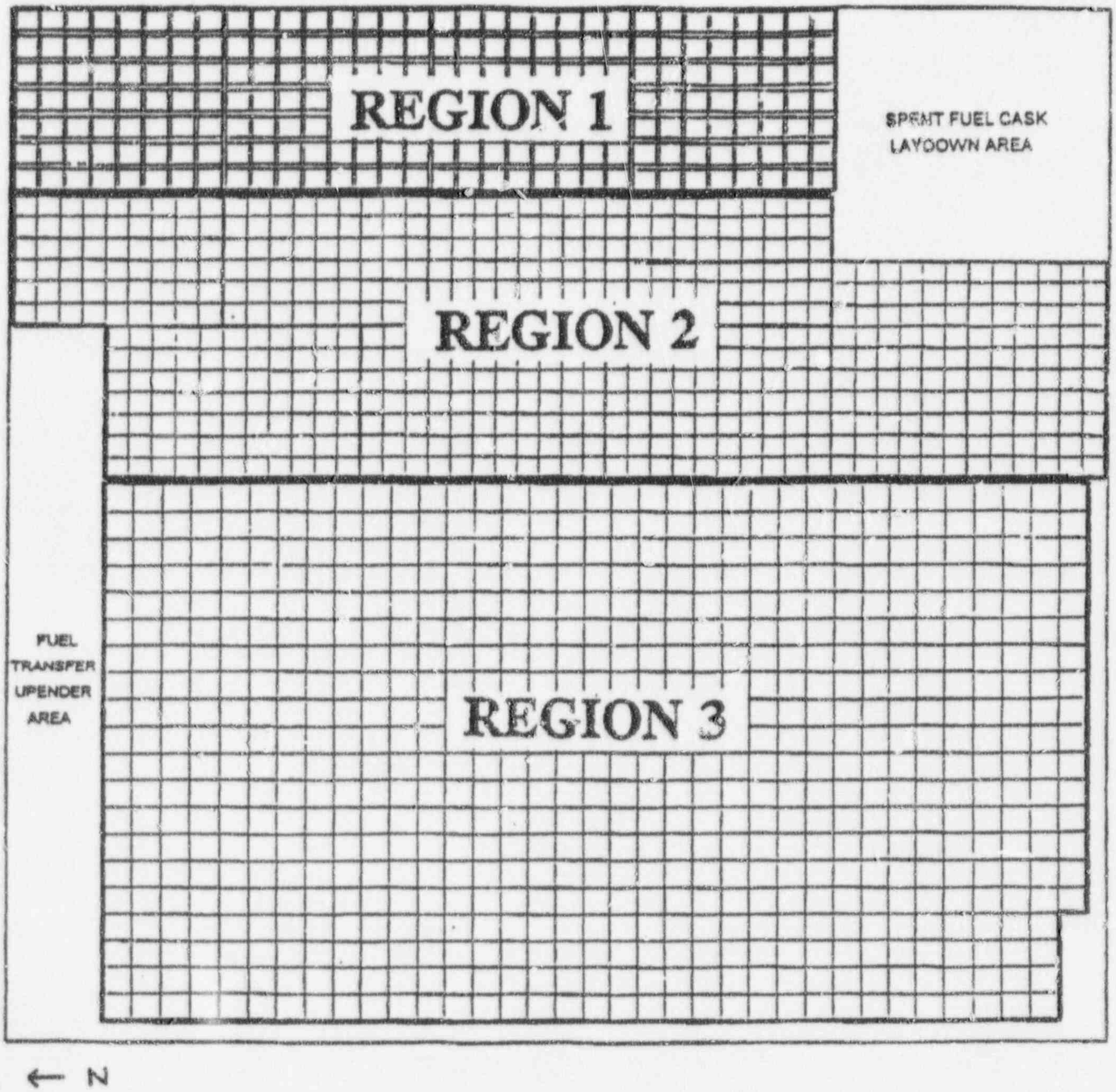

FIGURE 3.9-4 SPENT FUEL POOL RACK REGION LOCATIONS

3/4.9.15 SPENT FUEL POOL COOLING

LIMITING CONDITION FOR OPERATION

3.9.15 The spent fuel pool cooling system shall be OPERABLE with:

a. Both spent fuel pool cooling pumps OPERABLE and at least one spent fuel pool cooling pump and the plate heat exchanger in operation, and

b. Spent fuel pool temperature less than 150 degrees F.

APPLICABILITY: MODE 6, during transfer and storage of irradiated fuel from the reactor vessel to the spent fuel pool for a full core offload.*

ACTION: With less than the required equipment OPERABLE and in operation or with spent fuel pool temperature greater than 150 degrees F, suspend all operations involving the addition of irradiated fuel to the spent fuel pool and initiate corrective action to restore the spent fuel pool cooling system to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.9.15.1 Prior to movement of irradiated fuel to the spent fuel pool, verify that both spent fuel pool cooling pumps are lined up to provide flow to the plate heat exchanger.

4.9.15.2 At least once per 12 hours, verify that at least one spent fuel pool cooling pump is running.

4.9.15.3 At least once per 12 hours, verify that the spent fuel pool temperature is less than 150 degrees F.

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*Specification 3.9.15 does not apply if less than 53 fuel assemblies are transferred to the spent fuel pool.

SECTIONS 3.10 AND 4.10

SPECIAL TEST EXCEPTIONS

DELETED IN THEIR ENTIRETY

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-1) shall not exceed the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall not exceed 2×10^{-4} microCi/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, restore the concentration to within the above limits within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed in accordance with the sampling and analysis program specified in Section I of the REMODCM.

4.11.1.1.2 The results of the radioactive analysis shall be used in accordance with the methods of Section II of the REMODCM to assure that the concentration of the point of release are maintained within the limits of Specification 3.11.1.1.

DOSE, LIQUIDS

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to any MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the site (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose or dose commitment to any MEMBER OF THE PUBLIC from such release during the calendar year is within 3 mrem to the total body and 10 mrem to any organ.

b. DELETED

SURVEILLANCE REQUIREMENTS

4.11.1.2.1 Cumulative dose contributions from liquid effluents shall be determined in accordance with Section II of the REMODCM.

4.11.1.2.2 Relative accuracy or conservatisms of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in the REMODCM.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate, at any time, offsite (see Figure 5.1-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. The dose rate limit due to inhalation for tritium and for all radioactive materials in particulate form with half-lives greater than 8 days shall be less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, decrease the release rate within 15 minutes to comply with the limit(s) given in Specification 3.11.2.1.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The release rate, at any time, of noble gases in gaseous effluents shall be controlled by the offsite dose rate as established above in Specification 3.11.2.1. The corresponding release rate shall be determined in accordance with the methodology of Section II of the REMODCM.

4.11.2.1.2 The noble gas effluent monitors of Specification 3.3.3.8 shall be used to control release rates to limit offsite doses within the values established in Specification 3.11.2.1.

4.11.2.1.3 The release rate of radioactive materials in gaseous effluents shall be determined by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Section I of the REMODCM (Table D-1). The corresponding dose rate shall be determined using the methodology and parameters given in the ODCM (Section II of the REMODCM).

RADIOACTIVE EFFLUENTS

DOSE, NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose offsite (see Figure 5.1-1) due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year to less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose during the calendar year is within 10 mrad for gamma radiation and 20 mrad for beta radiation.

b. DELETED

SURVEILLANCE REQUIREMENTS

4.11.2.2.1 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with Section II of the REMODCM once every 31 days.

4.11.2.2.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in Section I of the REMODCM.

DOSE, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to any MEMBER OF THE PUBLIC tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents released offsite (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter to less than or equal to 7.5 mrem to any organ;

b. During any calendar year to less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radionuclides, radioactive materials in particulate form, or radionuclides other than noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to reduce the releases during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose or dose commitment to any MEMBER OF THE PUBLIC from such releases during the calendar year is within 15 mrem to any organ.

b. DELETED

SURVEILLANCE REQUIREMENTS

4.11.2.3.1 Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with Section II of the REMODCM once every 31 days.

4.11.2.3.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in the REMODCM.

RADIOACTIVE EFFLUENTS

3/4.11.3 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.3 The dose or dose commitment from the site to a MEMBER OF THE PUBLIC is limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) over a period of 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3, prepare and submit a Special Report to Commission pursuant to Specification 6.9.2 and limit the subsequent releases such that the dose or dose commitment from the site to any MEMBER OF THE PUBLIC is limited to less than or equal to 25 mrem to the total body or any organ (except thyroid, which is limited to less than or equal to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures from the site to any MEMBER OF THE PUBLIC (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. If the estimated doses exceed the above limits, the special report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. DELETED

SURVEILLANCE REQUIREMENTS

4.11.3 Cumulative dose contributions from liquid and gaseous effluents and direct radiation shall be determined in Specifications 4.11.1.2.1, 4.11.2.2.1 and 4.11.2.3.1 and in accordance with Section II of the REMODCM once per 31 days.

BASES FOR
SECTIONS 3.0 AND 4.0
LIMITING CONDITIONS FOR OPERATION
AND
SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

BASES

Specifications 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10CFR50.36(c)(2):

> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., specified applicable conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, it is required that the facility be placed in a specified applicable condition in which the specification no longer applies. It is not intended that the ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 DELETED

Specification 3.0.4 establishes limitations on changes from a specfied applicable condition when a Limiting Condition for Operation is not met. The purpose of this specification is to ensure that the facility is not changed from a specified applicable condition when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant. Therefore, in this case, entry into a specified applicable condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status.

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR50.36(c)(3):

BASES (Con't)

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the specified applicable conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a specified applicable condition for which the associated Limiting Conditions for Operation are applicable.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation.

If the allowable outage time limits of the ACTION requirements are less than 24 hours, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

<u>Specification 4.0.4</u> establishes the requirement that all applicable surveillances must be met before entry into a specified applicable condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a specified applicable condition for which these systems and components ensure safe operation of the facility.

BASES 4.0.5, 4.0.6;

REFERENCES 1, 2, 3; AND

PAGES B3/4 0-6 THROUGH B3/4 0-7

DELETED

BASES 3/4.1

REACTIVITY CONTROL SYSTEMS

DELETED IN THEIR ENTIRETY

BASES 3/4.2

POWER DISTRIBUTION LIMITS

DELETED IN THEIR ENTIRETY

BASES 3/4.3.1 THROUGH 3/4.3.3.2; AND

PAGES B 3/4 3-1 AND B 3/ 3-1a

DELETED

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

BASES 3/4.3.3.5 AND 3/4.3.3.6

DELETED

3/4.3.3.7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the REMODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

BASES 3/4.3.3.9 AND 3/4.3.4; AND

PAGE B 3/4 3-4

DELETED

BASES 3/4.4

REACTOR COOLANT SYSTEM

DELETED IN THEIR ENTIRETY

BASES 3/4.5

EMERGENCY CORE COOLING SYSTEMS

DELETED IN THEIR ENTIRETY

BASES 3/4.6

CONTAINMENT SYSTEMS

DELETED IN THEIR ENTIRETY

BASES 3/4.7.1 THROUGH 3/4.7.4; AND

PAGES B 3/4 7-1 THROUGH B 3/4 7-3a

DELETED

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

BASES 3/4.7.6 THROUGH 3/4.7.12; AND

PAGES B 3/4 7-5 THROUGH B 3/4 7-7

DELETED

BASES 3/4.8

ELECTRICAL POWER SYSTEMS

DELETED IN THEIR ENTIRETY

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. A value of 0.94 or less for K_{eff} is required for this accident. For a core configuration of all rods in an additional 0.05 K_{eff} penalty is required to account for a heavy load crushing the core into a more reactive configuration.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that monitoring capability is available to detect changes in the reactivity condition of the core. Redundant monitoring capability is required to detect changes in the reactivity condition of the core during fuel movement.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum requirement for reactor hold time prior to movement of irradiated fuel from the reactor vessel to the spent fuel pool ensures that sufficient decay time has elapsed for adequate spent fuel pool cooling should a failure occur to the cooling system. The reactor hold time is a function of cooling water temperature. The required decay time assumes a conservative maximum transfer rate of six assemblies per hour. These decay times are consistent with the assumptions used in the safety analysis.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling cavity manipulator crane area personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of control rod drive shafts and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive shaft or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting forces in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analysis.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one RHR LOOP be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR LOOPS OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR LOOP will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR LOOP, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE SUPPLY, PURGE EXHAUST, AND PURGE EXHAUST BYPASS ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations can be isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL STORAGE BUILDING AIR CLEANUP SYSTEM

The limitations on the Fuel Storage Building Air Cleanup System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analysis. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.9.13 MOVEMENT OF FUEL IN SPENT FUEL POOL

The limitations of this specification ensure that, in the event of any fuel handling accident in the spent fuel pool, K_{eff} will remain ≤ 0.95.

3/4.9.14 SPENT FUEL POOL - REACTIVITY CONDITION

The limitations described by Figures 3.9-2, 3.9-3, and Figure 3.9-4 ensure that the reactivity of fuel assemblies introduced into the spent fuel racks, with no credit taken for soluble boron in the pool, are conservatively within the assumptions of the safety analysis.

3/4.9.15 SPENT FUEL POOL COOLING

The limitations on the Spent Fuel Pool Cooling System ensure there is sufficient capacity to remove decay heat produced by the stored spent fuel elements with a full core offload and maintain the bulk pool temperature below 150°F with a maximum heat load of 22.4×10^6 BTU/hr. The maximum design heat load is based on an emergency full core offload scenario conservatively evaluated after the final operating cycle.

Requiring both spent fuel pool cooling pumps to be OPERABLE provides backup capability in the event the operating pump fails. In the event of a complete loss of forced cooling during a full core offload, the time to boil is greater than 7 hours in the most severe discharge scenario. This allows sufficient time to provide an alternate power source to the SFP pumps for an electrical failure or alternate SFP inventory makeup capability for a mechanical failure. In the event of a loss of offsite power and the A train diesel is out-of-service, there is sufficient time to repower the SFP cooling system from an alternate diesel generator. Therefore, operability of the spent fuel pool cooling system does not require the A train diesel generator to be available.

Should failure to restore operation of the cooling system occur before boiling takes place, cooling of the spent fuel pool can be accomplished by allowing the SFP to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

BASES 3/4.10

SPECIAL TEST EXCEPTIONS

DELETED IN THEIR ENTIRETY

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE, LIQUIDS

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". The dose calculation methodology and parameters in the REMODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the REMODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose rate at anytime from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for all areas offsite. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual offsite to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of that individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid or other organ dose rate above background to a child to less than or equal to 1500 mrem/year from inhalation.

3/4.11.2.2 DOSE, NOBLE GASES

This specification is provided to implement the requirements of Sections II.B., III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculational of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977.

The ODCM equations provided for determining the air doses at the site boundary are based upon utilizing successively more realistic dose calculational methodologies. More realistic dose calculational methods are used whenever simplified calculations indicate a dose approaching a substantial portion of the regulatory limits. The methods used, in order are, previously determined air dose per released activity ratio, historical meteorological data and actual radionuclide mix released, or real time meteorology and actual radionuclides released.

3/4.11.2.3 DOSE, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides for Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials will to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision I, July 1977. The release rate specifications for radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.3 TOTAL DOSE

This specification is provided to meet the reporting requirements of 40 CFR Part 190. For the purposes of the Special Report, it may be assumed that the dose commitment to any MEMBER OF THE PUBLIC from other fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 shall be 1740 feet.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

5.2 DELETED

5.3 DELETED

5.4 DELETED

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1-1.

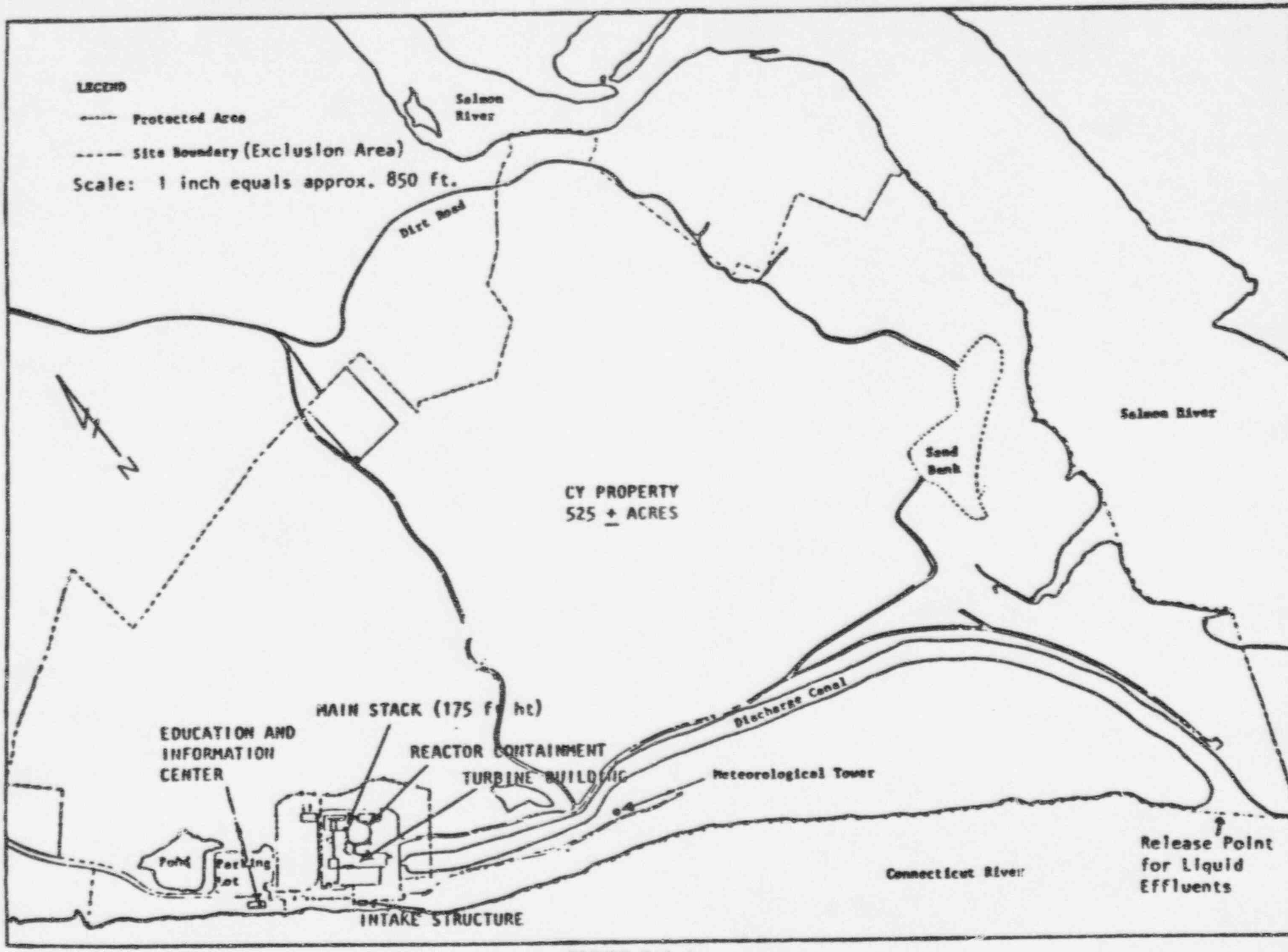

FIGURE 5.1-1
EXCLUSION AREA BOUNDRY AND SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

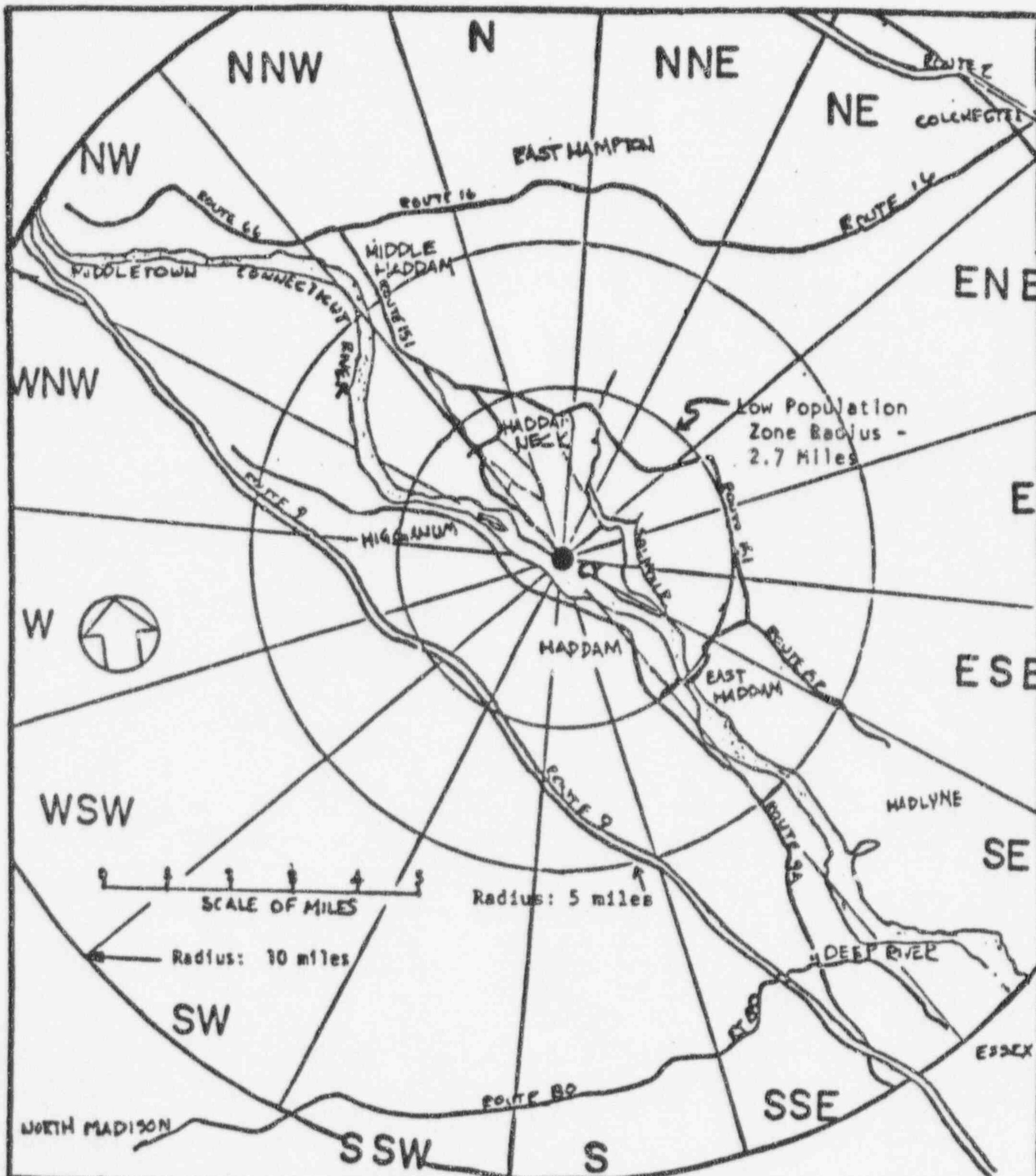

FIGURE 5.1-2

LOW POPULATION ZONE

5.6 SPENT FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are made up of 3 regions which are designed and shall be maintained with:

a. For Region 1, a nominal 10.98 inch (East/West) and a nominal 10.142 inch (North/South) center-to-center distance including neutron absorber surrounding each assembly to ensure a K_{eff} less than or equal to .95 when flooded with unborated water. Fuel with a maximum nominal enrichment of 5 weight percent U-235, without regard for fuel burnup, may be stored in this region. The location of Region 1 within the Spent Fuel Pool is shown in Figure 3.9-4.

b. For Region 2, a nominal 9.00 inch center-to-center distance including neutron absorber surrounding each assembly to ensure a K_{eff} less than or equal to .95 when flooded with unborated water. Storage of fuel in Region 2 must be in accordance with the requirements of Specification 3.9.14. The location of Region 2 within the Spent Fuel Pool is shown in Figure 3.9-4.

c. For Region 3, a nominal 10.75 inch center-to-center distance to ensure a K_{eff} less than or equal to .95 when flooded with unborated water. No credit is taken for the neutron absorber contained within the racks of this region. Storage of fuel in Region 3 must be in accordance with the requirements of Specification 3.9.14. The location of Region 3 within the Spent Fuel Pool is shown in Figure 3.9-4.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 27.0 feet MSL.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 224 storage locations in Region 1, 560 storage locations in Region 2, and 696 storage locations in Region 3 for a total of 1480 storage locations.

SECTION 5.7;
TABLE 5.7-1;
FIGURES 5.6-1 AND 5.6-2; AND
PAGES 5-5 THROUGH 5-8

DELETED

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The designated CYAPCO corporate officer shall be responsible for overall facility operation and shall delegate, in writing, the succession to this responsibility during his absence.

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 the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for 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 Connecticut Yankee Quality Assurance Program (CYQAP).

b. An individual management position in the onsite organization shall be responsible for overall unit safe operation and shall have control over those onsite activities and resources necessary for safe operation and maintenance of the plant.

c. The designated CYAPCO corporate 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 plant 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

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

b. At least one licensed Operator shall be in the control room when irradiated fuel is in the spent fuel pool;

c. An individual qualified in radiation protection procedures* shall be on site during fuel handling operations;

d. All fuel handling operations shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. DELETED

f. Administrative procedures shall be developed and implemented to limit the working hours of facility staff who perform safety-related functions. These procedures should follow the general guidance of the NRC Policy Statement on working hours (Generic Letter No. 82-12).

* The individual qualified in radiation protection procedures may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION
SOP	1
ROP	1
AOP	2

Abbreviations:

SOP - Licensed Senior Reactor Operator
ROP - Licensed Reactor Operator
AOP - Additional Operator

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 unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Manager from the control room, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

6.3.1.1 The position of Health Physics Manager shall meet the following minimum qualifications:

a. Academic degree in an engineering or science field or equivalent as per Section 6.3.1.1.c.

b. Minimum of five years professional technical experience in the area of radiological safety, three years of which shall be in applied radiation work in a nuclear facility dealing with problems similar to those encountered in a nuclear power reactor.

c. Technical experience in the area of radiological safety beyond the five year minimum may be substituted on a one-for-one basis towards the academic degree requirement (four years of technical experience being equivalent to a four year academic degree).

d. Academic and technical experience must total a minimum of nine years.

6.3.1.2 DELETED

6.4 TRAINING

A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Unit Director and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10CFR55.59. The Director-Nuclear Training has the overall responsibility for the implementation of the Training Program.

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 Unit Director on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of at least seven members. Members shall collectively have experience and expertise in the following areas:

Plant Operations,
Decommissioning,*
Engineering,
Maintenance,
Health Physics,
Chemistry/Radiochemistry,
Quality Assurance, and
Security.*

The minimum qualifications of PORC members shall be that all members have an academic degree in an engineering or physical science field, or hold a management position, and have a minimum of five years technical experience in their respective field of expertise. The members of PORC shall be appointed in writing by the Unit Director, who is the PORC Chairperson. The alternate Chairpersons of the PORC shall be drawn from the PORC members and be appointed in writing by the Unit Director.

* These areas are exempt from the 5-year experience requirement.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairperson 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 Chairperson or his/her designated alternate.

QUORUM

6.5.1.5 The quorum of the PORC shall consist of the Chairperson or his/her designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of: (1) all procedures required by Specification 6.8 and changes thereto, and 2) any other proposed procedures or changes thereto as determined by the Unit Director to affect nuclear safety; |

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to the Technical Specifications;

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the designated CYAPCO corporate officer and to the Nuclear Safety Assessment Board; |

f. Review of all REPORTABLE EVENTS;

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

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Nuclear Safety Assessment Board or the Unit Director. |

i. Not used.

j. Not used.

k. Review of the Fire Protection Program and Implementing Procedures.

AUTHORITY

6.5.1.7 The PORC shall:

a. Report to and be advisory to the Unit Director on those areas of responsibility specified in Section 6.5.1.6(a) through (k);

b. Render determinations in writing to the Unit Director if any item considered under Specification 6.5.1.6a. through d., above, as appropriate and as provided by 10CFR50.59 or 10CFR50.92 constitutes an unreviewed safety question or requires a significant hazards consideration determination.

c. Provide written notification, meeting minutes may be used for this purpose, to the designated CYAPCO corporate officer and the Chairperson of the Nuclear Safety Assessment Board of disagreement between the PORC and the Unit Director; however, the designated CYAPCO corporate officer shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these Technical Specifications. A Copy shall be provided to the Nuclear Safety Assessment Board.

6.5.2 NUCLEAR SAFETY ASSESSMENT BOARD (NSAB)

FUNCTION

6.5.2.1 The minimum qualifications of NSAB members are as follows:

a. The Chairperson and NSAB members shall have:

1. An academic degree in an engineering or physical science field, or hold a senior management position, and

2. A minimum of five years technical experience in their respective field of expertise.

b. The NSAB shall have experience in and shall function to provide independent oversight review and audit of designated activities in the areas of:

Plant Operations,
Decommissioning,*
Engineering,
Radiological Safety,
Chemistry/Radiochemistry
Quality Assurance, and
Environmental Protection.*

The NSAB serves to advise the designated CYAPCO corporate officer on matters related to nuclear safety and notify the designated CYAPCO corporate officer within 24 hours of a safety significant issue and/or a safety significant disagreement between the NSAB and the organization or function being reviewed.

COMPOSITION

6.5.2.2 The designated CYAPCO corporate officer shall appoint, in writing, a minimum of five members to the NSAB and shall designate from this membership, in writing, a Chairperson and a Vice Chairperson. The membership shall function to provide independent review and audit in the areas listed in Specification 6.5.2.1.

ALTERNATES

6.5.2.3 All alternate members shall be appointed, in writing, by the designated CYAPCO corporate officer; however, no more than two alternates shall participate as members in NSAB activities at any one time.

MEETING FREQUENCY

6.5.2.4 The NSAB shall meet at least once per calendar quarter.

QUORUM

6.5.2.5 The quorum of the NSAB shall consist of a majority of NSAB members including the Chairperson or Vice Chairperson. No more than a minority of the quorum shall have line responsibility for operation of the same Haddam Neck Plant. No more than two alternates shall be appointed as members at any meeting in fulfillment of the quorum requirements.

* These areas are exempt from the 5-year experience requirement.

REVIEW RESPONSIBILITIES

6.5.2.6 The NSAB shall be responsible for the review of:

a. The safety evaluations for changes to procedures, equipment, or systems, and tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question as defined in 10 CFR 50.59;

b. Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR 50.59;

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

e. Violations of applicable codes, regulations, orders, license requirements, or internal procedures having nuclear safety significance;

f. All Licensee Event Reports required by 10 CFR 50.73;

g. Indications of significant unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect nuclear safety;

h. Significant accidental, unplanned, or uncontrolled radioactive releases, including corrective actions to prevent recurrence;

i. Significant operating abnormalities or deviations from normal and expected performance of equipment that could affect nuclear safety;

j. The performance of the corrective action program; and

k. Audits and the Annual Audit Plan.

Reports or records of these reviews shall be forwarded to the designated CYAPCO corporate officer within 30 days following completion of the review.

AUDIT PROGRAM RESPONSIBILITIES

6.5.2.7 The NSAB audit program shall be the responsibility of the Nuclear Oversight organization. NSAB audits shall be performed at least once per 24 months in accordance with appropriate administrative procedures and shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The training and qualifications of the unit staff;

c. The implementation of all programs required by Specification 6.8;

d. The Fire Protection Program and implementing procedures;

e. The fire protection equipment and program implementation utilizing either a qualified offsite license fire protection engineer or an outside independent fire protection consultant.

f. Actions taken to correct deficiencies occurring in equipment, structures, systems, components, or method of operation that affect nuclear safety; and

g. Cther activities and documents as requested by the designated CYAPCO corporate officer.

RECORDS

6.5.2.8 Written records of reviews and audits shall be maintained. As a minimum, these records shall include results of the activities conducted under the provisions of Section 6.5.2.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

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

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of this review shall be submitted to the NS/B and the designated CYAPCO corporate officer.

6.7 DELETED

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures and/or administrative policies 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. The requirements and recommendations of Sections 5.1 and 5.3 of ANSI N 18.7-1976.

c. Fire Protection Program implementation.

d. Quality controls for effluent monitoring, using the guidance in Regulatory Guide 1.21 Rev. 1, June 1974.

e. RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM) implementation except for Section I.E, Radiological Environmental Monitoring.

f. Process Control Program implementation.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the PORC and shall be approved by the Unit Director prior to implementation and reviewed periodically as set forth in each document or in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 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 Operator license on the unit affected; and

c. The change is documented, reviewed by the PORC and approved by the Unit Director within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained covering Section I.E., Radiological Environmental Monitoring, of the REMODCM.

6.8.5 All procedures and procedure changes required for the Radiological Environmental Monitoring Program of Specification 6.8.4 above shall be reviewed by an individual (other than the author) from the Radiological Assessment Branch or the Production Operation Services Laboratory (POSL) and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the Radiological Assessment Branch or the POSL, within 14 days of implementation.

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, one copy to the appropriate Regional Office of the NRC, and one copy to the appropriate NRC Resident Inspector, unless otherwise noted.

6.9.1.1 DELETED

6.9.1.2 DELETED

6.9.1.3 DELETED

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the facility as described below for the previous calendar year shall be submitted prior to March 1 of each year.

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 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, thermoluminescent 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 shall be assigned to specific major work functions;

b. DELETED

c. DELETED

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine Annual Radiological Environmental Operating Reports covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include that information delineated in the REMODCM.

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* This tabulation supplements the requirements of Section 20.2206 of 10 CFR Part 20.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 A routine Annual Radioactive Effluent Report covering the operation of the facility during the previous calendar year of operation shall be submitted by May 1 of each year.

The report shall include that information delineated in the REMODCM.

Any changes to the REMODCM shall be submitted in the Annual Radioactive Effluent Release Report.

6.9.1.8 DELETED

6.9.1.9. DELETED

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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 one copy to the appropriate Regional Office of the NRC, and one copy to the appropriate NRC Resident Inspector, within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of facility operation;

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

c. All REPORTABLE EVENTS;

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

e. Records of tests and experiments;

f. Records of changes made to the 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; and

6.10 RECORD RETENTION (Cont)

6.10.3 The following records shall be retained for the duration of the facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the FSAR;

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

c. Records of facility radiation and contamination surveys;

d. Records of radiation exposure for all individuals entering radiation control areas;

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

f. Records of transient or operational cycles of the reactor vessel;

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

h. Records of inservice inspections performed under previous amendments to these Technical Specifications;

i. Records of quality assurance activities required by the Quality Assurance Manual not listed in Specification 6.10.2;

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

k. Records of meetings of the PORC and the NSAB;

l. Records for Environmental Qualification.

m. Records of reviews performed for changes made to the RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM) and the Process Control Program.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mR/h but equal to or less than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physics Manager in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device or continually guarded.

6.13 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

Section I, RADIOLOGICAL EFFLUENTS MONITORING MANUAL (REMM), shall outline the sampling and analysis programs to determine the concentration of radioactive materials released offsite as well as dose commitments to individuals in those exposure pathways and for those radionuclides released as a result of facility operation. It shall also specify operating guidelines for RADIOACTIVE WASTE TREATMENT SYSTEMS and report content.

Section II, the OFFSITE DOSE CALCULATION MANUAL (ODCM), shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculations of gaseous and liquid effluent monitoring instrumentation Alarm/Trip Setpoints consistent with the applicable LCO's contained in these Technical Specifications.

Changes to the REMODCM:

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

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

   2) 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 10CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by PORC and the approval of the Unit Director.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire REMM or ODCM, as appropriate, as a part of or concurrent with the Annual Radioactive Effluent Report for the period of the report in which any change 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.

6.14 RADIOACTIVE WASTE TREATMENT

Procedures for liquid and gaseous radioactive effluent discharges from the facility shall be prepared, approved, maintained and adhered to for all operations involving offsite releases of radioactive effluents. These procedures shall specify the use of appropriate RADIOACTIVE WASTE TREATMENT SYSTEMS utilizing the guidance provided in the REMODCM.

The Solid RADIOACTIVE WASTE TREATMENT SYSTEM shall be operated in accordance with the Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

6.15 DELETED

6.16 DELETED

Docket No. 50-213
CY-97-006

Attachment 3

Haddam Neck Plant

Proposed Revision To Operating License And Technical Specifications

Defueled Operating License And Technical Specifications

Description Of Changes

May 1997

General Reason For The Changes

In a letter dated December 5, 1996,[1] the Connecticut Yankee Atomic Power Company (CYAPCO) informed the NRC that the Board of Directors of CYAPCO had decided to permanently cease operations at the Haddam Neck Plant (HNP) and that the fuel had been permanently removed from the reactor.

The majority of the changes proposed, reflect the limitations and requirements appropriate to the present defueled configuration of the plant. The primary reason for this change is to simplify and to improve clarity by eliminating the large volume of non-applicable material in the current Operating License and Technical Specifications. The objective is a clear and concise document for maintaining the plant in a permanently defueled condition. When the above is the reason for a change, the phrase "Defueled Condition" will signify that the above rationale is applicable.

In addition, because this proposed amendment is affecting most pages in the Operating License and the Technical Specifications, some changes were made to correct typographical errors or grammar; or to provide consistent terminology, consistent format, clarity, or brevity; or to update the Index (Table of Contents) to reflect the proposed changes. When this is the reason for a change, the phrase "Editorial Changes" will signify that this rationale is applicable.

However, some of the changes discussed herein are proposed for other reasons. When this is the case, the reason will be provided.

FACILITY OPERATING LICENSE

1. The following are retained with noted changes:

   B.(3)

   Change "reactor startup" to "reactor startup (possession only)," and change "reactor instrumentation" to "reactor instrumentation (possession only)".

---

(1) T. C. Feigenbaum letter to the U. S. Nuclear Regulatory Commission, "Certifications Of Permanent Cessation Of Power Operation And That Fuel Has Been Permanently Removed From The Reactor," dated December 5, 1996.

C.(1) Maximum Power Level

Change "The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1825 megawatts (thermal)." to "The licensee is not authorized to operate the reactor. Fuel may not be placed in the reactor vessel."

C.(2) Technical Specifications

Change "Amendment No. 190," to "Amendment No. ,". (the blank to be filled in by the NRC prior to issuance)

C.(3) Deleted

Change "[(3) Deleted per Amdt. 29, 10-24-78.]" to "(3) Deleted by Amendment No. 29."

C.(4) Fire Protection

Replace the 2nd paragraph with: "The licensee may make changes to the fire protection program without NRC approval, if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment which could result in a radiological hazard, taking into account the decommissioning plant conditions and activities."

C.(5) Physical Protection
No change.

C.(6) Integrated Implementation Schedule
No change.

2. The following, Item C.(7), is added:

"(7) Fuel Movement

The movement of special nuclear material used as reactor fuel into the containment is prohibited."

Reasons

Defueled Condition and Editorial Changes.

Bases

1. For the proposed revisions to the license, the changes are discussed below:

   B.(3)

   The HNP may have such sources in it possession from previous operational cycles. The change reflects the permanent defueled plant condition by only allowing possession, not use of these sources.

   C.(1) Maximum Power Level

   Pursuant to 10CFR50.82(a)(1)(i) and 10CFR50.82(a)(1)(ii), CYAPCO has certified that the plant has permanently ceased operations and the reactor has been permanently defueled. The change maintains this present plant configuration by prohibiting the placement of fuel in the reactor vessel.

   C.(2) Technical Specifications

   Editorial change reflecting the new amendment.

   C.(3) Deleted

   Editorial change for consistency.

   C.(4) Fire Protection

   Because the plant has permanently ceased operations and the reactor has been permanently defueled, there no longer is the need to achieve and maintain safe shutdown of the reactor in the event of a fire. The change reflects that, in the event of a fire, the Fire Protection Program is now focused on potential radiological hazards, taking into account the decommissioning plant conditions and activities.

2. The plant has permanently ceased operations and the reactor has been permanently defueled. The new license condition maintains this present plant configuration by prohibiting the movement of fuel into the containment.

TECHNICAL SPECIFICATIONS

GENERAL

Changes

1. Where multiple pages in sequence have been deleted, a single page has been used as a replacement. This includes replacing pages that already denote deleted material (this portion of the replacement is editorial). These new single pages denote what section(s), table(s), figure(s) and/or page(s) have been deleted. In addition, minor typographical errors were corrected.

2. On pages that are unchanged but are being reissued, the revision bars are being removed.

Reason

Editorial Changes.

Basis

1. For clarity and brevity.

2. That this proposed amendment, when issued, will replace the entire Operating License and Technical Specifications to assure that pages that were supposed to be retained will not be inadvertently discarded. Since this will be a new amendment, the revision bars are being removed.

INDEX

Changes

1. Revise according to the changes described herein.

2. Retain numbering sequences.

Reason

Editorial Changes.

Basis

1. The Index must be revised to reflect the changes made to the Technical Specification Sections identified below. The following revisions were implemented:

   - Deleted sections are indicated in the Index by "DELETED".

   - If all the subsections of a Technical Specification are deleted, then only the major section is shown to be deleted. The deleted subsections are removed from the Index.

   - If a Technical Specification table or figure is deleted, then the table or figure is removed from the Index.

   - Tables and figures that are retained, continue to be listed in the Index.

2. Numbering sequences were retained to preclude the need to renumber cross-references and preclude the need to revise procedure references to the Technical Specifications.

1.0 DEFINITIONS

Changes

1. The following definitions have been retained:

1.1	ACTION
1.2	ANALOG CHANNEL OPERATIONAL TEST
1.4	CHANNEL CALIBRATION
1.5	CHANNEL CHECK
1.8	CORE ALTERATION
1.12	FREQUENCY NOTATION (including Table 1.1)
1.15	MEMBER(S) OF THE PUBLIC
1.17	OPERABLE - OPERABILITY
1.18	OPERATIONAL MODE - MODES (including Table 1.2)
1.23	RADIOACTIVE WASTE TREATMENTS SYSTEMS
1.24	RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)
1.26	REPORTABLE EVENT
1.27	RHR LOOP
1.29	SITE BOUNDARY
1.30	SOURCE CHECK

2. The following definitions have been deleted:

1.3	AXIAL OFFSET
1.6	CONTAINMENT INTEGRITY
1.7	CONTROLLED LEAKAGE
1.9	DOSE EQUIVALENT I-131
1.10	Ē - AVERAGE DISINTEGRATION ENERGY
1.11	END-OF-CORE LIFE
1.13	IDENTIFIED LEAKAGE
1.14	LINEAR HEAT GENERATION RATE
1.16	NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR
1.19	PHYSICS TESTS
1.20	PRESSURE BOUNDARY LEAKAGE
1.21	PURGE-PURGING
1.22	QUADRANT POWER TILT RATIO
1.25	RATED THERMAL POWER
1.28	SHUTDOWN MARGIN
1.31	STAGGERED TEST BASIS
1.32	THERMAL POWER
1.33	TRIP ACTUATING DEVICE OPERATIONAL TEST
1.34	UNIDENTIFIED LEAKAGE
1.35	VENTING
1.36	TECHNICAL REPORT SUPPORTING CYCLE OPERATION
1.37	REACTOR TRIP SYSTEM RESPONSE TIME

3. In Definition 1.23, change "RADIOACTIVE WASTE TREATMENTS SYSTEMS" to "RADIOACTIVE WASTE TREATMENT SYSTEMS".

4 In Definition 1.24, add "(REMM)" after "RADIOLOGICAL EFFLUENT MONITORING MANUAL".

5. In Table 1.1, delete the notation "S/U" and delete its frequency "Prior to each reactor criticality."

6. In Table 1.2, delete the information in MODES 1 through 5 and insert "DELETED" after each number for MODES 1 through 5.

<u>Reason</u>

Defueled Condition and Editorial Changes.

Basis

1. The definitions retained are referenced by the remaining Technical Specifications.

2. The definitions deleted are not referenced by the remaining Technical Specifications.

3. The change is editorial and is provided for consistency.

4. The change is editorial and is provided for consistency.

5. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this frequency notation is no longer required.

6. Because the plant has permanently ceased operations and the reactor has been permanently defueled, MODES 1 through 5 are no longer required.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Changes

1. Delete the entire section, including bases.

Reason

Defueled Condition.

Basis

1. Section 2.0 deals with the reactor core, reactor coolant system pressure and reactor system instrumentation setpoints. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this section is no longer required.

3/4.0 APPLICABILITY

Changes

1. The following were retained with noted changes:

3.0.1 Change "OPERATIONAL MODES or other conditions specified within;" to "specified applicable conditions;".

3.0.2 On the last line, change "Action" to "ACTION".

3.0.4 Delete the 1st sentence. In the 2nd sentence change "an OPERATIONAL MODE or specified condition" to "a specified applicable condition". Delete the 3rd sentence.

4.0.1 Change "OPERATIONAL MODES or other conditions specified" to "specified applicable conditions". Move to Page 3/4 0-2.

4.0.2 No change. Move to Page 3/4 0-2.

4.0.3 No change.

4.0.4 Change "an OPERATIONAL MODE or other specified condition" to "a specified applicable condition".

B3.0.1/B3.0.4 No change.

B3.0.1 In the 1st paragraph, 1st sentence change "(i.e., in which OPERATIONAL MODES or other specified conditions)" to "(i.e., specified applicable conditions)". In the 2nd paragraph, 5th sentence change "a shutdown is required to place the facility in a MODE or condition" to "it is required that the facility be placed in a specified applicable condition". In the 2nd paragraph, 6th sentence delete "shutdown". In the 3rd paragraph, delete the last sentence.

B3.0.2 No change.

B3.0.4 In the 1st paragraph:

- 1st sentence, change "on MODE changes when" to "on changes from a specified applicable condition when",
- delete the 2nd sentence,
- 3rd sentence, change "ensure that facility operation is not initiated or that higher MODES of operation are not entered when" to "ensure that the facility is not changed from a specified applicable condition when",
- 4th sentence delete "before or after a MODE change",
- 5th sentence change "into an OPERATIONAL MODE or other specified condition may" to "into a specified applicable condition may", and
- 6th sentence delete "before plant startup".

Delete the 2nd paragraph.

B4.0.1/B4.0.5 In the 1st sentence, change "<u>Specifications 4.0.1 through 4.0.5</u>" to "<u>Specifications 4.0.1 through 4.0.4</u>".

B4.0.1 In the 1st sentence, change "the OPERATIONAL MODES or other conditions" to "the specified applicable conditions". In the 2nd sentence, change "a MODE or other specified condition" to "a specified applicable condition". Delete the 3rd and 4th sentences.

B4.0.2 Delete the 3rd and 4th sentences.

B4.0.3 In the 2nd paragraph, 1st sentence, change "less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance" to "less than 24 hours, a 24-hour allowance". In the 2nd paragraph, 3rd sentence, delete "a shutdown is required to comply with ACTION requirements or before other".

B4.0.4 In the 1st paragraph:

- 1st sentence, change "an OPERATIONAL MODE or other specified condition" to "a specified applicable condition",

- 2nd sentence, change "into a MODE or condition for" to "into a specified applicable condition for", and

- delete 3rd sentence.

Delete the 2nd and 3rd paragraphs.

2. The following were deleted, including associated tables, figures and bases:

3.0.3

4.0.5

4.0.6

References (Page B3/4 0-7)

Reason

Defueled Condition and Editorial Changes.

Basis

1. Deleting references to OPERATING MODES, refueling, plant startup and plant shutdown is consistent with the plant having permanently ceased operations and the reactor being permanently defueled. Substituting the words "specified applicable condition" for "OPERATIONAL MODES or other conditions specified" retains the intent of the applicability for the conditions specified in each Limiting Condition for Operation. The change to 3.0.2 is editorial. The movement of Specifications 4.0.1 and 4.0.2 to Page 3/4 0-2 is editorial and is being done for clarification, since these two sections should be on a page labeled "Surveillance Requirements." The change to B4.0.1/B4.0.5 reflects the deletion of B4.0.5.

2. These subsections are associated with fuel in the reactor vessel and/or power operation of the facility. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these subsections are no longer required. Specifically:

3.0.3

The deleted specification has requirements specific to operating modes. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this specification is no longer required.

4.0.5

Because the plant has permanently ceased operations and the reactor has been permanently defueled, the remaining functional systems covered by this Specification are the service water system (SWS) and the spent fuel cooling system (SFCS). The SWS design pressure is 110 psig and the design temperature is 120 °F. The SFCS design pressure is 200 psig and the design temperature is 150 °F. These systems, while not designed to ASME III, are classified as ASME Section III, Code Class 3 (Reference: UFSAR Table 3.2-1).

This specification addresses the ASME Section XI inspection, examination and testing requirements for Class 1, 2 and 3 components. The inspections, examinations and test verify the integrity of high pressure and temperature systems. The remaining functional systems at the HNP are relatively low pressure and temperature systems. Therefore, high temperature/pressure erosion and corrosion is not expected. In ASME Section XI, Table IWD-2500-1 ("Test and Examination Categories"), 1983 Edition with Summer 83 Addenda, the SWS examination requirements are delineated in Items D2.10 through D2.60 and the SFCS examination requirements are delineated in Items D3.10 through D3.60. In both cases the examinations consist of a system pressure test with a visual inspection for leakage (Item D2.10 and D3.10) and a visual inspection of supports (Items D2.20 through D2.60 and Items D3.20 through D3.60). There are no requirements for surface or volumetric examinations. Thus, while ASME Section XI requires an additional recordkeeping workload, from a technical standpoint the code requirements are being met daily by Operations and other plant personnel. Modified ISI and IST requirements customized to the defueled plant condition will be incorporated in the Technical Requirements Manual to assure component and system readiness in performing their required functions.

Furthermore, 10CFR50.55a(g), which requires an examination and testing program in accordance with ASME Section XI, is predicated on power operations. Thus, following the submittal of the 10CFR50.82 certification, ASME Section XI is no longer applicable to the HNP.

Therefore, based on the above, this specification is no longer needed.

4.0.6

This specification is concerned with maintaining integrity of auxiliary feedwater system piping that is considered high energy piping (i.e., operating pressure ≥ 275 psig and/or operating temperature ≥ 200 °F). Because the plant has permanently ceased operations and the reactor has been permanently defueled, this auxiliary feedwater system piping will no longer see these operating pressures and/or temperatures. Therefore, this specification is no longer required.

References (Page B3/4 0-7)

Because the plant has permanently ceased operations and the reactor has been permanently defueled, these references which deal with EQ and HELB are no longer applicable and are no longer required.

3/4.1 REACTIVITY CONTROL SYSTEMS

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. Subsection 3/4.1 deals with limiting the reactivity condition of the reactor core. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this subsection is no longer required.

3/4.2 POWER DISTRIBUTION LIMITS

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. The Subsection 3/4.2 deals with limiting the power distribution in the reactor to ensure fuel integrity is maintained during operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this subsection is no longer required.

3/4.3 INSTRUMENTATION

Changes

1. The following are retained with noted changes:

   3/4.3.3.3 SEISMIC INSTRUMENTATION
   Delete ACTION "b."

   TABLE 3.3-5 SEISMIC MONITORING INSTRUMENTATION
   Change "N/A" to "N.A."

   TABLE 4.3-4 SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
   No change.

   3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION
   Delete ACTION "b."

   TABLE 3.3-6 METEOROLOGICAL MONITORING INSTRUMENTATION
   No change.

TABLE 4.3-5 METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

No change.

3/4.3.3.7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Delete ACTION "c." In footnote, change "outages are permitted" to "outages of monitoring channels are permitted".

TABLE 3.3-9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Delete Items 1.b and 3.b, ACTION 47, and associated notes.

TABLE 4.3-7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

In Item 3.B, change "B." to "b." Delete Items 1.b and 3.b, NOTATION (5), and the associated notes. Delete "and steam generator blowdown line" in the note associated with NOTATION (3)1.

3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

In 1 place in lead paragraph and in 2 places in ACTION "a" change "Alarm/Trip" to "Alarm". Delete ACTION "c." In footnote, change "outages are permitted" to "outages of monitoring channels are permitted".

TABLE 3.3-10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

In Item 1.a, delete "and Automatic Termination of Waste Gas System Releases". Delete Item 1.b.

TABLE 4.3-8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Delete Item 1.b. In NOTATION (3)a, change "Alarm/Trip" to "Alarm". Delete note associated with NOTATION (3)a.

B3/4.3.3.3 SEISMIC INSTRUMENTATION

No change.

B3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION
No change.

B3/4.3.3.7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
No change.

B3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
No change.

2. The following were deleted, including associated tables and bases:

3/4.3.1	REACTOR TRIP SYSTEM INSTRUMENTATION
3/4.3.2	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
3/4.3.3.1	RADIATION MONITORING FOR PLANT OPERATIONS
3/4.3.3.2	MOVABLE INCORE DETECTORS
3/4.3.3.5	ACCIDENT MONITORING INSTRUMENTATION
3/4.3.3.6	DELETED
3/4.3.3.9	BORON DILUTION ALARM
3/4.3.4	INTERNAL FLOOD PROTECTION

<u>Reason</u>

Defueled Condition and Editorial Changes.

<u>Basis</u>

1. For these subsections, the changes are discussed below:

3/4.3.3.3 Seismic Instrumentation

ACTION "b." refers to deleted Specification 3.0.3.

TABLE 3.3-5 Seismic Monitoring Instrumentation

Editorial change for consistency.

3/4.3.3.4 Meteorological Instrumentation

ACTION "b." refers to deleted Specification 3.0.3.

3/4.3.3.7 Radioactive Liquid Effluent Monitoring Instrumentation

ACTION "c." refers to deleted Specification 3.0.3. The change to the footnote clarifies that "outages" refers to the monitoring channels, not plant outages.

TABLE 3.3-9 Radioactive Liquid Effluent Monitoring Instrumentation

These changes refer to steam generator blowdown. Because the plant has permanently ceased operations and the reactor has been permanently defueled, it is no longer necessary to monitor steam generator blowdown.

TABLE 4.3-7 Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

The first change is an editorial change for consistency. The second set of changes refer to steam generator blowdown. Because the plant has permanently ceased operations and the reactor has been permanently defueled, it is no longer necessary to monitor steam generator blowdown.

3/4.3.3.8 Radioactive Gaseous Effluent Monitoring Instrumentation

ACTION "c." refers to deleted Specification 3.0.3. The change to the footnote clarifies that "outages" refers to the monitoring channels, not plant outages.

With respect to the change of "Alarm/Trip" to "Alarm", the automatic isolation of waste gas system releases is no longer necessary. The purpose of the waste gas system is to collect and store for decay the radioactive gasses generated in the RCS during reactor operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this system is no longer operated and is no longer required.

TABLE 3.3-10 Radioactive Gaseous Effluent Monitoring Instrumentation

With respect to Item 1.a, the deletion refers to the automatic termination of waste gas system releases. The purpose of the waste gas system is to collect and store for decay the radioactive gasses generated in the RCS during reactor operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this system is no longer operated and is no longer required.

With respect to the Iodine Sampler in Item 1.b, this instrument is not needed for the following reasons. The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

TABLE 4.3-8 Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

With respect to the Iodine Sampler in Item 1.b, this instrument is not needed for the following reasons. The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

With respect to the change to NOTATION (3)a and the note associated with NOTATION (3)a, the automatic isolation of waste gas system releases is no longer necessary. The purpose of the waste gas system is to collect and store for decay the radioactive gasses generated in the RCS during reactor operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this system is no longer operated and is no longer required.

2. These subsections are associated with fuel in the reactor vessel and/or power operation of the facility. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these subsections are no longer required. Specifically:

3/4.3.1 Reactor Trip System Instrumentation

This specification ensures operability of the Reactor Trip System. Because the plant has permanently ceased operations and the reactor has been permanently defueled, reactor trip system is no longer needed.

3/4.3.2 Engineered Safety Feature Actuation System Instrumentation

This specification ensures operability of the Engineered Safety Feature Actuation System. Because the plant has permanently ceased operations and the reactor has been permanently defueled, the accidents for which the engineered safety features (ESFs) would be required to mitigate are no longer possible. Therefore, the ESF actuation system is no longer needed.

3/4.3.3.1 Radiation Monitoring for Plant Operations

The purpose of this specification was to ensure reactor coolant system (RCS) leakage detection inside containment. Because the plant has permanently ceased operations and the reactor has been permanently defueled, RCS leakage detection is no longer needed.

3/4.3.3.2 Movable Incore Detectors

The purpose of this specification was to ensure accurate neutron flux readings were obtained. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these detectors are no longer needed.

3/4.3.3.5 Accident Monitoring Instrumentation

The purpose of this specification was to monitor accidents affecting the reactor coolant system (RCS) and/or containment. Because the plant has permanently ceased operations and the reactor has been permanently defueled, the accidents that these instruments would monitor are no longer possible. Therefore, this instrumentation is no longer needed.

3/4.3.3.6 Deleted

This specification had been previously deleted in Amendment No. 179.

3/4.3.3.9 Boron Dilution Alarm

The purpose of this specification was to assure that boron levels inside the reactor core precluded criticality. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these alarms are no longer needed.

3/4.3.4 Internal Flood Protection

The purpose of this subsection was to ensure protection against flooding of safety-related equipment that is associated with power operation of the reactor coolant system (RCS) or RCS/containment accident mitigation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, and fuel is no longer allowed in the containment; this subsection is no longer required.

3/4.4 REACTOR COOLANT SYSTEMS

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. The purpose of Subsection 3/4.4 was to ensure that fuel in the reactor vessel and/or power operation of the facility was maintained and/or operated in a safe manner. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this subsection is no longer required.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. The purpose of Subsection 3/4.5 was to ensure operability of the emergency core cooling system (ECCS). Because the plant has permanently ceased operations and the reactor has been permanently defueled, the accidents the ECCS would mitigate are no longer possible, thus this subsection is no longer required.

3/4.6 CONTAINMENT SYSTEM

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. The purpose of Subsection 3/4.6 was to ensure containment of radioactive releases due to design basis accidents with fuel in the reactor core or during in-containment movement of spent fuel. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these in-containment design basis accidents are no longer possible, this subsection is no longer required.

3/4.7 PLANT SYSTEMS

Changes

1. The following are retained with noted changes:

   3/4.7.5 SEALED SOURCE CONTAMINATION
   Delete ACTION "b." In Surveillance 4.7.5.2.c, delete "within 31 days prior to being subjected to core flux or installed in the core".

   B3/4.7.5 SEALED SOURCE CONTAMINATION
   No change.

2. The following were deleted, including associated tables and bases:

   3/4.7.1 TURBINE CYCLE

   3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

   3/4.7.3 SERVICE WATER SYSTEM

   3/4.7.4 SNUBBERS

   3/4.7.6 DELETED

3/4.7.7 DELETED

3/4.7.8 DELETED

3/4.7.9 FEEDWATER ISOLATION VALVES

3/4.7.10 EXTERNAL FLOOD PROTECTION

3/4.7.11 PRIMARY AUXILIARY BUILDING AIR CLEANUP SYSTEM

3/4.7.12 ULTIMATE HEAT SINK

Reason

Defueled Condition.

Basis

1. For these subsections, the changes are discussed below:

3/4.7.5 Sealed Source Contamination

ACTION "b." refers to deleted Specification 3.0.3. The deleted portion of Surveillance 4.7.5.2.c is no longer needed because it related specifically to power operations and fuel in the reactor. Because the plant has permanently ceased operations and the reactor has been permanently defueled, the deleted portion of Surveillance 4.7.5.2.c is no longer needed.

2. These subsections are associated with fuel in the reactor vessel and/or power operation of the facility. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these subsections are no longer required. Specifically:

3/4.7.1 Turbine Cycle

The purpose of this specification was to ensure that the effects of in-containment accidents (i.e., LOCA, MSLB, MFWLB) did not exceed containment design limits or off-site dose limits. Because the plant has permanently ceased operations and the reactor has been permanently defueled, accidents affecting the steam system are no longer possible and the specification is no longer needed.

3/4.7.2 Steam Generator Pressure/Temperature Limitation

The purpose of this specification was to ensure that pressure-induced stresses within the steam generator did not exceed established limits. Because the plant has permanently ceased operations and the reactor has been permanently defueled, there is no longer a need to ensure steam generator integrity. Therefore, the limits are no longer needed.

3/4.7.3 Service Water System

The purpose of this specification (which was applicable only in MODES 1 through 4) was to ensure that sufficient cooling water was provided to safety-related systems during normal operation and accident conditions. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this specification is no longer required.

With respect to spent fuel pool cooling in the defueled condition, the plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150 °F. The initial pool temperature of 150 °F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150 °F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide an alternate forced cooling or makeup to the spent fuel pool for a service water system failure. Therefore, operability of spent fuel pool cooling does not require the service water system to be immediately available.

Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 ½ hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

3/4.7.4 Snubbers

The purpose of this specification was to ensure that the structural integrity of applicable safety-related systems is maintained following a seismic or other event initiating dynamic loads. Snubbers are located on the following safety-related systems: RCS, CVCS, RHR system and feedwater system (safety-related portion). None of these safety-related systems are required to be operable because the plant has permanently ceased operations and the reactor has been permanently defueled. Therefore, this specification is no longer needed.

3/4.7.6 Deleted

This specification had been previously deleted in Amendment No. 179.

3/4.7.7 Deleted

This specification had been previously deleted in Amendment No. 179.

3/4.7.8 Deleted

This specification had been previously deleted in Amendment No. 179.

3/4.7.9 Feedwater Isolation Valves

The purpose of this specification was to ensure that the effects of in-containment MFWLB accidents did not exceed containment design limits or off-site dose limits. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these accidents are no longer possible and the specification is no longer needed.

3/4.7.10 External Flood Protection

The purpose of this specification was to ensure that initiated certain actions to protect safety-related equipment (i.e., safety-related equipment in the PAB, the service water system and the EDG system) from flooding by the Connecticut River. Because the plant has permanently ceased operations and the reactor has been permanently defueled, the safety-related equipment that was protected by this specification is no longer required. Therefore, this specification is no longer needed.

3/4.7.11 Primary Auxiliary Building Air Cleanup System

The purpose of this specification was to ensure that the effects of in-containment accidents (i.e., LOCA, MSLB, MFWLB, fuel handling accident) did not exceed off-site dose limits. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these accidents are no longer possible and the specification is no longer needed.

3/4.7.12 Ultimate Heat Sink

The purpose of this specification (which was applicable only in MODES 1 through 4) was to ensure that sufficiently cold water (i.e., $\leq$ 90 °F) was provided to safety-related systems during normal operation and accident conditions. The ACTION statement required shutdown of the plant. Because the plant has permanently ceased operations and the reactor has been permanently defueled, this specification is no longer required.

3/4.8 ELECTRICAL POWER SYSTEMS

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. The purpose of Subsection 3/4.8 was to ensure that sufficient power was available to supply the safety-related equipment required for: (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The first item is no longer applicable because the plant has permanently ceased operations and the reactor has been permanently defueled. With respect to the second item, the defueled condition leaves only a loss of spent fuel pool forced cooling and the off-site dose consequences of a fuel handling accident. As discussed below, this specification is not needed to mitigate either of these postulated accidents.

   The plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150 °F. The initial pool temperature of 150 °F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150 °F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide an alternate forced cooling or makeup to the spent fuel pool for a service water system failure. Therefore, operability of spent fuel pool cooling does not require a power source to be immediately available.

   Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 ½ hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

3/4.9 REFUELING OPERATIONS

No change, retain the entire subsection, including associated tables, figures and bases.

3/4.10 SPECIAL TEST EXCEPTIONS

Changes

1. Delete the entire subsection, including associated tables, figures and bases.

Reason

Defueled Condition.

Basis

1. Subsection 3/4.10 imposes limits that restrict plant operation during certain tests. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these limits on plant operation are no longer necessary and this subsection is no longer required.

3/4.11 RADIOACTIVE EFFLUENTS

Changes

1. The following are retained with noted changes:

3/4.11.1.1 LIQUID EFFLUENTS - CONCENTRATION
No change.

3/4.11.1.2 LIQUID EFFLUENTS - DOSE, LIQUIDS
Delete ACTION "b."

3/4.11.2.1 GASEOUS EFFLUENTS - DOSE RATE
In Subsection 3.11.2.1.b, delete "Iodine - 131, Iodine - 133".

3/4.11.2.2 GASEOUS EFFLUENTS - DOSE-NOBLE GASES
In title, change "DOSE-NOBLE GASES" to "DOSE, NOBLE GASES". Delete ACTION "b."

3/4.11.2.3 GASEOUS EFFLUENTS - DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES
In the title, delete "RADIOIODINES,". In Subsection 3.11.2.3, delete "Iodine - 131, Iodine - 133". Delete ACTION "b."

3/4.11.3 TOTAL DOSE
Delete ACTION "b."

B3/4.11.1.1 LIQUID EFFLUENTS - CONCENTRATION
No change.

B3/4.11.1.2 LIQUID EFFLUENTS - DOSE, LIQUIDS
No change.

B3/4.11.2.1 GASEOUS EFFLUENTS - DOSE RATE
No change.

B3/4.11.2.2 GASEOUS EFFLUENTS - DOSE, NOBLE GASES
No change.

B3/4.11.2.3 GASEOUS EFFLUENTS - DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES
In the title, delete "RADIOIODINES,". In the 6th sentence, delete "radioiodines,".

B3/4.11.3 TOTAL DOSE
No change.

Reason

Defueled Condition and Editorial Changes.

Basis

1. For these subsections, the changes are discussed below:

3/4.11.1.2 Liquid Effluents - Dose, Liquids

ACTION "b." refers to deleted Specification 3.0.3.

3/4.11.2.1 Gaseous Effluents - Dose Rate

The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

3/4.11.2.1 Gaseous Effluents - Dose-Noble Gases
(now titled: Gaseous Effluents - Dose, Noble Gases)

Title change is editorial and is done for consistency. ACTION "b." refers to deleted Specification 3.0.3.

3/4.11.2.3 Gaseous Effluents - Dose, Radioiodines, Radioactive Material in Particulate Form and Radionuclides Other Than Noble Gases
(now titled: Gaseous Effluents - Dose, Radioactive Material in Particulate Form and Radionuclides Other Than Noble Gases)

The first two items deal with radioiodines. The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

ACTION "b.", in the last item, refers to deleted Specification 3.0.3.

3/4.11.3 Total Dose

ACTION "b." refers to deleted Specification 3.0.3.

B3/4.11.2.3 Gaseous Effluents - Dose, Radioiodines, Radioactive Material in Particulate Form and Radionuclides Other Than Noble Gases
(now titled: Gaseous Effluents - Dose, Radioactive Material in Particulate Form and Radionuclides Other Than Noble Gases)

The two deletions deal with radioiodines. The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

5.0 DESIGN FEATURES

Changes

1. The following are retained with noted changes:

5.1 SITE
No change.

FIGURE 5.1-1 EXCLUSION AREA BOUNDARY AND SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS
No change

FIGURE 5.1-2 LOW POPULATION ZONE
No change.

5.5 METEOROLOGICAL TOWER LOCATION
No change.

5.6 FUEL STORAGE

In the title, change "FUEL STORAGE" to "SPENT FUEL STORAGE". Change "5.6.1 CRITICALITY" to "CRITICALITY". Change "5.6.1.1" to "5.6.1". Delete "NEW FUEL". Delete Subsection 5.6.1.2 Change Page "5-5" to Page "5-4". Move the remainder of Section 5.6 from Page 5-5a to the new Page 5-4. Delete Figures 5.6-1 and 5.6-2.

2. The following were deleted, including associated tables:

5.2 CONTAINMENT

5.3 REACTOR CORE

5.4 REACTOR COOLANT SYSTEM

5.7 REACTOR VESSEL DESIGN TRANSIENTS

Reason

Defueled Condition and Editorial Changes.

The changes to Section 5.6 are proposed, since the new fuel should be removed from the site prior to this proposed amendment being approved.

Basis

1. These subsections are retained at this time since they reflect systems that are retained in Technical Specification Sections 3/4. The changes to Section 5.6 reflect the removal of the new fuel from the HNP.

2. The deleted subsections and their associated tables address those design features associated with fuel in the reactor vessel or in the containment. Because the plant has permanently ceased operations, the reactor has been permanently defueled and fuel is no longer allowed in the containment; these subsections are no longer required.

6.0 ADMINISTRATIVE CONTROLS

Changes

1. The following are retained with noted changes:

6.1 RESPONSIBILITY

Change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer".

6.2. ORGANIZATION

- In Subsection 6.2.1.a, last line, change "the Quality Assurance Topical Report." to "the Connecticut Yankee Quality Assurance Program (CYQAP)."

- In Subsection 6.2.1.b, replace the statement with, "An individual management position in the onsite organization shall be responsible for overall unit safe operation and shall have control over those onsite activities and resources necessary for safe operation and maintenance of the plant."

- In Subsection 6.2.1.c, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer".

- In Subsection 6.2.2.b, replace the statement with, "At least one licensed Operator shall be in the control room when irradiated fuel is in the spent fuel pool;".

- In Subsection 6.2.2.c, change "when fuel is in the reactor;" to "during fuel handling operations;".

- In Subsection 6.2.2.d, change "CORE ALTERATIONS" to "fuel handling operations".

TABLE 6.2-1 MINIMUM SHIFT CREW COMPOSITION

Delete the entire "MODE 1, 2, 3, or 4" column (including the column header). Delete the column header "MODE 5 or 6". Delete the STA row (i.e., "STA 1* None"). Under "Abbreviations" delete "STA - Shift Technical Advisor". In the 2nd paragraph, delete the 1st sentence. In the 2nd paragraph, 2nd sentence, change "the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual" to "the Shift Manager from the control room, an individual".

6.3 FACILITY STAFF QUALIFICATIONS

Delete Subsection 6.3.1.2.

6.4 TRAINING

No change.

6.5 REVIEW AND AUDIT

In Subsection 6.5.1.2, in the 1st sentence, change "eleven" to "seven". In the 2nd sentence, delete the existing listing and replace with:

> "Plant Operations,
> Decommissioning,*
> Engineering,
> Maintenance,
> Health Physics,
> Chemistry / Radiochemistry,
> Quality Assurance, and
> Security.*"

Add footnote:

> "*These areas are exempt from the 5-year experience requirement."

In Subsections 6.5.1.6.e, 6.5.1.6.h, and 6.5.1.8, delete "Chairperson of the".

In Subsection 6.5.1.6.e, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer".

In Subsection 6.5.1.7.c, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer", in two places.

In Subsection 6.5.2.1.b, delete the existing listing and replace with:

"Plant Operations,
Decommissioning,*
Engineering,
Radiological Safety,
Chemistry / Radiochemistry,
Quality Assurance, and
Environmental Protection.*"

Add footnote:

"*These areas are exempt from the 5-year experience requirement."

In Subsection 6.5.2.1, last paragraph, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer", in two places.

In Subsection 6.5.2.2, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer" and change "seven members" to "five members".

In Subsection 6.5.2.3, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer".

In Subsection 6.5.2.5, change "operation of the same Northeast Utilities' nuclear unit." to "operation of the Haddam Neck Plant."

In Subsection 6.5.2.6.k, change "Audits and audit plans." to "Audits and the Annual Audit Plan."

In Subsection 6.5.2.6, last paragraph, change "Executive Vice President and Chief Nuclear Officer" to "designated CYAPCO corporate officer".

In Subsection 6.5.2.7, change "Nuclear Group Procedures" to "appropriate administrative procedures". In Subsection 6.5.2.7.c, delete "all". In Subsection 6.5.2.7.g, change "Executive Vice President and Chief Nuclear Officer." to "designated CYAPCO corporate officer."

In Subsection 6.5.2.8 items b and c are deleted. This subsection should now read: "Written records of reviews and audits shall be maintained. As a minimum these records shall include results of the activities conducted under the provisions of Section 6.5.2."

6.6 REPORTABLE EVENT ACTION

In Subsection 6.6.1.b, delete "Chairperson of the" and change "Executive Vice President and Chief Nuclear Officer." to "designated CYAPCO corporate officer."

6.8 PROCEDURES AND PROGRAMS

In Subsection 6.8.1.f, change "PROCESS CONTROL PROGRAM" to "Process Control Program".

6.9 REPORTING REQUIREMENTS

In the footnote associated with Subsection 6.9.1.5.a change "Section 20.407" to "Section 20.2206". Delete Subsections 6.9.1.1, 6.9.1.2, 6.9.1.3, 6.9.1.5.b, 6.9.1.5.c, 6.9.1.8, and 6.9.1.9.

6.10 RECORD RETENTION

In Subsection 6.10.2.a, delete "covering time interval at each power level". In Subsection 6.10.2.e, delete "reactor". In Subsection 6.10.3.d, change "expsure" to "exposure". In Subsection 6.10.3.f, change statement to read "Records of transient or operational cycles of the reactor vessel;". In Subsection 6.10.3.h, change statement to read "Records of inservice inspections performed under previous amendments to these Technical Specifications;". In Subsection 6.10.3.m, change "Radiological Effluent Monitoring and Offsite Dose Calculation Manual" to "RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL".

6.11 RADIATION PROTECTION PROGRAM

No change.

6.12 HIGH RADIATION AREA

In Subsection 6.12.1, 1st paragraph, 1st sentence:

- Change "20.203(c)(5)" to "20.1601(c)";
- Change "20.203(c)," to "20.1601,"; and
- Change "45 cm (18 in.)" to "30 cm (12 in.)".

In Subsection 6.12.2, 1st paragraph, 1st sentence, change "but less than 1000 mR/h at 45 cm (18 in.)" to "but equal to or less than 1000 mR/h at 30 cm (12 in.)".

6.13 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

Change "Radiological Effluents Monitoring Manual" to "RADIOLOGICAL EFFLUENT MONITORING MANUAL (REMM)". Change "Offsite Dose Calculation Manual" to "OFFSITE DOSE CALCULATION MANUAL (ODCM)".

6.14 RADIOACTIVE WASTE TREATMENT

Change "PROCESS CONTROL PROGRAM" to "Process Control Program".

2. The following were deleted:

6.7 SAFETY LIMIT VIOLATION

6.15 SYSTEMS INTEGRITY

6.16 PASS/SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

Reason

Defueled Condition and Editorial Changes.

The changes associated with Subsection 6.2.1.b and with "designated CYAPCO corporate officer" was to allow flexibility for CYAPCO to delineate who has the responsibility, without requesting a Technical Specification change. The changes to PORC and NSAB membership requirements reflect the change in staffing levels and mission scope (i.e., power operation to decommissioning). The changes of "the Chairperson of the NSAB" to "the NSAB" is to streamline the distribution of material to NSAB members. The changes to Subsection 6.5.2.8 is to delete unnecessary requirements. The changes associated with 10CFR20 are for the sections to reflect the latest version of 10CFR20.

Basis

1. For these subsections, the changes are discussed below:

   6.1 Responsibility

   The change to "designated CYAPCO corporate officer", provides the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

   6.2. Organization

   The change to Subsection 6.2.1.a, reflects the correct title. The change to Subsection 6.2.1.b, provides the flexibility for CYAPCO to delineate who has the overall onsite responsibility for safety without requesting a Technical Specification change, since as decommissioning progresses, the plant organization will change. In Subsection 6.2.1.c, the change to "designated CYAPCO corporate officer", provides the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change. The changes to Subsection 6.2.2, reflect the permanent defueled plant condition.

TABLE 6.2-1 Minimum Shift Crew Composition

The deleted material deals with staffing requirements during MODES 1 through 4. Although in the defueled condition, the plant is not in MODE 6, those requirements were retained. Because the plant has permanently ceased operations and the reactor has been permanently defueled, MODES 1 through 6 cannot be achieved. Therefore, the deleted staffing requirements are no longer necessary. The "Shift Supervisor" to "Shift Manager" change, provides the correct title.

6.3 Facility Staff Qualifications

The deleted material deals with Shift Technical Advisor (STA) requirements. The STA is only required during reactor operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these STA requirements are no longer necessary.

6.5 Review And Audit

In Subsection 6.5.1.2, the changes reflect the smaller plant staff and that the mission for the HNP has changed (i.e., power operation to the defueled condition and subsequent decommissioning). The exceptions to the 5-year requirement are for those areas of expertise where it may be difficult to find someone with that level of experience.

In Subsections 6.5.1.6.e and 6.5.1.8, the changes are administrative. The NSAB staff distributes material to the NSAB members. Presently, material for the NSAB which is required to be sent to the NSAB Chairperson, would then handed to the NSAB staff for distribution. The change streamlines the process by allowing material to be sent directly to the NSAB staff.

In Subsection 6.5.1.6.e, the change to "designated CYAPCO corporate officer", provides the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

In Subsection 6.5.1.7.c, the changes to "designated CYAPCO corporate officer", provide the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

In Subsection 6.5.2.1.b, the change in NSAB experience, reflects that the mission for the HNP has changed (i.e., power operation to the defueled condition and subsequent decommissioning). The exceptions to the 5-year requirement are for those areas of expertise where it may be difficult to find someone with that level of experience.

In the last paragraph of Subsection 6.5.2.1, the changes to "designated CYAPCO corporate officer", provide the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

In Subsection 6.5.2.2, the change to "designated CYAPCO corporate officer", provide the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change. The change in the size of the NSAB reflects the change in workload and scope (i.e., power operation to the defueled condition and subsequent decommissioning).

In Subsection 6.5.2.3, the change to "designated CYAPCO corporate officer", provide the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

In Subsection 6.5.2.5, since there is a separate NSAB for the HNP, the change delineates that the quorum restriction is only against the HNP management.

In Subsection 6.5.2.6.k, NSAB reviews the Annual Audit Plan. It is not intended that the NSAB review all audit plans.

In the last paragraph of Subsection 6.5.2.6, the change to "designated CYAPCO corporate officer", provides the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

In Subsection 6.5.2.7, the editorial change reflects that the NSAB audits are now controlled by the administrative procedures not the Nuclear Group Procedures. In Subsection 6.5.2.7.c, the word "all" is unnecessary. In Subsection 6.5.2.7.g, the change to "designated CYAPCO corporate officer", provides the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

In Subsection 6.5.2.8, the deleted material is redundant to what is already required of the NSAB. Items considered under Subsections 6.5.1.6.a through 6.5.1.6.d and Section 6.8 are also considered by the NSAB under Subsection 6.5.2.6. In addition, the requirement of oversight review of PORC minutes was deleted in Technical Specification Amendment 181.

6.6 Reportable Event Action

In Subsection 6.6.1.b, the 1st change is administrative. The NSAB staff distributes material to the NSAB members. Presently, material for the NSAB which is required to be sent to the NSAB Chairperson, would then handed to the NSAB staff for distribution. The change streamlines the process by allowing material to be sent directly to the NSAB staff. The 2nd change, which is the change to "designated CYAPCO corporate officer", provides the flexibility for CYAPCO to delineate who has the responsibility without requesting a Technical Specification change, since as decommissioning progresses, the CYAPCO organization will change.

6.8 Procedures and Programs

In Subsection 6.8.1.f, the change is editorial.

6.9 Reporting Requirements

The change to the note is due to the revisions made to 10CFR20. NUREG-1431 ("Standard Technical Specifications Westinghouse Plants") provided the basis for the reference change. While the requirements of the revised 10CFR20 had already been implemented by CYAPCO, the reference in the Technical Specifications had not been updated.

The deleted material deals with reports associated with startup or reactor operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these reports are no longer necessary.

6.10 Record Retention

The Subsection 6.10.2.a and Subsection 6.10.2.e changes delete references to reactor operation. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these references are no longer necessary.

The Subsection 6.10.3.d change is editorial since it corrects a misspelling. Subsection 6.10.3.f requires that records of transient or operational cycles be retained for components described in Table 5.7-1. Because the plant has permanently ceased operations and the reactor has been permanently defueled, Table 5.7-1 has been deleted (i.e., reactor vessel design transients are no longer applicable). Therefore, Subsection 6.10.3.f now refers to the reactor vessel. In Subsection 6.10.3.h, the inservice inspection requirements have been deleted from the technical specifications. Therefore, Subsection 6.10.3.h now refers to previous Technical Specification amendments. In Subsection 6.10.3.m, the change is editorial.

6.12 High Radiation Area

The change "equal to or less than 1000 mR/h" eliminates the ambiguity of what are the controls at precisely 1000 mR/h and make the statement consistent with the rest of Section 6.12. The remainder of the changes are due to the revisions made to 10CFR20. NUREG-1431 ("Standard Technical Specifications Westinghouse Plants") provided the basis for the reference changes and the requirements of 10CFR20.1601(a)(1) provided the basis for the distance changes. While the more stringent requirements of 10CFR20.1601 had already been implemented by CYAPCO, the references and requirements in the Technical Specifications had not been updated.

6.13 Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)

The changes are editorial.

6.14 Radioactive Waste Treatment

The change is editorial.

2. The deleted sections address administrative controls of aspects of the plant design features associated with fuel in the reactor vessel. Because the plant has permanently ceased operations and the reactor has been permanently defueled, these subsections are no longer required. Additional details are provided below:

6.7 Safety Limit Violation

Section 6.7 describes what actions must be taken if a Safety Limit in Section 2.0 is violated. Because the plant has permanently ceased operations and the reactor has been permanently defueled, there are no Safety Limits that can be violated. Section 2.0 is no longer required. Therefore, Section 6.7 is no longer required.

6.15 Systems Integrity

Section 6.15 requires implementing a program to reduce the leakage from systems that could contain highly radioactive fluids during a serious transient or accident (i.e., LOCA, MSLB, MFWLB, SBLOCA, SG tube failure). Because the plant has permanently ceased operations and the reactor has been permanently defueled, such serious transients or accidents are no longer possible. Therefore, Section 6.15 is no longer required.

6.16 Pass/Sampling and Analysis of Plant Effluents

Section 6.16 requires implementing a program for post-accident sampling. Because the plant has permanently ceased operations and the reactor has been permanently defueled, such accidents are no longer possible. Therefore, Section 6.16 is no longer required.

Docket Nos. ')-213
C ɪ -97-006

Attachment 4

Haddam Neck Plant

Proposed Revision To Operating License And Technical Specifications

Defueled Operating License And Technical Specifications

Safety Assessment And No Significant Hazards Consideration

May 1997

Haddam Neck Plant
Proposed Revision To Operating License And Technical Specifications
Defueled Operating License And Technical Specifications
Safety Assessment And No Significant Hazards Consideration

Safety Assessment

The plant has permanently ceased operations, the reactor has been permanently defueled, and the spent fuel is stored in the spent fuel pool.

The Operating License and Technical Specification requirements that are proposed to be removed deal with structures, systems, components, programs and/or administrative controls that are only needed when the plant is operating, fuel is in the reactor and/or fuel is in containment. In addition, the editorial changes included in this proposed revision provide correct spelling, brevity, clarity and/or consistency. None of these changes affect plant configuration nor do they affect plant operations. Therefore, the changes do not affect the safety of the plant.

An engineering review has been completed which will result in some of the specifications that are proposed to be deleted, being appropriately addressed in the Technical Requirements Manual (TRM). Although these specifications are no longer required due to mode applicability, the relevant portions will be retained to support spent fuel pool cooling and monitoring and Reactor Coolant System (RCS) decontamination. As the decay heat load decreases and the RCS is decontaminated, the need for these administrative controls will be reevaluated under 10CFR50.59. The appropriate Technical Specification requirements that will be transferred to the TRM provide an additional set of provisions that are formally controlled. This defense-in-depth approach increases the level of plant safety above the minimum set of controls required by the revised accident analyses.

With respect to the Service Water System (Specification 3/4.7.3), Electrical Power Systems (Specification 3/4.8) and spent fuel pool makeup, the basis for placing appropriate requirements in the TRM is due to the reduced heat load in the spent fuel pool.

The plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150 °F. The initial pool temperature of 150 °F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150 °F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide an alternate forced cooling or makeup to the spent fuel pool for a service water system failure. Therefore, operability of spent fuel pool cooling does not require service water, electrical power, or makeup water to be immediately available.

Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 ½ hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

The third type of Technical Specification changes discussed in this proposed license amendment are those Technical Specifications that are retained in modified form. The modified Technical Specifications are those requirements that deal with structures, systems, components, programs and/or administrative controls that are needed while spent fuel is stored in the spent fuel pool.

The specifications that are in this category are:

- 3.0.1, 3.0.4, 4.0.1, 4.0.4, "Applicability of Limiting Conditions for Operation and Applicability of Surveillance Requirements"
- 3.3.3.8, "Radioactive Gaseous Effluent Monitoring Instrumentation"
- Table 3.3-10, Item 1.a, "Main Stack Noble Gas Activity Monitor"
- 3.11.2.3, "Dose Rate" and Bases
- 3.11.2.3, "Dose, Radioiodines, Radioactive Material in Particulate Form and Radionuclides Other Than Noble Gases" and Bases
- 6.1.1, "Responsibility"
- 6.2.1, "Onsite and Offsite Organizations"
- 6.2.2, "Facility Staff"
- 6.5.1, "Plant Operations Review Committee"
- 6.5.2, "Nuclear Safety Assessment Board"
- 6.6, "Reportable Event Action"
- 6.9.15, "Annual Reports"
- 6.10, "Record Retention"
- 6.12, "High Radiation Area"

The changes to Technical Specification 3/4.0, "Applicability of Limiting Conditions for Operation and Surveillance Requirements" replace the operational mode conditions with the applicable conditions specified in the individual Technical Specifications. These changes remove words that are no longer relevant to these Technical Specifications but can still be applied to the remaining Technical Specifications. Therefore, these changes preserve the level of safety provided by the remaining specifications.

The changes to Technical Specification 3.3.3.8, "Radioactive Gaseous Effluent Monitoring Instrumentation" and Table 3.3.-10 delete the trip function from the main stack noble gas activity monitor. The changes to Technical Specifications 3.11.2.1, Dose Rate, and 3.11.2.3, Dose, delete the requirement to include the radioiodine isotopes in the dose calculations. These changes are based on the following:

> The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

Therefore, the changes to this group of Technical Specifications do not diminish the level of protection of the public health and safety.

The changes to Section 6 are discussed by the following categories:

<u>Titles</u>

The changes to Section 6.1.1, 6.2.1.c, 6.5.1, 6.5.2, 6.6 and 6.10 change the specific title of the corporate officer to a general title. This change is editorial in that the specified title of the corporate officer is not required for Technical Specifications. The change to Section 6.2.1.b also removes the specific title for the senior management position onsite. The change to Table 6.2.1 reflects the change in title of the Shift Supervisor to the Shift Manager. The duties have not changed.

<u>Defueled Condition</u>

The changes to Section 6.2.2 and 6.10 reflect the permanently defueled condition. The present wording for operational modes, fuel in the reactor, core alterations, power operation logs and reactor tests were replaced, where applicable, with restrictions for irradiated fuel in the spent fuel pool.

<u>Quality Assurance</u>

The change to 6.2.1.a reflects an editorial change in title for the Quality Assurance organization. These changes were discussed in the Connecticut Yankee Quality Assurance Program submitted to the NRC staff on April 25, 1997.

<u>PORC</u>

The changes to Specification 6.5.1.2 revise the PORC composition to reflect the permanently defueled condition and the need to add decommissioning expertise to the committee. The reduction to seven members reflects the areas of expertise that are applicable to the permanently defueled condition.

<u>NSAB</u>

The changes to Specification 6.5.2.1 revise the NSAB composition to reflect the permanently defueled condition and the need to add decommissioning expertise to the committee. The reduction to five members reflects the areas of expertise that are applicable to the permanently defueled condition.

The changes to Specification 6.5.2.8, NSAB Records, removes the duplicative requirements of parts b and c. The NSAB maintains records of its reviews conducted under 6.5.2. PORC maintains records of its reviews and this requirement does not need to be repeated in this section. In addition, the requirement of oversight review of PORC minutes was deleted in Technical Specification Amendment 181.

10CFR20 References

The change to the footnote associated with Technical Specification 6.9.1.5.a ("Annual Reports") and changes to Technical Specification 6.12 ("High Radiation Area") are due to the revisions made to 10CFR20. NUREG-1431 ("Standard Technical Specifications Westinghouse Plants") provided the basis for the reference changes and the requirements of 10CFR20.1601(a)(1) provided the basis for the distance changes. While the more stringent requirements of 10CFR20.1601 had already been implemented by CYAPCO, the references and requirements in the Technical Specifications had not been updated.

Therefore, the changes to Section 6 of the Technical Specifications are administrative or editorial in nature and do not diminish the level of protection of the public health and safety.

The last group of Technical Specifications are those that will remain. Changes to these specifications are editorial in order to maintain consistency with the modified specifications or to change specification titles to reflect the permanently defueled condition. These specifications provide restrictions for seismic and meteorological instrumentation, radioactive effluent monitoring instrumentation, sealed sources, spent fuel pool water level, fuel movement, and reactivity conditions, radioactive effluents, design features and administrative controls.

Significant Hazards Consideration

CYAPCO has reviewed the proposed changes to the Operating License and the Technical Specifications in accordance with 10CFR50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated.*

   Because the present plant configuration, many of the postulated accidents previously evaluated (i.e., LOCA, MSLB, etc.) are no longer possible. The accidents previously evaluated that are still applicable to the plant are fuel handling accidents and gaseous and liquid radioactive releases.

There is no significant increase in the probability of a fuel handling accident since refueling operations have ceased. In fact, there is more likely a decrease in probability of a fuel handling accident since the need to move/rearrange fuel assemblies is minimal until they are removed from the spent fuel pool (i.e., for dry cask storage or for transferring to USDOE possession).

The radiological consequences of a gaseous or liquid radioactive release are bounded by the fuel handling accident. With the plant defueled and permanently shutdown, the demands on the radwaste systems is lessened since no new radioisotopes are being generated by irradiation or fission. Therefore, there is no increase in the probability or consequences of a gaseous or liquid radioactive release.

The changes to the Operating License reflect the permanently defueled condition for power level and fuel movement restrictions and the fire protection regulation which is applicable for a permanently defueled plant.

With respect to the Service Water System (Specification 3/4.7.3), Electrical Power Systems (Specification 3/4.8) and spent fuel pool makeup, the basis for placing appropriate requirements in the TRM is due to the reduced heat load in the spent fuel pool.

The plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150 °F. The initial pool temperature of 150 °F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150 °F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide an alternate forced cooling or makeup to the spent fuel pool for a service water system failure. Therefore, operability of spent fuel pool cooling does not require service water, electrical power, or makeup water to be immediately available.

Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 ½ hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

The changes to Technical Specification 3.3.3.8, "Radioactive Gaseous Effluent Monitoring Instrumentation" and Table 3.3.-10 delete the trip function from the main stack noble gas activity monitor. The changes to Technical Specifications 3.11.2.1, Dose Rate, and 3.11.2.3, Dose, delete the requirement to include the radioiodine isotopes in the dose calculations. These changes are based on the following:

> There is no significant increase in the consequences of a fuel handling accident since the accident scenarios assume an assembly with significant amounts of radioactive iodine or noble gas. The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs. In fact, due to this decreased radioactive inventory, there is a significant decrease in the consequences of a fuel handling accident.

Section 6 Changes

Titles

The changes to Section 6.1.1, 6.2.1.c, 6.5.1, 6.5.2, 6.6 and 6.10 change the specific title of the corporate officer to a general title. This change is editorial in that the specified title of the corporate officer is not required for Technical Specifications. The change to Section 6.2.1.b also removes the specific title for the senior management position onsite. The change to Table 6.2.1 reflects the change in title of the Shift Supervisor to the Shift Manager. The duties have not changed.

Defueled Condition

The changes to Section 6.2.2 and 6.10 reflect the permanently defueled condition. The present wording for operational modes, fuel in the reactor, core alterations, power operation logs and reactor tests were replaced, where applicable, with restrictions for irradiated fuel in the spent fuel pool.

Quality Assurance

The change to 6.2.1.a reflects an editorial change in title for the Quality Assurance organization. These changes were discussed in the Connecticut Yankee Quality Assurance Program submitted to the NRC staff on April 25, 1997.

PORC

The changes to Specification 6.5.1.2 revise the PORC composition to reflect the permanently defueled condition and the need to add decommissioning expertise to the committee. The reduction to seven members reflects the areas of expertise that are applicable to the permanently defueled condition.

NSAB

The changes to Specification 6.5.2.1 revise the NSAB composition to reflect the permanently defueled condition and the need to add decommissioning expertise to the committee. The reduction to five members reflects the areas of expertise that are applicable to the permanently defueled condition.

The changes to Specification 6.5.2.8, NSAB Records, removes the duplicative requirements of parts b and c. The NSAB maintains records of its reviews conducted under 6.5.2. PORC maintains records of its reviews and this requirement does not need to be repeated in this section. In addition, the requirement of oversight review of PORC minutes was deleted in Technical Specification Amendment 181.

10CFR20 References

The change to the footnote associated with Technical Specification 6.9.1.5.a ("Annual Reports") and changes to Technical Specification 6.12 ("High Radiation Area") reflect the current NRC requirements for defining and controlling a High Radiation Area. These requirements do not modify equipment nor do they affect how the equipment is maintained or operated. Therefore, there is no effect on the probability or consequences of an accident previously evaluated.

The editorial changes included in this proposed revision provide correct spelling, brevity, clarity and/or consistency. None of these changes affect plant configuration nor do they affect plant operations.

In summary, the administrative and editorial changes do not have an effect on the probability or consequences of an accident previously evaluated. These changes remove text no longer applicable and provide consistency and clarity throughout.

Based on the above, the proposed changes to the Operating License and the Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Create the possibility of a new or different kind of accident from any accident previously evaluated.*

There is no change in how spent fuel is stored or moved in the spent fuel pool. Therefore, the postulated fuel handling accidents are still bounding and are still considered as credible postulated accidents.

With respect to the Service Water System (Specification 3/4.7.3), Electrical Power Systems (Specification 3/4.8) and spent fuel pool makeup, the basis for placing appropriate requirements in the TRM is due to the reduced heat load in the spent fuel pool.

The plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150 °F. The initial pool temperature of 150 °F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150 °F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide an alternate forced cooling or makeup to the spent fuel pool for a service water system failure. Therefore, operability of spent fuel pool cooling does not require service water, electrical power, or makeup water to be immediately available.

Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 ½ hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

The changes to Technical Specification 3.3.3.8, "Radioactive Gaseous Effluent Monitoring Instrumentation" and Table 3.3-10 delete the trip function from the main stack noble gas activity monitor. The changes to Technical Specifications 3.11.2.1, Dose Rate, and 3.11.2.3, Dose, delete the requirement to include the radioiodine isotopes in the dose calculations. These changes are based on the following:

> The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

Therefore, the changes to Technical Specification related to radioactive iodine and noble gas isotopes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Based on the above, the proposed changes to the Operating License and the Technical Specifications do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Involve a significant reduction in a margin of safety.*

With respect to the Service Water System (Specification 3/4.7.3), Electrical Power Systems (Specification 3/4.8) and spent fuel pool makeup, the basis for placing appropriate requirements in the TRM is due to the reduced heat load in the spent fuel pool.

The Technical Specification basis states that the time to spent fuel pool boiling after a loss of forced cooling following a full core offload is 7 hours.

The plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150 °F. The initial pool temperature of 150 °F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150 °F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide an alternate forced cooling or makeup to the spent fuel pool for a service water system failure. Therefore, operability of spent fuel pool cooling does not require service water, electrical power, or makeup water to be immediately available.

Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 ½ hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

There is no change in how spent fuel is stored or moved in the spent fuel pool.

The existing Technical Specification basis includes a significant inventory of I-131, I-132, I-133, I-134 and I-135 in the spent fuel.

The plant was shutdown on July 22, 1996. Except for I-125 (half-life ~59.5 days), I-129 (half-life ~1.6E7 years), and Kr-85 (half-life ~10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for "Dose Equivalent I-131" ("Standard Technical Specifications, Westinghouse Plants," NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 10CFR100 dose limits and the EPA PAGs.

Based on the above, the proposed changes to the Operating License and the Technical Specifications do not involve a significant reduction in a margin of safety.

Docket Nos. 50-213
CY-97-006

Attachment 5

Haddam Neck Plant

Proposed Revision To Operating License And Technical Specifications

Defueled Operating License And Technical Specifications

Calculations

May 1997

Enclosed are the following calculations:

- SFP-97-1575-DY, Revision 1, "Decay Heat and Heatup Rate Analysis for the Connecticut Yankee SFP."

- XX-XXX-60RA, Revision 1, "Radiological Assessment of a Spent Fuel Shipping Cask Drop in the CY Spent Fuel Pool," Note: This calculation also provides the radiological assessment for a fuel handling accident.

- CYRESIN-01578-RY, Revision 0, "Radiological Consequences From a Resin Accident."

Number of Calc Pages: 46
Number of Attachment Pages: 0
Total Number of Pages: 46

Decay Heat and Heatup Rate Analysis for the Connecticut Yankee SFP

TITLE

CALCULATION #	REV #	Vendor Calc#
SFP-97-1575-DY	1	HI-971o7/

System SFC **Structure** FB **Component**

Executive Summary

The purpose of this calculation is to incorporate the subject Holtec International calculation into the CY Nuclear Records System.

The CY facility is currently entering the decommissioning phase of its operating life. While the decommissioning process is carried out and plant systems are downgraded and removed, adequate cooling and radiation shielding for the spent fuel assemblies stored in the plant's spent fuel pool (SFP) must still be provided.

The most credible scenario which could threaten the security of the stored fuel in the SFP is a loss of forced cooling. Following a loss of forced cooling, only passive means of heat rejection will be available to reject the decay heat load. If the available passive heat rejection mechanisms are not sufficient to remove all of the generated decay heat, the temperature of the SFP will rise.

In order to quantify the effects of a loss of forced cooling event on the CY SFP during the decommissioning process, the decay heat load and corresponding maximum heatup rate are calculated over this time period. The SFP decay heat load will continually reduce over the decommissioning period. The SFP heatup rate is proportional to the net decay heat load and will therefore decrease over time. Thus, as the decay heat load reduces so to does the maximum heatup rate.

The results of these evaluations provide a conservative picture of the potential consequences on the SFP of, and the time available for recovery from, a postulated loss of forced cooling. These results can then be used to assist decommissioning schedule planning and emergency recovery procedures development

Does this calculation:		
1.	Support a DCR, MMOD, an independent review method for a DCR ,or confirm test results for an installed DCR? If yes, indicate the DCR, MMOD number and/or Test Procedure number.	Yes ☐ No ☒
2.	Support independent analysis? If yes, indicate the procedure, work control or other reference it supports. Proposed Technical Specification Change Request C-19-96	Yes ☒ No ☐
3.	Revise, supersede, or void existing calculations? If yes, indicate the calculation number and revisions.	Yes ☐ No ☒
4.	Involve QA or QA-related systems, components or structures?	Yes ☒ No ☐
5.	Impact the Unit licensing basis, including technical specifications, FSAR, procedures or licensing commitments? If yes, identify appropriate change documents. PTSCR C-19-96	Yes ☒ No ☐

Approvals (Print/Signature)		
Preparer Edward P. Mullarkey	Edward P. Mullarkey	Date: 4-23-97
Independent Reviewer Stephen J. Weyland	S. J. Weyland	Date: 4/23/97
Supervisor Robert W. Pritchard / R. W. P		Date: 4/23/97

NOTE: See Nuclear Engineering Memo NE-97-F-100 for verification of fuel data.

CTP DATA BASE INPUTS

Calculation Number: SFP-97 (prefix) 1575 (sequence number) DY (suffix) Date: 4/16/97 Revision: 1

Vendor Calculation Number/Other: HI-971677

CCN # ______ QA ☒ Yes ☐ No Supersedes Cacl: ______

Superseded By: ______

Unit	EWR Number	Component Id	Computer Code	Rev. #/Level
CYAPCo		SFC		

PMMS CODES *		
Structure	System	Component
FB	SFC	

*The codes required must be alpha codes designed for structure, system and component.

Comments:

This calculation determines the spent fuel decay heat load in the spent fuel pool and the resultant heat-up rates. For conservatism, the calculation does not account for ambient losses.

Reference Calculation	Reference Drawing	Sheet