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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.152
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of of New York, Inc. (the licensee) dated March 18, 1988, which was superseded by letter dated March 27, 1990, as supplemented by letter dated May 11, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as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;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

ATTACHMENT TO LICENSE AMENDMENT NO. 152

FACILITY OPERATING LICENSE NO. DPR-5

DOCKET NO. 50-003

Revise Appendix A as follows:

Remove Pages	Insert Pages
All	All attached pages

Revise Appendix B as follows:

Remove Pages	Insert Pages
All	All attached pages

APPENDIX

TO

FACILITY OPERATING LICENSE DPR-26

FOR

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2

DOCKET NO. 50-247

TECHNICAL SPECIFICATIONS AND BASES

Amendment No. 152

TECHNICAL SPECIFICATIONS

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1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

1.1 a. RATED POWER

A steady state reactor thermal power of 3071.4 MWT.

b. THERMAL POWER

The total core heat transfer rate from the fuel to the coolant.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 <u>Cold Shutdown Condition</u>

When the reactor is subcritical by at least 1% $\Delta k/k$ and Tavg is $\leq 200^{\circ}F$.

1.2.2 <u>Hot Shutdown Condition</u>

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is $> 200^{\circ}F$ and $\leq 555^{\circ}F$.

1.2.3 <u>Reactor Critical</u>

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

1.2.4 <u>Power Operation Condition</u>

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

1.2.5 Refueling Operation Condition

Any operation involving movement of core components when the vessel head is completely unbolted.

1.3 OPERABLE-OPERABILITY

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its intended safety function(s). Implicit in this definition shall be the assumption that necessary instrumentation, controls, electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its safety function(s) are also capable of performing their related support functions.

1.4 PROTECTIVE INSTRUMENTATION LOGIC

1.4.1 Analog Channel

An arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. An analog channel loses its identity where single action signals are combined.

1.4.2 Logic Channel

A group of relay contact matrices which operate in response to the analog channels signals to generate a protective action signal.

1.5 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

1.6 INSTRUMENTATION SURVEILLANCE

1.6.1 Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel with other independent channels measuring the same variable.

1.6.2 Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.6.3 Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.6.4 Source Check

A Source Check is the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.7 CONTAINMENT INTEGRITY

Containment integrity is defined to exist when:

a. All non-automatic containment isolation valves which are not required to be open during accident conditions, except those required to be open for normal plant operation or testing as identified in Specification 3.6.A, are closed and blind flanges are installed where required.

b. The equipment door is properly closed.

1.12 IDENTIFIED LEAKAGE

Identified leakage shall be:

a. Reactor coolant system leakage into closed systems such as pump seal or valve packing leaks that are captured and conducted to a collecting tank, or

b. Reactor coolant system leakage through a steam generator to the secondary system, or

c. Reactor coolant system leakage through the RCS/RHR pressure isolation valves, or

d. Reactor coolant system leakage into the containment free volume from sources that are both specifically located and known either not to interfere with the operation of required leakage detection systems or not to be pressure boundary leakage.

1.13 UNIDENTIFIED LEAKAGE

Unidentified leakage shall be all reactor coolant system leakage which is not identified leakage.

1.14 DOSE EQUIVALENT I-131

The dose equivalent I-131 is that concentration of I-131 which 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 conversion factors shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

1.15 GASEOUS RADWASTE TREATMENT SYSTEM

A Gaseous Radwaste Treatment System is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.16 MEMBER(S) OF THE PUBLIC

Member(s) of the public includes all persons who are not occupationally associated with the site. This category does not include employees of either utility, their contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.

1.17 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The Offsite Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

1.18 PROCESS CONTROL PROGRAM (PCP)

The Process Control Program (PCP) is a manual containing and/or referencing selected operational information concerning the solidification of radioactive wastes from liquid systems.

1.19 PURGE - PURGING

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

1.20 SITE BOUNDARY

The site boundary is that line beyond which the land is neither owned, leased, nor otherwise controlled by either site licensee.

1.21 SOLIDIFICATION

Solidification is the conversion of wet wastes into a form that meets shipping and burial ground requirements.

1.22 UNRESTRICTED AREA

An unrestricted area is any area at or beyond the site boundary access to which is not controlled by either site licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.23 VENTILATION EXHAUST TREATMENT SYSTEM

A Ventilation Exhaust Treatment System is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmosphere cleanup systems are not considered to be Ventilation Exhaust Treatment System components.

1.24 VENTING

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required.

TABLE 1-1

Frequency Notation

Notation	Test Frequency/Requirements	Surveillance Interval
Shift (S)	At least twice per calendar day	N.A.
Daily (D)	At least once per calendar day	N.A.
Weekly (W)	At least once per week	7 days
Monthly (M)	At least once per month	31 days
Quarterly (Q)	At least once per three months	92 days
Semi-Annually (SA)	At least once per six months	6 months
Annually (A)	At least once per 12 months	12 months
Refueling (R)	At least once per 18 months	18 months
S/U	Prior to each reactor startup	--
P	Completed prior to each release	--
N.A.	Not Applicable	--

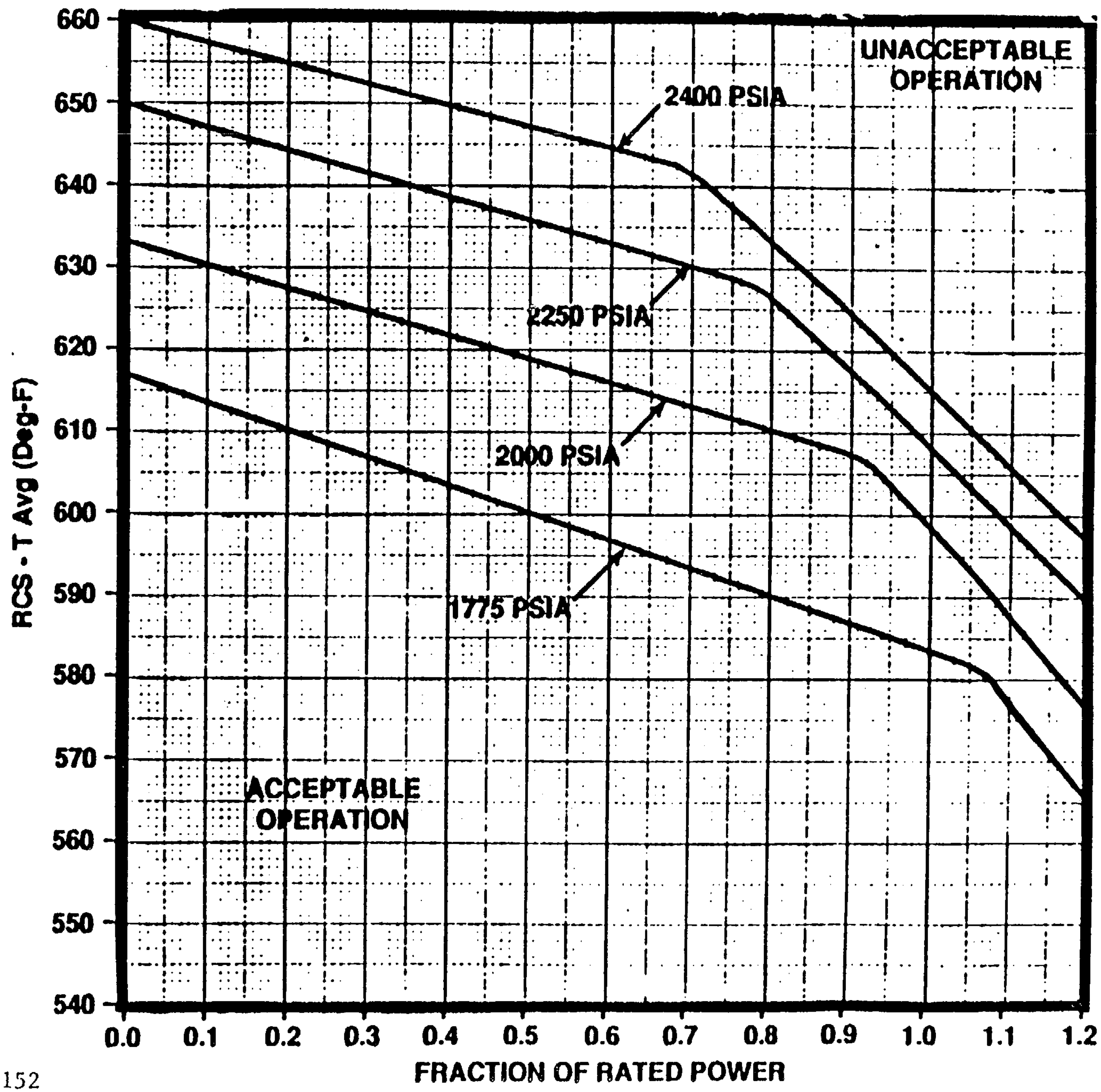
Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters: thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 L-grid correlation for analysis of the LOPAR fuel, and the WRB-1 correlation for evaluation of the OFA. These DNB correlations have been developed to predict the DNB flux and location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB design basis is as follows: There must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that the DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin is maintained by performing DNB design evaluations to a higher DNBR value, called the Safety Limit DNBR. This margin is sufficient to cover applicable rod bow DNB penalties and provide margin for use in design and operational flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. These curves are based on a peak nuclear hot channel factor of 1.62 for the LOPAR fuel and a 1.65 for the OFA and a 1.55 cosine axial power shape.

Figure 2.1-1
Reactor Core Safety Limit – Four Loops in Operation



2.2 SAFETY LIMIT: REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System pressure.

Objective

To maintain the integrity of the Reactor Coolant System and to prevent the release of excessive amounts of fission product activity to the containment.

Specification

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis

The Reactor Coolant System [1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.

The settings of the power operated relief valves (2335 psig)[2] and the reactor high pressure trip (2385 psig)[2] have been established to assure that the Reactor Coolant System pressure limit is never reached and that the system pressure does not exceed the design limits of the fuel cladding.

In addition, the Reactor Coolant System safety valves[3] are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, all components in the system are hydrotested at 3110 psig prior to initial operation[4].

References

(1) UFSAR Chapter 4

(2) UFSAR Table 4.1-1

(3) UFSAR Section 4.3.4

(4) UFSAR Section 4.3.3

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, and pressurizer level.

Objective

To provide for automatic protective action such that the principal process variables do not exceed a safety limit.

Specifications

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, power range (low setpoint): $\leq$ 25% of rated power.

   B. Core limit protection

      (1) High flux, power range (high setpoint): $\leq$ 109% of rated power.

      (2) High pressurizer pressure: $\leq$ 2385 psig.

      (3) Low pressurizer pressure: $\geq$ 1870 psig.

      (4) Overtemperature ΔT:

      $$\Delta T \leq \Delta T_o \left[K_1 - K_2 (T - T') + K_3 (P - P') - f(\Delta I)\right]$$

      where:

      ΔT = Measured ΔT by hot and cold leg RTDs, ^{o}F

      ΔT_o $\leq$ Indicated ΔT at rated power, ^{o}F

      T = Average temperature, ^{o}F

      T' = Design full power T_{avg} at rated power, $\leq 579.7^{o}F$

P = Pressurizer pressure, psig

P' = 2235 psig

$K_1 \leq 1.25$

$K_2 = 0.022$

$K_3 = 0.00095$

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -36% and +7%, $f(\Delta I) = 0$, where q_t and q_b are percent rated power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total power in percent of rated power;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -36%, the ΔT trip setpoint shall be automatically reduced by 2.14% of its value at rated power; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT trip setpoint shall be automatically reduced by 2.15% of its value at rated power.

(5) Overpower ΔT:

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT}{dt} - K_6 (T - T'')\right]$$

where:

ΔT = Measured ΔT by hot and cold leg RTDs, ^oF

$\Delta T_o \leq$ Indicated ΔT at rated power, ^oF

T = Average temperature, ^oF

C. The anticipatory reactor trip upon turbine trip shall be unblocked when the power range nuclear instrumentation indicates $\geq$ 35% of rated power.

3. The Control Rod Protection System shall open the reactor trip breakers during RCS cooldown prior to T_{cold} decreasing below 350°F.

Basis

The high flux reactor trips provide redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis[(1)].

The power range nuclear flux reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis[(2,3)].

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (UFSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis[(4)].

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detector (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by a difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced[6,7].

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors[2].

The low flow reactor trip protects the core against DNB in the event of a loss of one or two reactor coolant pumps. The undervoltage reactor trip protects the core against DNB in the event of a loss of two or more reactor coolant pumps. The setpoints specified are consistent with the values used in the accident analysis[8]. The low frequency reactor coolant pump trip also protects against a decrease in flow. The specified setpoint assures a reactor trip signal by opening the reactor coolant pump breaker before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1600 ft^3 of water (39.75 ft above the lower instrument tap) corresponds to 92% of span. The specified setpoint allows margin for instrument error and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against postulated loss of feedwater accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System[9].

Specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point at which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 60% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below the safety limit DNBRs during normal operational transients.

A turbine trip causes a direct reactor trip, when operating at or above 35% power, in order to reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of this trip. Functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

The steam-feedwater flow mismatch trip does not appear in the specification as this setting is not used in the transient and accident analysis (UFSAR Section 14).

To avoid mechanical interference due to thermal contraction between the fuel and the control rods, an automatic backup to manual tripping of the control rods is provided. Prior to T_{cold} decreasing below 350^{o}F during RCS cooldown, the Control Rod Protection System will open the reactor trip breakers which unlatches the control rod drive shafts from the CRDMs.

References

(1) UFSAR 14.1.1

(2) UFSAR 14.1.2

(3) UFSAR Table 7.4.2

(4) UFSAR 14.3.1

(5) UFSAR 14.1.2

(6) UFSAR 7.2

(7) UFSAR 3.2.1

(8) UFSAR 14.1.6

(9) UFSAR 14.1.9

3.0 LIMITING CONDITIONS FOR OPERATION

3.0.1 In the event a Limiting Condition for Operation (LCO) and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within the next 7 hours, and in at least cold shutdown within the following 30 hours unless corrective measures are completed that restore compliance to the LCO within these time intervals as measured from initial discovery or until the reactor is placed in a condition in which the LCO is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

3.0.2 A system, subsystem, train, component or device shall not be considered inoperable solely because its normal power source is inoperable, or solely because its emergency power source (i.e., diesel, battery) is inoperable. In such instances the equipment served by the inoperable power source shall be considered operable for purposes of compliance with their individual equipment LCOs and only the LCO for the inoperable power source shall apply.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

A. OPERATIONAL COMPONENTS

1. Coolant Pump

a. Except as noted in 3.1.A.1.b below, four reactor coolant pumps shall be in operation during power operation.

b. During power operation, one reactor coolant pump may be out of service for testing or repair purposes for a period not to exceed four hours.

c. During shutdown conditions with fuel in the reactor, the operability requirements for reactor coolant and/or residual heat removal pumps specified in Table 3.1.A-1 shall be met.

d. When RCS temperature is less than or equal to 295°F, the requirements of Specification 3.1.A.4 regarding startup of a reactor coolant pump with no other reactor coolant pumps operating shall be adhered to.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor coolant system is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange shall be provided to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

5. Power Operated Relief Valves (PORVs)/Block Valves (for operation above 350°F)

a. Whenever the reactor coolant system is above 350°F, the PORVs and their associated block valves shall be operable with the block valves either open or closed.

b. If a PORV becomes inoperable when above 350°F, its associated block valve shall be maintained in the closed position.

c. If a PORV block valve becomes inoperable when above 350°F, the block valve shall be closed and deenergized.

d. If the requirements of Specification 3.1.A.5.a, 3.1.A.5.b or 3.1.A.5.c above cannot be satisfied, compliance shall be established within four (4) hours, or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled below 350°F.

e. With regard to the use of the PORVs/Block Valves as a reactor coolant system vent, the requirements of Specification 3.16 shall be adhered to.

6. Pressurizer Heaters

a. Whenever the reactor coolant system is above 350°F, the pressurizer shall be operable with at least 150kW of pressurizer heaters.

b. If the requirements of Specification 3.1.A.6.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled to below 350°F.

Basis

When the boron concentration of the Reactor Coolant System (RCS) is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. The requirement for at least one reactor coolant pump or one residual heat removal pump to be in operation is to provide flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. Below 350°F, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant system.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only(1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

The specification that all reactor coolant pumps be operational during power operation is to assure that adequate core cooling will be provided. This flow will keep the minimum departure from nucleate boiling ratio above the safety limit DNBRs; therefore, cladding damage and release of fission products will not occur.

The Overpressure Protection System (OPS) is designed to relieve the RCS pressure for certain unlikely overpressure transients to prevent these incidents from causing the peak RCS pressure from exceeding 10 CFR 50, Appendix G limits. When the OPS is "armed," MOVs 535 and 536 are in the open position, and the PORVs will open upon receipt of the appropriate signal. This OPS arming can be accomplished either automatically by the OPS when the RCS is below a prescribed temperature or manually by the operator.

The OPS will be set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G limits for the following events:

1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature hotter than the RCS water temperature.

2. Letdown isolation with three charging pumps operating.

3. Startup of one safety injection pump.

4. Loss of residual heat removal causing pressure rise from heat additions from core decay heat or reactor coolant pump heat.

5. Inadvertant activation of the pressurizer heaters.

Consideration of the above events provides bounding PORV setpoints for other potential overpressure conditions caused by heat or mass additions at low temperature.

The RCS is protected against overpressure transients when RCS temperature is less than or equal to 295°F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the PORVs or the gas space in the pressurizer, (2) providing administrative controls on starting of a reactor coolant pump when the primary water temperature is less than the secondary water temperature, or (3) providing vent area from the RCS to containment for those situations where neither the PORVs nor the available pressurizer gas space are sufficient to preclude the pressure resulting from postulated transients from exceeding the limits of 10 CFR 50, Appendix G.

The restrictions on starting a reactor coolant pump with the secondary side water temperature higher than the primary side will prevent RCS overpressurizations from the resultant volumetric swell into the pressurizer that is caused by potential heat additions from the startup of a reactor coolant pump without any other reactor coolant pumps operating. When pressurizer level is between 30 and 85% of span, protection is provided through the use of the PORVs. When pressurizer level is less than 30% of span, additional restrictions on pressurizer pressure make reliance on the PORVs unnecessary since the gas compression resulting from the insurge of liquid from the RCS pump start is insufficient to cause RCS pressure to exceed the Appendix G limits. The same method, i.e., control of pressurizer pressure and level, is used to accommodate the mass insurge into the pressurizer from safety injection and charging pump starts when the PORVs are not operational.

An additional restriction is put on the reactor coolant pump start when the secondary system water temperature is less than or equal to 40°F higher than the primary system water temperature and the pressurizer level is greater than 30%. This restriction is to prohibit starting the first reactor coolant pump when the RCS temperature is between 267°F and 295°F. The purpose of the restriction is to assure that the temperature rise resulting from the transient will not be outside the temperature limits for OPS actuation.

When comparison to the Appendix G limits is made, the comparison is to the isothermal Appendix G curve. Other than the delay time associated with opening the PORVs, and the error caused by non-uniform RCS metal and water temperatures during heat addition transients, the analysis does not make any allowance for instrument error. Instrument error will be taken into account when the OPS is set; i.e., the instrumentation will be set so that the PORVs will open at less than the required setpoint, including allowance for instrument errors.

The determination of reactor coolant temperature may be made from the Control Room instrumentation. The determination of the steam generator water temperature may be made in the following ways:

(a) assuming that the secondary side water temperature is at the saturation temperature corresponding to the secondary side steam pressure indicated on the Control Room instrumentation, or

(b) conservatively assuming that the secondary side water temperature is at the reactor coolant temperature at which the last RCP was stopped during cooldown, or

(c) actual or inferred measurement of the secondary side steam generator water temperature at those times it can be measured (such as return from a refueling outage).

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of the saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperatures and pressure[(2)].

If no residual heat were removed by the Residual Heat Removal System, the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load[(3)] without a direct trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove decay heat after a reactor shutdown.

All pressurizer heaters are supplied electrical power from an emergency bus. The requirement that 150kW of pressurizer heaters and their associated controls be operable when the reactor coolant system is above 350°F provides assurance that these heaters will be available and can be energized during a loss of offsite power condition to assist in maintaining natural circulation at hot shutdown.

The power-operated relief valves (PORVs) can operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to provide a relief path when desirable and to ensure the ability to seal off possible RCS leakage paths. Both the PORVs and the PORV block valves are subject to periodic valve testing for operability in accordance with the ASME Code Section XI as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program.

<u>Reference</u>

(1) UFSAR Section 14.1.12
(2) UFSAR Section 9.3.1
(3) UFSAR Section 14.1.8

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing

(1) Reactor Condition	(2) Required No. of Pumps Operating	(3) Required No. of Pumps Operable (including operating pump)	(4) Action Required if Condition of Column (2) or (3) is not met
Hot shutdown T_{avg} > 350°F (Excluding loss of offsite power)	Two RCPs	Two RCPs	With less than two reactor coolant pumps operating, maintain the reactor trip breakers open. With no reactor coolant pumps operating, T_{avg} may be maintained above 350°F for up to one hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system, and (2) RCS temperature is maintained at least 10°F below saturation temperature. If a RCP has not been restored to operating status within the one hour permitted, take action as listed below for no operable pumps. With only one RCP operable, restore a second RCP to operable status within 72 hours or bring the RCS temperature to 350°F. Except for testing, with no RCPs operable, immediately initiate action to bring RCS temperature to 350°F.

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing

(1) Reactor Condition	(2) Required No. of Pumps Operating	(3) Required No. of Pumps Operable (including operating pump)	(4) Action Required if Condition of Column (2) or (3) is not met
Hot shutdown $T_{avg} \leq 350^{o}F$	One RCP or one RHR pump	Two RCPs or two RHR pumps or one RCP and one RHR pump	The requirement to have at least one RCP or RHR pump in operation may be suspended for up to one hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system, and (2) RCS temperature is maintained at least $10^{o}F$ below saturation temperature. If a pump has not been restored to operating status within the one hour permitted, take action as listed below for no operable pumps. With only one pump (RHR or RCP) operable, either restore a second pump to operable status or be in cold shutdown within 20 hours. With no pumps operable, suspend all operations involving a reduction in boron concentration and immediately initiate action to restore at least one pump to operable status.

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing

(1) Reactor Condition	(2) Required No. of Pumps Operating	(3) Required No. of Pumps Operable (including operating pump)	(4) Action Required if Condition of Column (2) or (3) is not met
Cold shutdown	One RCP or one RHR pump	Two RCPs or two RHR pumps or one RCP and one RHR pump	The requirement to have at least one reactor coolant pump or RHR pump in operation may be suspended for up to one hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system, and (2) RCS temperature is maintained at least 10°F below saturation temperature. With only one pump operable, stay in cold shutdown until a second pump is restored to operable status. The requirements of columns (2) and/or (3) may be suspended during maintenance, modifications, testing, inspection or repair. During operation under this provision, the following shall apply:

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing

(1) Reactor Condition	(2) Required No. of Pumps Operating	(3) Required No. of Pumps Operable (including operating pump)	(4) Action Required if Condition of Column (2) or (3) is not met
Cold shutdown (cont'd)			(1) An alternate means of decay heat removal shall be available and return of the system within sufficient time to prevent exceeding cold shutdown requirements shall be assured. (2) RCS temperature and the source range detectors shall be monitored hourly. (3) No operations are permitted that would cause dilution of the reactor coolant system.
Refueling	See Specification 3.8	See Specification 3.8	See Specification 3.8

Table 3.1.A-2

OPS Operability Requirements

Reactor Coolant Pumps

With OPS operable at or below 295°F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

(1) The temperature of all steam generators is less than or equal to the RCS temperature, or

(2) The temperature of all steam generators is less than or equal to 40°F higher than the RCS temperature and:

- o RCS temperature is less than or equal to 267°F,
- o Pressurizer level is between 30 - 85% of span; or

(3) The temperature of all steam generators is less than or equal to 100°F higher than RCS temperature and:

- o RCS pressure is less than or equal to 450 psig,
- o RCS temperature is greater than or equal to 145°F,
- o Pressurizer level is less than or equal to 30% of span.

With OPS inoperable at or below 295°F, a reactor coolant pump can be started (or jogged) with no other reactor coolant pumps operating if:

(1) The temperature of all steam generators is less than or equal to the RCS temperature, or

(2) The temperature of all steam generators is less than or equal to 100°F higher than RCS temperature and:

- o RCS pressure is less than or equal to 450 psig,
- o RCS temperature is greater than or equal to 145°F,
- o Pressurizer level is less than or equal to 30% of span.

Table 3.1.A-2

OPS Operability Requirements

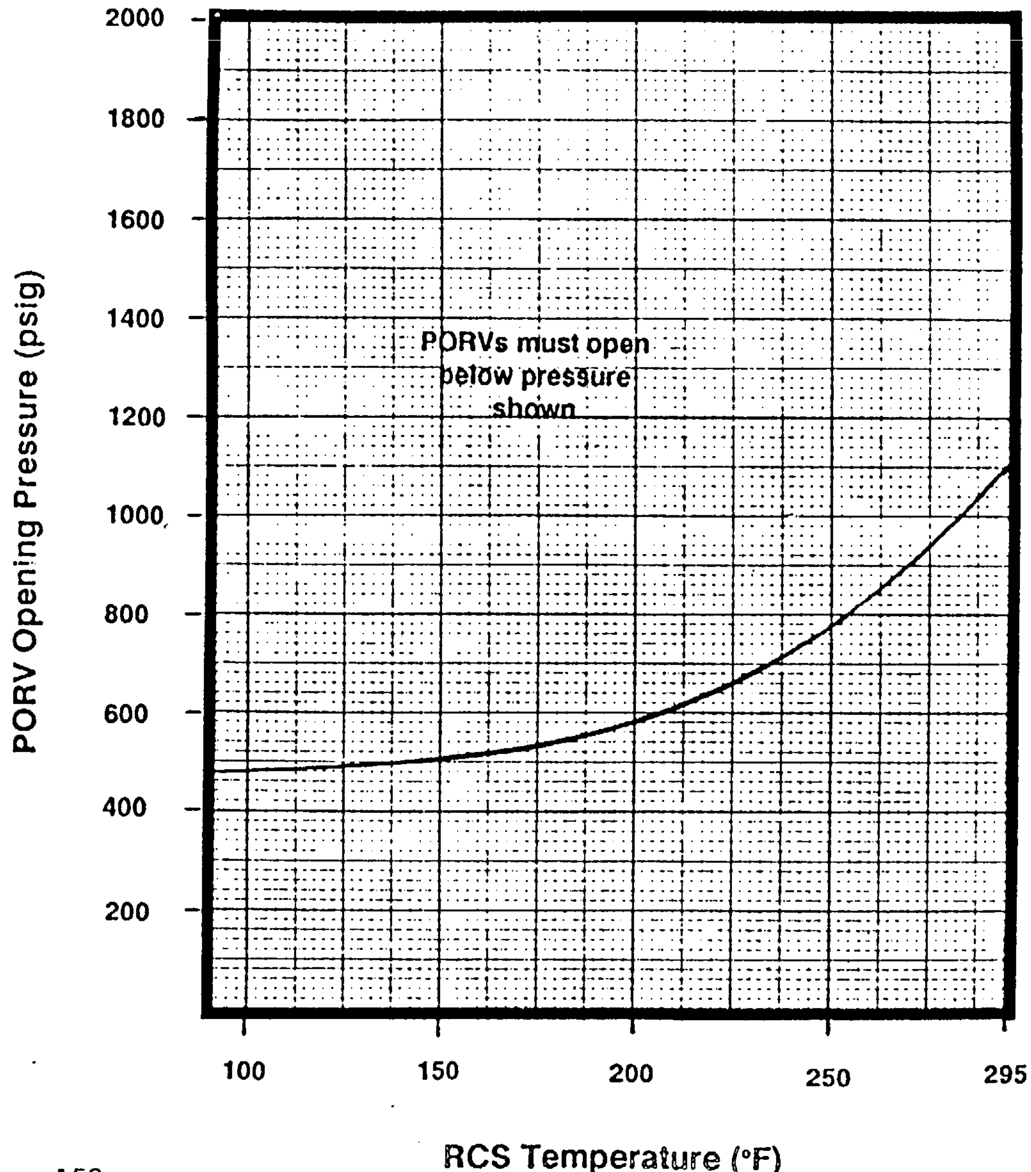
Safety Injection and Charging Pumps

With OPS operable at or below 295^{o}F, no more than one (1) safety injection (SI) and three (3) charging pumps may be energized.

OPS is not required to be operable at or below 295^{o}F if either the conditions of Column I or the conditions of Column II below are met for the specified conditions:

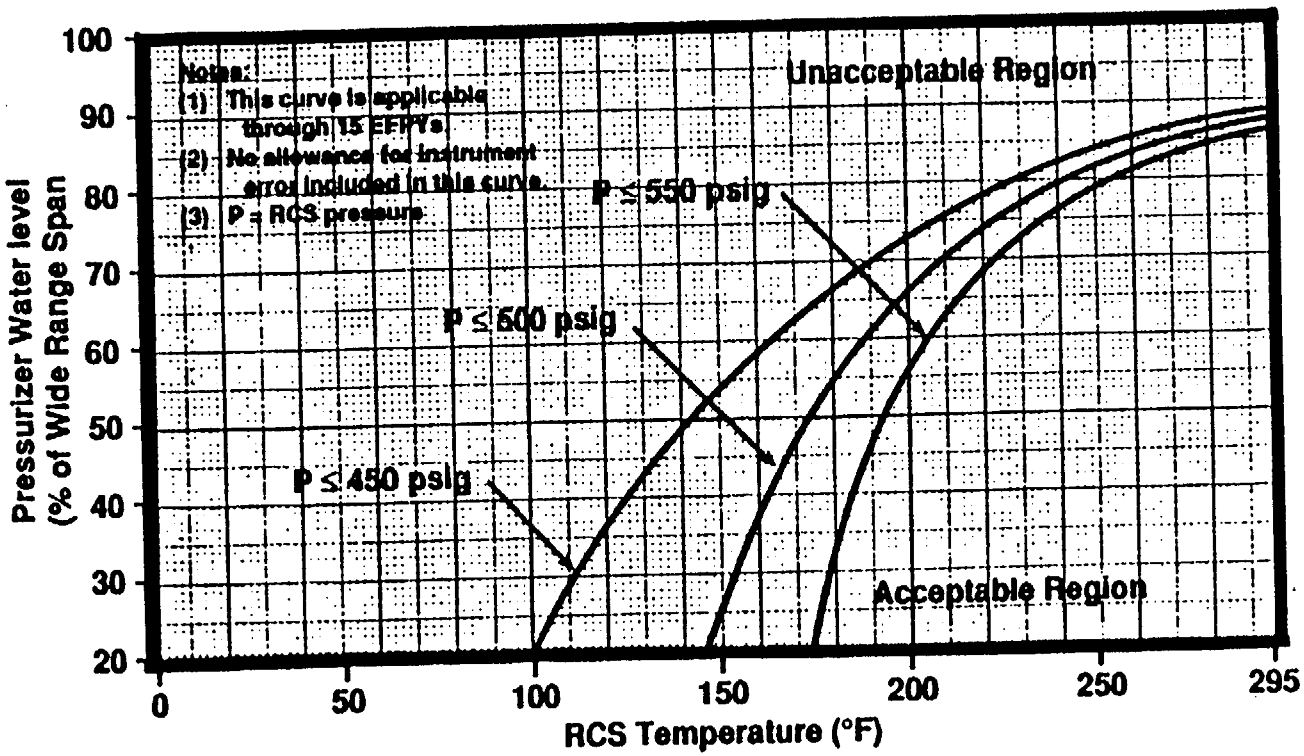
Maximum Number of Energized Pumps (SI and/or charging)		I Operating Restrictions (pressurizer pressure, pressurizer level, and RCS temperature)	II Vent Area to Containment Atmosphere (square inches)
SI	Charging		
0	1	See Figure 3.1.A-2	2.00
1	3	See Figure 3.1.A-3	2.00
3	3	---------	5.00

FIGURE 3.1.A-1
PORV OPENING PRESSURE FOR OPERATION
LESS THAN EQUAL TO 295°F



Notes:
1. Applicable through 15 EFPYs
2. No allowance for instrument error included on this curve.

Figure 3.1.A-2
Maximum Pressurizer Level with PORVs Inoperable and One Charging Pump Energized



B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to 15 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those present may be obtained by interpolation.

   b. Figure 3.1.B-1 and Figure 3.1.B-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1.B-1 and Figure 3.1.B-2 shall be recalculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323[7] and as specified in Specification 3.1.B.3 below. The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than six months prior to scheduled specimen removal.

3. The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and

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* Refer to UFSAR Section 4.5, WCAP-7323, and Indian Point Unit No. 2, "Application for Amendment to Operating License," sworn to on February 3, 1981.

post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens. The specimens will be removed and examined at the following intervals:

Capsule 1	End of Cycle 1 operation
Capsule 2	End of Cycle 2 operation
Capsule 3	End of Cycle 5 operation
Capsule 4	End of Cycle 8 operation
Capsule 5	End of Cycle 16 operation
Capsule 6	Spare

4. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70^{o}F.

5. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100^{o}F/hr and 200^{o}F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320^{o}F.

6. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

Basis

Fracture Toughness Properties

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes[(1)]. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the UFSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100^{o}F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation[(2)].

The reactor vessel plate opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a Nil-Ductility Transition Temperature (NDTT) of 40°F or less. The material has been tested to verify conformity to specified requirements and a NDTT value of 20°F has been determined. In addition, this plate has been 100 percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other Reactor Coolant System components, meet the appropriate design code requirements and specific component function[3].

As a result of fast neutron irradiation in the region of the core, there will be an increase in the Reference Nil-Ductility Transition Temperature (RT_{NDT}) with nuclear operation. The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample location are described in Appendix 4A of the UFSAR. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

An approximation of the maximum integrated fast neutron (E > 1 Mev) exposure is given in Figure 2-4 of WCAP-7924A[4]. Exposure of the Indian Point Unit No. 2 vessel will be less than that indicated in this figure.

The actual shift in RT_{NDT} will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. These samples are evaluated according to ASTM E185[6]. To compensate for any increase in the RT_{NDT} caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4].

The first reactor vessel material surveillance capsule was removed during the 1976 refueling outage. That capsule was tested by Southwest Research Institute (SWRI) and the results were evaluated and reported[8,9]. The second surveillance capsule was removed during the 1978 refueling outage. That capsule has been tested by SWRI and the results have been evaluated and reported[10]. The third vessel material surveillance capsule was removed during the 1982 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported[11]. Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed for up to fifteen (15) effective full power years (EFPYs) of reactor operation.

The maximum shift in RT_{NDT} after 15 EFPYs of operation is projected to be 142°F at the 1/4 T and 71°F at the 3/4 T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for the IP2 reactor vessel was 34°F as described in Table 3.1.B-1. The heatup and cooldown curves for 15 EFPYs have been computed on the basis of the RT_{NDT} of Plate B2002-3 because it is anticipated that the RT_{NDT} of the reactor vessel beltline material will be highest for Plate B2002-3, at least through that time period[11].

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G in Section III 1974 Edition of the ASME Boiler and Pressure Vessel Code and are discussed in detail in WCAP-7924A[4].

The approach specifies that the allowable total stress intensity factor (K_I), at any time during heatup or cooldown, cannot be greater than that shown on the K_{IR} curve[5] for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \qquad (1)$$

where:

K_{Im} is the stress intensity factor caused by membrane (pressure) stress,

K_{It} is the stress intensity factor caused by the thermal gradients,

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state condition (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature, and thus tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1.B-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in WCAP-7924A[4].

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 UFSAR, Section 4.1.5.

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 2 UFSAR, Section 4.2.5.

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, S.L. Anderson, S. E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G.

(6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.

(7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, May 1969.

Table 3.1.B-1

Indian Point Unit No. 2

Reactor Vessel Core Region Material

Plate	Copper Content	Initial RT_{NDT}
B 2002-1	0.25	$34^{o}F$
B 2002-2	0.14	$21^{o}F$
B 2002-3*	0.14	$21^{o}F$
HAZ	--	$0^{o}F$
Weld Material	--	$0^{o}F$

References:

(1) Letter No. IPP-75-50, Westinghouse to Con Edison dated May 16, 1975.

(2) Letter dated March 29, 1978 from W. J. Cahill, Jr. (Consolidated Edison) to R. W. Reid (NRC), "Indian Point Unit No. 2 Reactor Vessel Material Surveillance Program."

(3) Final Report - SWRI Project No. 06-7379 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z", E. B. Norris, April 1984.

Notes:

* Based on Reference (3) above, the bounding values for copper (0.25%) and initial RT_{NDT} ($34^{o}F$) are applied to the controlling plate (B2002-3) for the purpose of generating the heatup and cooldown limitations.

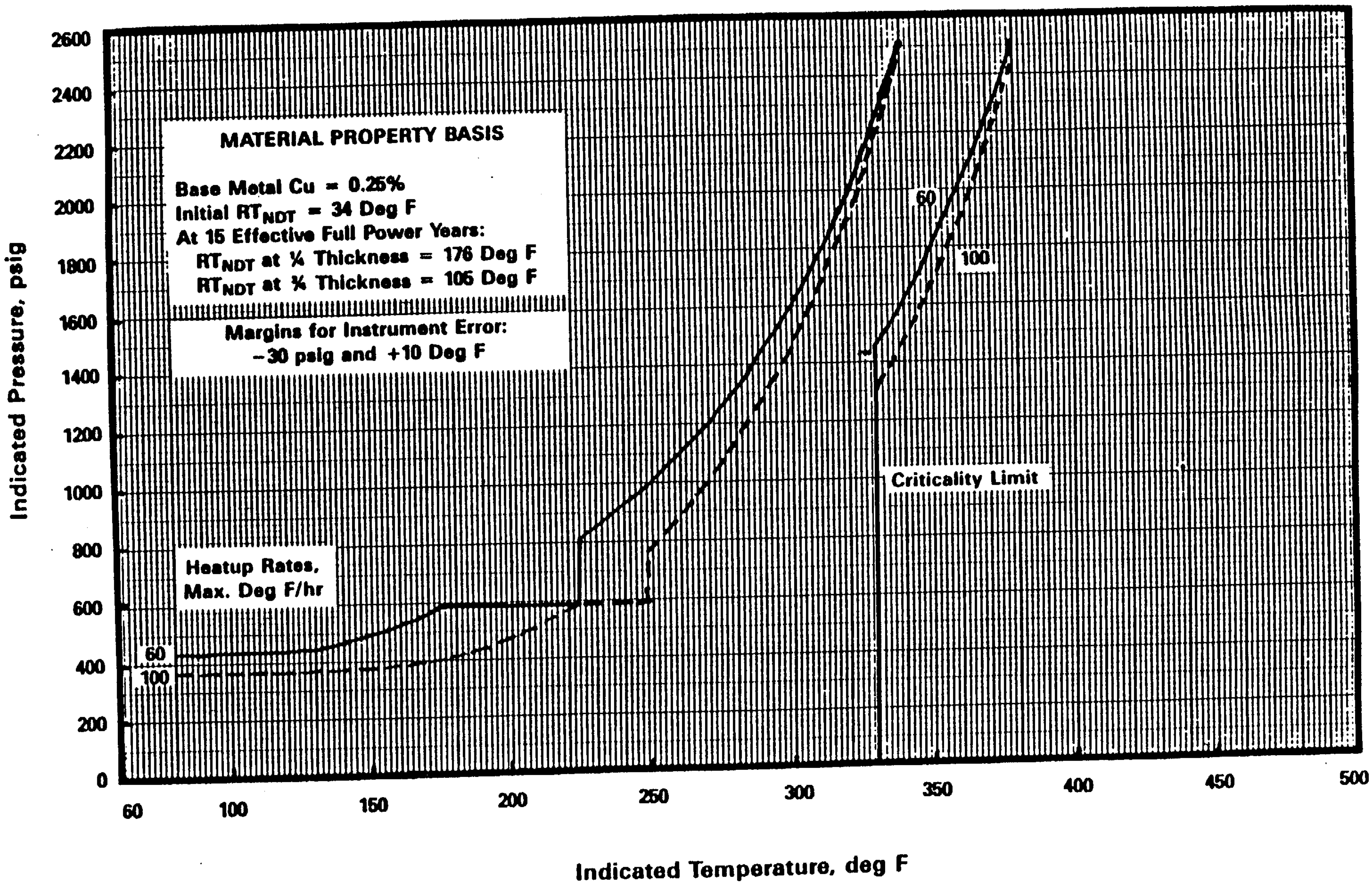

FIGURE 3.1.B-1.
INDIAN POINT UNIT NO. 2 COOLANT HEATUP LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

C. MINIMUM CONDITIONS FOR CRITICALITY

Specifications

1. Except during low-power physics tests, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

2. In no case shall the reactor be made critical below the temperature and pressure limits shown in Figure 3.1.B-1.

3. When the reactor coolant temperature is below the minimum temperature specified in (1) above, the reactor shall be subcritical by an amount greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range[(1)]. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or negative. At all times, the moderator coefficient is negative in the power operating range[(1)]. Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during lower power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical below the temperature and pressure limits shown in Figure 3.1.B-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization in accordance with the requirements of 10 CFR 50 Appendix G, as amended February 2, 1976. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin specified in 3.1.C.3 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

(1) UFSAR Section 3.2

D. MAXIMUM REACTOR COOLANT ACTIVITY*

Specification

1. The total specific activity of the reactor coolant, excluding tritium, due to nuclides with half-lives of more than 30 minutes, shall not exceed $60/\bar{E}$ μCi/cc, whenever the reactor is critical or the average reactor coolant temperature is greater than 500°F. ($\bar{E}$ is the weighted average of the beta and gamma energies per disintegration in Mev.)

Basis

The specified limit provides protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the following analysis of a steam generator tube rupture accident. [1]

Rupture of a steam generator tube would allow a portion of the reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases which are diverted to the containment within a few seconds after the air ejector monitors high activity signal. The activity release to atmosphere is not significant.

In the event the air ejector discharge is not diverted to the containment, a portion of the reactor coolant noble gas activity would be released to the atmosphere through the secondary system. Activity could continue to be released until the operator would reduce the primary system pressure below the lowest setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, with the air ejector discharging to the atmosphere, followed by isolation of the faulty steam generator by the operator within 30 minutes after the event. During that time approximately one-eighth of the total reactor coolant could be released to the Steam and Feedwater System.

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* See Specification 3.4 for activity limits on the secondary side.

The limiting offsite dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis will employ the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$\text{Dose (rem)} = 1/2\ [\bar{E} \cdot A \cdot V \cdot X/Q\ (3.7 \times 10^{10}) \cdot (1.33 \times 10^{-11})]$$

Where: $\bar{E}$ = weighted average energy of betas and gammas per disintegration (Mev/dis),

A = primary coolant activity (micro Ci/cc),

V = primary coolant volume released to the secondary side (44.5 m^3),

$X/Q = 7.5 \times 10^{-4}$ sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary,

3.7×10^{10} dis/sec - Ci, and

1.33×10^{-11} rem/Mev/m^3.

The resulting dose is 0.5 rem at the site boundary when A is equal to $60/\bar{E}$, which is the expression used in this specification.

If the air ejector discharge is diverted to the containment, the only activity released to atmosphere is that contained in the steam flow to the turbine gland seal (5000 lb/hr). For this case the activity release to atmosphere during the 30-minute period would be 1.1% of the values given above. It is concluded that a tube rupture accident would not result in significant radiation exposure.

The basis for the 500°F temperature contained in the specification is that saturation pressure corresponding to 500°F, 680.8 psia is well below the pressure at which the atmospheric relief valves on the secondary side would be actuated.

Calculations required to determine $\bar{E}$ will consist of the following:

1. Quantitative measurement in units of µCi/cc of radionuclides with half-lives longer than 30 minutes making up at least 95% of the total activity in the primary coolant.

2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes. (Table of Isotopes, Sixth Edition, March 1968).

3. A calculation of $\bar{E}$ by appropriate weighting of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

References

(1) UFSAR Section 14.2.4

E. MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Specifications

1. Concentration of contaminants in the reactor shall not exceed the following limits when the reactor coolant is above 250°F:

Contaminant	Normal Steady-State Operation (PPM)	Transients not to Exceed 24 Hours (PPM)
a. Oxygen	0.10	1.00
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in 3.1.E.1 above are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the limits of Specification 3.1.E.1 above, the reactor shall be brought to the cold shutdown condition, utilizing normal operating procedures, and the cause of the out-of-specification operation ascertained and corrected. The reactor may then be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise, a safety review is required before startup.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is below 250°F:

Contaminant	Normal Concentration (PPM)	Transient not to Exceed 48 Hours (PPM)
a. Oxygen	Saturated	Saturated
b. Chloride	0.15	1.5
c. Fluoride	0.15	1.5

If the limits above are exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition ascertained and corrected.

5. For the purposes of correcting the contaminant concentrations to meet Specifications 3.1.E.1 and 3.1.E.4 above, increase in coolant temperature consistent with operation of reactor coolant pumps for a short period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250°F.

Basis

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in 3.1.E.1 and 3.1.E.4, the integrity of the reactor coolant system is assured under all operating conditions.

If these limits are exceeded, measures can be taken to correct the condition (e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[1]); because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus the period of 24 hours for corrective action to restore concentrations within the limits has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the cold shutdown condition and the corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit a transient concentration to exist for a longer period of time and still provide the assurance that the integrity of the primary coolant system will be maintained.

In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

References

(1) UFSAR Section 9.2

F. REACTOR COOLANT SYSTEM LEAKAGE AND LEAKAGE INTO THE CONTAINMENT FREE VOLUME

Specifications

1. Leakage Detection And Removal Systems

a. The reactor shall not be brought above cold shutdown unless the following leakage detection and removal systems are operable:

(1) two containment sump pumps,

(2) two containment sump level monitors,

(3) a containment sump discharge line flow monitoring system,

(4) two recirculation sump level monitors,

(5) two reactor cavity level monitors,

(6) two of the following three systems:

(a) a containment atmosphere gaseous radioactivity monitoring system,

(b) a containment atmosphere particulate radioactivity monitoring system,

(c) the containment fan cooler condensate flow monitoring system.

b. When the reactor is above cold shutdown, the requirements of Specification 3.1.F.1.a may be modified as follows:

(1) One containment sump pump may be inoperable for a period not to exceed seven (7) consecutive days provided that, on a daily basis, the other containment sump pump is started and discharge flow is verified.

(2) One of the two required containment sump level monitors may be inoperable for a period not to exceed seven (7) consecutive days.

(3) The containment sump discharge line flow monitoring system may be inoperable for a period not to exceed seven (7) consecutive days provided a detailed Waste Holdup Tank water inventory balance is performed daily.

(4) One of the two required recirculation sump level monitors may be inoperable for a period not to exceed fourteen (14) consecutive days.

(5) One of the two required reactor cavity level monitors may be inoperable for a period not to exceed thirty (30) consecutive days.

(6) Two of the three monitoring systems specified in Specification 3.1.F.1.a.(6) may be inoperable for a period not to exceed thirty (30) consecutive days. If either of the radioactivity monitoring systems specified in Specification 3.1.F.1.a.(6) is inoperable, grab samples of the containment atmosphere shall be obtained and analyzed daily.

c. If the conditions of Specification 3.1.F.1.b cannot be met or an inoperable system(s) is not restored to operable status within the time period(s) specified therein, then either perform a visual inspection of containment once a shift, or place the reactor in the hot shutdown condition within the next 6 hours and, if the inoperability continues, place the reactor in the cold shutdown condition within the following 30 hours.

2. Operational Leakage Limits

a. Primary to Secondary Leakage

(1) Primary to secondary leakage through the steam generator tubes shall not exceed 0.3 gpm in any steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours.

(2) If leakage from two or more steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If tube leaks attributable to the tube denting phenomena are observed in two or more steam generators after the reactor is in cold shutdown, Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Whenever the reactor is shut down in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the NRC shall be informed before any tube is plugged or, if no tube is plugged, before the steam generator is returned to service.

b. RCS/RHR Pressure Isolation Valves Leakage

(1) Whenever the reactor is above cold shutdown, leakage through each of the RCS/RHR pressure isolation valves 897A, B, C and D, and 838A, B, C and D shall satisfy the following acceptance criteria:

(a) Leakage rates of less than or equal to 1.0 gpm are acceptable.

(b) Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

(c) Leakage rates greater than 5.0 gpm are unacceptable.

(2) If any RCS/RHR pressure isolation valve listed in Specification 3.1.F.2.b.(1) is determined to be inoperable based on the acceptance criteria presented therein, an orderly plant shutdown shall be initiated and the reactor shall be placed in the cold shutdown condition within 24 hours.

c. <u>Total Reactor Coolant System Leakage</u>

(1) Whenever the reactor is above cold shutdown, reactor coolant system leakage shall be limited to:

(a) No pressure boundary leakage,

(b) 1 gpm unidentified leakage, and

(c) 10 gpm identified leakage.

(2) With any pressure boundary leakage, the reactor must be placed in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

(3) If the Reactor Coolant System leakage exceeds the limits in either c.(1)(b) or c.(1)(c) above, the leakage rate must be reduced to within limits within 4 hours or the reactor must be placed in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. Leakage Into The Containment Free Volume

(1) Whenever the reactor is above cold shutdown, the total leakage into the containment free volume from both reactor coolant and non-reactor coolant sources combined shall not exceed 10 gpm.

(2) Notwithstanding the action which may be required by Specification 3.1.F.2.d.(3) below, with the combined leakage into the containment free volume greater than the above limit, the leakage rate must be reduced to within the specified limit within 12 hours or the reactor must be placed in cold shutdown within the following 36 hours.

(3) If water level in the containment sump reaches EL. 45′, or the water level in the recirculation sump reaches EL. 35′, or the water level in the reactor cavity reaches EL. 20′, the reactor shall be placed in a cold shutdown condition within the next 36 hours unless the water level(s) is reduced below the specified limit(s).

(4) If the water level in the containment sump increases above EL. 45′ and the water level in the recirculation sump increases above EL. 39′ 9", or the water level in the reactor cavity increases above EL. 20′ 5", immediately place the reactor in a subcritical condition and initiate an expeditious cooldown of the reactor to the cold shutdown condition.

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any pressure boundary of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are are within the recommended sensitivity guidelines of Regulatory Guide 1.45.

b. The containment radiogas monitor.

c. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units including leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows from approximately 1 gpm to 15 gpm per detector can be measured by this system. Condensate flows greater than 15 gpm can be determined using weir calibration curves. Condensate flows less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.

d. Leakage detection via the containment sump level and discharge flow monitoring systems will determine leakage losses from all fluid systems to the containment free volume. Water collecting on the containment floor will normally be delivered to the containment sump via the containment floor trench system. Level monitoring of the containment sump is in part provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the containment sump. In addition, another level monitoring device provides a continuous level readout in the control room. When the water level in the containment sump reaches predetermined levels, one or both containment sump pumps will automatically start and pump the fluid out of containment to the liquid waste disposal system. Flow in the containment sump pump discharge line from containment to the Waste Holdup Tank is monitored on a continuous basis. Thus, monitoring of both flow indication systems will provide a positive means for determining leakage into the containment free volume.

e. Water may also collect in the recirculation sump and/or the reactor cavity depending on the size and location of the leak. However, under most circumstances, the containment sump will be filled prior to the recirculation sump filling and both sumps will be filled prior to water level increasing on the containment floor (EL. 46′) sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. In addition, another level monitoring device provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a single analog continuous level indication in the control room and by two separate and independent level switches, each of which actuates an audible alarm in the control room.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances. Determined leakage rates are an average over the applicable surveillance interval. Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure detection of additional leakage.

The 10 gpm identified leakage limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage by the leakage detection systems.

Pressure boundary leakage of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any pressure boundary leakage requires the unit to be promptly placed in cold shutdown. Primary system leakage through packing, gaskets, seal welds or mechanical joints is not considered to be pressure boundary leakage.

The leakage limit and surveillance testing for RCS/RHR Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS/RHR Pressure Isolation Valves is identified leakage and will be considered as a portion of the allowed limit.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by limitation of steam generator leakage between the reactor coolant system and the secondary coolant system. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown and the leaking tube(s) will be located and plugged.

The 10 gpm limit for combined reactor coolant and non-reactor coolant leakage into the containment free volume provides allowance for a limited amount of leakage from sources other than the reactor coolant system within containment while conservatively limiting total leakage into the containment free volume to the same limit (i.e., 10 gpm) for identified reactor coolant leakage alone. This leakage is within the capabilities of the leakage detection and waste processing system and will not interfere with the detection of independent unidentified reactor coolant system leakage.

For those circumstances where high energy line failures occur inside containment resulting in flooding of the containment building sumps and/or floor, automatic actuation of reactor protection, safety injection and/or containment spray systems places the plant in a safe condition and, in some cases, provides intended flooding of the containment building. However, for those circumstances resulting from leakage or failure of low energy systems such as service water or component cooling inside containment, operator action is necessary to prevent accumulation of water on the containment floor to undesirable levels.

If the water level in the containment sump reaches EL. 45′, or the water level in the recirculation sump reaches EL. 35′, or the water level in the reactor cavity reaches EL. 20′, the reactor is placed in cold shutdown within the next 36 hours. If the water level in the containment sump increases above EL. 45′ and the water level in the recirculation sump increases above EL. 39′ 9", or the water level in the reactor cavity increases above EL. 20′ 5", the operator will immediately bring the reactor subcritical and initiate an expeditious cooldown of the plant.

The above actions are necessary to (1) preclude accumulation of water inside containment so that if a LOCA were to occur safety-related equipment would not become submerged, (2) prevent the reactor cavity from becoming filled with water, (3) prevent the reactor vessel from being wetted while it is at an elevated temperature, and (4) prevent the immersion of the in-core instrument conduits. The amount of water estimated to be inside containment after actuation of the emergency core cooling system following a loss of coolant accident is approximately 423,000 gallons. This amount of water would, by itself, reach approximately EL. 50′ 1". An additional 28,000 gallons (a total of approximately 451,000 gallons) would have to accumulate inside containment before any safety-related electrical component would be submerged (approximately EL. 50′ 5"). The combined volume of the containment sump, the recirculation sump and the containment floor trenches is approximately 18,000 gallons. Since operator action is required by these specifications to shut the reactor down before these volumes are filled, sufficient margin between the water level inside containment following a loss of coolant accident and the level at which a safety-related electrical component may become submerged is maintained. Furthermore, since both sumps, the floor trenches and the containment floor up to EL. 46′ 5 3/8" must be flooded (i.e., approximately 50,000 gallons) before the water level is sufficiently high to flood over the curb leading to the reactor cavity, the forementioned operator actions taken to preclude excessive flooding plus LOCA water levels will conservatively preclude flooding of the reactor cavity and subsequent wetting of the reactor vessel at an elevated temperature.

References

UFSAR Sections 6.7, 11.2.3 and 14.2.4

G. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW RATE

Specifications

The following DNB related parameters pertain to four loop steady-state operation at power levels greater than 98% of rated full power:

a. Reactor Coolant System $T_{avg} \leq 587.2^{o}F$

b. Pressurizer Pressure $\geq$ 2190 psia

c. Reactor Coolant System Total Flow Rate $\geq$ 331,840 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Basis

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than the safety limit DNBRs.

The limits on reactor coolant system temperature, pressure and loop coolant flow represent those used in the accident analyses and are specified to assure that the values assumed in the accident analyses are not exceeded during steady-state four loop operation. Indicator uncertainties have not been accounted for in determining the DNB parameter limits on temperature and pressure.

Compliance with the specified ranges on reactor coolant system temperature and pressurizer pressure is demonstrated by verifying that the parameters are within their applicable ranges at least once each 12 hours.

Compliance with the specified range on Reactor Coolant System total flow rate is demonstrated by verifying the parameter is within its range after each refueling cycle.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specifications

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met:

1. Two charging pumps shall be operable.

2. The boric acid storage system shall contain a minimum of 6000 gallons of 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F, and at least one boric acid transfer pump shall be operable.

3. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

4. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system.

C. During power operation, the requirements of 3.2.B may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.2.B within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to operable status within 24 hours.

2. The boric acid storage system (including the boric acid transfer pumps) may be inoperable provided the RWST is operable and provided that the boric acid storage system and at least one boric acid transfer pump is restored to operable status within 48 hours.

3. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

4. Both channels of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided at least one channel is restored to operable status within 48 hours, the required flow path is shown to be clear of blockage, and the second channel is restored to operable status within 7 days.

D. When RCS temperature is less than or equal to 295°F, the requirements of Table 3.1.A-2 regarding the number of charging pumps allowed to be energized shall be adhered to.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank.

A third method will be to depressurize and use the safety injection pumps. There are three sources of borated water available for injection through three different paths.

1. The boric acid transfer pumps can deliver the contents of the boric acid storage system to the charging pumps.

2. The charging pumps can take suction from the refueling water storage tank (2000 ppm boron solution). Reference is made to Technical Specification 3.3.A.

3. The safety injection pumps can take their suction from the refueling water storage tank.

The quantity of boric acid in storage from either the boric acid storage system or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

Approximately 5700 gallons of the 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) of boric acid are required to meet cold shutdown conditions.

Thus, a minimum of 6000 gallons in the boric acid storage system is specified. An upper concentration limit of 13% (22,500 ppm of boron) boric acid in the boric acid storage system is specified to maintain solution solubility at the specified low temperature limit of 145^{o}F. One of two channels of heat tracing is sufficient to maintain the specified low temperature limit. Since both channels out of service could result in boron precipitation, it is necessary to show that the required flow path is clear of blockage following operation in this condition.

Reference

UFSAR - Section 9.2

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specifications

The following specifications apply except during low-temperature physics tests.

A. SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

1. The reactor shall not be made critical except for low-temperature physics tests, unless the following conditions are met:

   a. The refueling water storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

   b. Deleted

   c. The four accumulators are pressurized to at least 615 psig and each contains a minimum of 787.5 ft^3 and a maximum of 802.5 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

d. Three safety injection pumps together with their associated piping and valves are operable.

e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

f. Two recirculation pumps together with the associated piping and valves are operable.

g. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

h. Valves 856A, C, D and E, in the discharge header of the safety injection header, are in the open position. Valves 856B and F, in the discharge header of the safety injection header, are in the closed position. The hot-leg valves (856B and F) shall be closed with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

i. The four accumulator isolation valves shall be open with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

j. Valve 1810 on the suction line of the high-head SI pumps and valves 882 and 744, respectively on the suction and discharge line of the residual heat removal pumps, shall be blocked open by de-energizing the valve-motor operators.

k. The refueling water storage tank low-level alarms are operable and set to alarm between 74,200 gallons and 99,000 gallons of water in the tank.

2. During power operation, the requirements of 3.3.A.1 may be modified to allow any one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.A.1

within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.A.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours and the remaining two pumps are demonstrated to be operable.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours and the other residual heat removal pump is demonstrated to be operable.

c. One residual heat removal heat exchanger may be out of service provided that it is restored to operable status within 48 hours.

d. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

e. Deleted

f. One refueling water storage tank low-level alarm may be inoperable for up to 7 days provided the other low-level alarm is operable.

3. When RCS temperature is less than or equal to 295^{o}F, the requirements of Table 3.1.A-2 regarding the number of safety injection (SI) pumps allowed to be energized shall be adhered to.

B. <u>CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS</u>

1. The reactor shall not be made critical unless the following conditions are met:

a. The spray additive tank contains not less than 4000 gallons of solution with a sodium hydroxide concentration of not less than 33% by weight.

b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. During power operation, the requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.B.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One fan cooler unit may be inoperable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are operable.

b. One containment spray pump may be inoperable during normal reactor operation, for a period not to exceed 72 hours, provided the five fan cooler units and the remaining containment spray pump are operable.

c. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to operable status within 7 days or 72 hours for the fan cooler or containment spray systems respectively, and all valves in the system that provide the duplicate function are operable.

d. The spray additive tank and its associated piping, valves and eductors may be inoperable during normal reactor operation for a period not to exceed 72 hours provided both containment spray pumps and the five fan cooler units are operable.

C. ISOLATION VALVE SEAL WATER SYSTEM (IVSWS)

1. The reactor shall not be brought above cold shutdown unless the following requirements are met:

   a. The IVSWS shall be operable.

   b. The IVSW tank shall be maintained at a minimum pressure of 52 psig and contain a minimum of 144 gallons of water.

2. The requirements of 3.3.C.1 may be modified to allow any one of the following components to be inoperable at any one time:

   a. Any one header of the IVSWS may be inoperable for a period not to exceed seven consecutive days.

   b. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within seven days and all valves in the system that provide a duplicate function have been demonstrated to be operable.

3. If the IVSWS System is not restored to an operable status within the time period specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start not later than at the end of the specified time period.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

c. In either case, if the IVSW System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48-hour period.

D. WELD CHANNEL AND PENETRATION PRESSURIZATION SYSTEM (WC & PPS)

1. The reactor shall not be brought above cold shutdown unless:

   a. All required portions of the four WC & PPS zones are pressurized at or above 47 psig.

   b. The uncorrected air consumption for the WC & PPS is less than or equal to 0.2% of the containment volume per day.

2. The requirements of 3.3.D.1 may be modified as follows:

   a. Any one zone of the WC & PPS may be inoperable for a period not to exceed seven consecutive days.

   b. The uncorrected air consumption for the WC & PPS may be in excess of 0.2% of the containment volume per day for a period not to exceed seven consecutive days.

3. If the WC & PP System is not restored to an operable status within the time period specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25^{o}F and 100 psi, respectively, over existing values.

c. In either case, if the WC & PP System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48-hour period.

E. COMPONENT COOLING SYSTEM

1. The reactor shall not be made critical unless the following conditions are met:

   a. Two component cooling pumps on busses supplied by different diesels together with their associated piping and valves are operable.

   b. Two auxiliary component cooling pumps together with their associated piping and valves are operable.

   c. Two component cooling heat exchangers together with their associated piping and valves are operable.

2. During power operation, the requirements of 3.3.E.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.E.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.E.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

   a. One of the two operable component cooling pumps may be out of service provided the pump is restored to operable status within 24 hours.

b. One auxiliary component cooling pump may be out of service provided the pump is restored to operable status within 24 hours and the other pump is demonstrated to be operable.

c. One component cooling heat exchanger or other passive component may be out of service for a period not to exceed 48 hours provided the system may still operate at design accident capability.

F. SERVICE WATER SYSTEM

1. DESIGNATED ESSENTIAL HEADER

a. The reactor shall not be above 350°F unless three service water pumps with their associated piping and valves are operable on the designated essential header.

b. When the reactor is above 350°F and one of the three service water pumps or any of its associated piping or valves is found inoperable, and an essential service water header that meets the requirements of 3.3.F.1.a. cannot be restored within 12 hours, the reactor shall be placed in the hot shutdown condition within the next 6 hours and subsequently cooled below 350°F using normal operating procedures.

2. DESIGNATED NON-ESSENTIAL HEADER

a. The reactor shall not be above 350°F unless two service water pumps with their associated piping and valves are operable on the designated non-essential header.

b. When the reactor is above 350°F and one of the two service water pumps or any of its associated piping or valves is found inoperable, and a non-essential service water header that meets the requirements of 3.3.F.2.a cannot be restored within 24 hours, the reactor shall be placed in the hot shutdown condition within the next 6 hours and subsequently cooled below 350°F using normal operating procedures.

3. INTERCONNECTION OF HEADERS

Isolation shall be maintained between the essential and non-essential headers at all times when the reactor is above 350°F except for a period of up to 8 hours when the header may be connected to facilitate safety-related activities.

4. SERVICE WATER INLET TEMPERATURE

a. The reactor shall not be above 350°F unless the service water inlet temperature is less than or equal to 95°F, or

b. When the reactor is above 350°F and the service water inlet temperature exceeds 95°F, the reactor shall be placed in the hot shutdown condition within the next 7 hours and subsequently cooled below 350°F using normal operating procedures.

c. The provisions of Specification 3.0.1 do not apply.

5. SERVICE WATER INLET TEMPERATURE MONITORING INSTRUMENTATION

a. The service water inlet temperature monitoring instrumentation shall measure the Hudson River water temperature at the Indian Point Unit No. 2 intake structure,

b. The service water inlet temperature monitoring instrumentation shall be operable when intake water temperature, averaged over a 24 hour period, reaches 80°F, and when the reactor is above 350°F,

c. When the requirements of Specification 3.3.F.5.b apply, temperature measurements shall be taken every 4 hours up to and including a service water inlet temperature of 90°F; when the service water inlet temperature exceeds 90°F, temperature measurements shall be taken once an hour,

d. If the service water inlet temperature monitoring instrumentation is declared inoperable, it shall be either restored to operable status or alternative measurements shall be taken with a calibrated portable instrument within the applicable measurement time frame requirements of Specification 3.3.F.5.c, and

e. If the requirements of Specification 3.3.F.5.d cannot be met, the reactor shall be placed in the hot shutdown condition within the next 7 hours and subsequently cooled below 350°F using normal operating procedures.

G. HYDROGEN RECOMBINER SYSTEM AND POST-ACCIDENT CONTAINMENT VENTING SYSTEM

1. The reactor shall not be made critical unless the following conditions are met:

   a. Both hydrogen recombiner units together with their associated piping, valves, oxygen supply system and control system are operable, with the exception of one recombiner unit's equipment located outside the containment which may be inoperable, provided it is under repair and can be made operable if needed.

   b. The post-accident containment venting system is operable.

   c. Hydrogen and oxygen supplies shall not be connected to the hydrogen recombiner units except under conditions of an accident or those specified in Specification 4.5.C.1.

2. During power operation, the requirements of 3.3.G.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.G.1 within the time specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures.

a. One hydrogen recombiner unit or its associated flow path, or oxygen supply system or control system may be inoperable for a period not to exceed thirty days, provided the other recombiner unit and the post-accident containment venting system are operable.

b. The post-accident containment venting system may be inoperable for a period not to exceed thirty days provided that both hydrogen recombiners are operable.

H. CONTROL ROOM AIR FILTRATION SYSTEM

1. The control room air filtration system shall be operable at all times when containment integrity is required.

2. From the date that the control room air filtration system becomes and remains inoperable for any reason, operations requiring containment integrity are permissible only during the succeeding 3.5 days. At the end of this 3.5 days period, if the conditions for the control room air filtration system cannot be met, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the conditions are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3. Two independent toxic gas detection systems, each capable of detecting chlorine, anhydrous ammonia, and hydrogen cyanide, shall be operable at all times except as specified in 3.a, 3.b, or 3.c below. The alarm/trip setpoint for each toxic gas system shall be adjusted to actuate at a toxic gas concentration of less than or equal to 3.5 ppm.

   a. With one toxic gas detection system inoperable, restore the inoperable detection system to operable status within 7 days.

b. If 3.a above cannot be satisfied within the specified time, then, within the next 6 hours, initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

c. With both toxic gas detection systems inoperable for any one toxic gas, within one hour initiate and maintain operation of the control room ventilation in the recirculation mode of operation.

I. CABLE TUNNEL VENTILATION FANS

1. The reactor shall not be made critical unless the two cable tunnel ventilation fans are operable.

2. During power operation, the requirement of 3.3.I.1 may be modified to allow one cable tunnel ventilation fan to be inoperable for seven days, provided the other fan is operable.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant[1]. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and therefore the minimum required engineered safeguards and auxiliary cooling systems are required to be operable. During low-temperature physics tests there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safeguards systems are not required.

When the reactor is critical, the probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus operation with the reactor critical with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. Inoperability of a single component does not negate the ability of the system to perform its function[2], but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. If it develops that (1) the inoperable component is not repaired within the specified allowable time period, or (2) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. The specified repair times do not apply to regularly scheduled maintenance of the engineered safeguards systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on two considerations:

1. assurance with high reliability that the safeguard system will function properly if required to do so, and

2. allowance of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full-rated power for at least 100 days, the magnitude of the decay heat decreases after initiating hot shutdown. Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance, and therefore in such a case the reactor is to be put into the cold shutdown condition.

Valves 1810, 744 and 882 are kept in the open position during plant operation to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As an additional assurance of flow passage availability, the valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves to take place. This additional precaution is acceptable since failure to manually re-establish power to close valves 1810 and 882, following the injection phase, is tolerable as a single failure. Valve 744 will not need to be closed following the injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes[3]. The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met[9]. The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-9 of the UFSAR.

The requirement regarding the maximum number of SI pumps that can be energized when RCS temperature is less than or equal to 295°F is discussed under Specification 3.1.A.

The containment cooling and iodine removal functions are provided by two independent systems: (1) fan-coolers plus charcoal filters and (2) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F)[12]. In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting offsite doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also, in the event of a Design Basis Accident, three charcoal filters (and their associated recirculation fans) in operation, along with one containment spray pump and sodium hydroxide addition, will reduce airborne organic and molecular iodine activities sufficiently to limit offsite doses to acceptable values. These constitute the minimum safeguards for iodine removal, and are capable of being operated on emergency power with one diesel generator inoperable.

If offsite power is available or all diesel generators are operating to provide emergency power, the remaining installed iodine removal equipment (two charcoal filters and their associated fans, and one containment spray pump and sodium hydroxide addition) can be operated to provide iodine removal in excess of the minimum requirements. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of offsite power or operation of all emergency diesel generators.

One of the five fan cooler units is permitted to be inoperable during power operation. This is an abnormal operating situation, in that the normal plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical.

However, because of the difficulty of gaining access to make repairs, it is important on occasion to be able to operate temporarily without at least one fan-cooler. Compensation for this mode of operation is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

The Component Cooling System is different from the system discussed above in that the pumps are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident(6). During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards(7).

A total of six service water pumps are installed. Only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident(8). The limit on the service water maximum inlet temperature assures that the service water and component cooling water systems will be able to dissipate the heat loads generated in the limiting design basis accident.(12)

During the second phase of the accident, one additional service water pump on the non-essential header will be manually started to supply the minimum cooling water requirements for the component cooling loop.

The limits for the accumulators and their pressure and volume assure the required amount of water injection following a loss-of-coolant accident, and are based on the values used for the accident analysis[9].

Two independent diverse systems are provided for removal of combustible hydrogen from the containment building atmosphere: (1) the hydrogen recombiners, and (2) the post-accident containment venting system. Either of the two (2) hydrogen recombiners or the post-accident containment venting system are capable of wholly providing this function in the event of a design basis accident.

Two full-rated hydrogen recombination systems are provided in order to control the hydrogen evolved in the containment following a loss-of-coolant accident. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the containment. Each system is separate from the other and is provided with redundant features. Power supplies for the blowers and ignitors are separate, so that loss of one power supply will not affect the remaining system. Hydrogen gas is used as the externally supplied fuel. Oxygen gas is added to the containment atmosphere through a separate containment feed to prevent depletion of oxygen in the air below the concentration required for stable operation of the combustor (12%). The containment atmosphere sampling system consists of a sample line which originates in each of the containment fan cooler units. The fan and sampling pump head together are sufficient to pump containment air in a loop from the fan cooler through a containment penetration to a sample vessel outside the containment, and then through a second penetration to the sample termination inside the containment. The design hydrogen concentration for operating the recombiner is established at 2% by volume. Conservative calculations indicate that the hydrogen content within the containment will not reach 2% by volume until 13 days after a loss-of-coolant accident. There is therefore no need for immediate operation of the recombiner following an accident, and the quantity of hydrogen fuel stored at the site will be only for periodic testing of the recombiners.

The Post-Accident Containment Venting System consists of a common penetration line which acts as a supply line through which hydrogen-free air can be admitted to the containment, and an exhaust line, with parallel valving and piping, through which hydrogen-bearing gases from containment may be vented through a filtration system.

The supply flow path makes use of instrument air to feed containment. The nominal flow rate from either of the two instrument air compressors is 200 scfm. If the instrument air system is not available, the station air system is available as a backup.

The exhaust line penetrates the containment and then is divided into two parallel lines. Each parallel line contains a pressure sensor and all the valves necessary for controlling the venting operation. The two lines then rejoin and the exhaust passes through a flow sensor and a temperature sensor before passing through roughing, HEPA and charcoal filters. The exhaust is then directed to the plant vent.

The post-accident containment venting system is a passive system in the sense that a differential pressure between the containment and the outside atmosphere provides the driving force for the venting process to take place. The system is designed such that a minimum internal containment pressure of 2.14 psig is required for the system to operate properly.

The flow rate and the duration of venting required to maintain the hydrogen concentration at or below 3 percent of the containment volume are determined from the containment hydrogen concentration measurements and the hydrogen generation rate. The containment pressure necessary to obtain the required vent flow is then determined. Using one of the air compressors, hydrogen-free air is pumped into the containment until the required containment pressure is reached. The air supply is then stopped and the supply/exhaust line is isolated by valves outside the containment. The addition of air to pressurize the containment dilutes the hydrogen; therefore, the containment will remain isolated until analysis of samples indicates that the concentration is again approaching 3 percent by volume. Venting will then be started. This process of containment pressurization followed by venting is repeated as may be necessary to maintain the hydrogen concentration at or below 3 volume percent.

The post-accident venting system is used only in the absence of hydrogen recombiners and only when absolutely necessary. From the standpoint of minimizing offsite radiation doses, the optimum starting time for the venting system, if needed, is the latest possible time after the accident. Consistent with this philosophy, the selected venting initiation point of 3 percent hydrogen maximizes the time period before venting is required while at the same time allows a sufficient margin of safety below the lower flammability limit of hydrogen.

The control room air filtration system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room system is designed to automatically start upon control room isolation. Control room isolation is initiated either by a safety injection signal or by detection of high radioactivity in the control room. If the control room air filtration system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 3.5 days, the reactor is placed in the hot shutdown condition.

The control room ventilation system is equipped with a toxic gas detection system consisting of redundant monitors capable of detecting chlorine, anhydrous ammonia, and hydrogen cyanide. These toxic gas detection systems are designed to isolate the control room from outside air upon detection of toxic concentration of the monitored gases in the control room ventilation system. The operability of the toxic gas detection systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored and the setpoint established for the monitors are based on the results described in the Indian Point Unit No. 2 Control Room Habitability Study dated May, 1981.

The cable tunnel is equipped with two temperature-controlled ventilation fans. Each fan has a capacity of 21,000 cfm and is connected to a 480v bus. One fan will start automatically when the temperature in the tunnel reaches 100°F. Under the worst conditions, i.e., loss of outside power and all the Engineered Safety Features in operation, one ventilation fan is capable of maintaining the tunnel temperature below 104°F. Under the same worst conditions, if no ventilation fans were operating, the natural air circulation through the tunnel would be sufficient to limit the gross tunnel temperature to below the tolerable value of 140°F. However, in order to provide for ample tunnel ventilation capacity, the two ventilation fans are required to be operable when the reactor is made critical. If one ventilation fan is found inoperable, the other fan will ensure that cable tunnel ventilation is available.

Valves 856A, C, D and E are maintained in the open position during plant operation to assure a flow path for high-head safety injection during the injection phase of a loss-of-coolant accident. Valves 856B and F are maintained in the closed position during plant operation to prevent hot-leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot-leg injection, the valve motor operators are de-energized to prevent spurious opening of these valves. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot-leg recirculation.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

The specified quantities of water for the RWST include unavailable water (4687 gals) in the tank bottom, inaccuracies (24,800 gals) in the alarm setpoints, the minimum quantity required during the injection (246,000 gals)[12] for accident mitigation and the minimum quantity required during the recirculation phase (60,000 gals) for post-LOCA NaOH requirements inside containment. The minimum RWST inventory (i.e., 345,000 gals) provides approximately 9,500 gallons margin.

The seven-day out-of-service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of offsite accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems[11]. The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems and assures that the containment design pressure of 47 psig is not exceeded.

References

(1) UFSAR Section 9
(2) UFSAR Section 6.2
(3) UFSAR Section 6.2
(4) UFSAR Section 6.4
(5) Reference Deleted
(6) UFSAR Section 9.3
(7) UFSAR Section 9.3
(8) UFSAR Section 9.6.1
(9) UFSAR Section 14.3
(10) Indian Point Unit No. 2, UFSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.
(11) UFSAR Sections 6.5 and 6.6
(12) WCAP-12312, "Safety Evaluation for An Ultimate Heat Sink Temperature to 95°F at Indian Point Unit 2", July, 1989.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and City Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specifications

A. The reactor shall not be heated above 350°F unless the following conditions are met:

1. A minimum ASME code-approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing). With up to three (per steam generator) of the twenty main steam line code-approved safety relief valves inoperable, heat-up above 350°F and power operation is permissible provided either the inoperable valve(s) is restored to operable status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.4-1.

2. Three auxiliary feedwater pumps, each capable of pumping a minimum of 380 gpm, must be operable.

3. A minimum of 360,000 gallons of water is in the condensate storage tank and a backup supply from the city water supply.

4. Required system piping, valves, and instrumentation directly associated with the above components are operable.

5. The main steam stop valves are operable and capable of closing in five seconds or less.

6. The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. Except as modified by 3.4.B(1) and 3.4.C below, if any of the conditions of 3.4.A above cannot be met within 72 hours, the reactor shall be placed in the hot shutdown condition within the next 12 hours and subsequently cooled below 350°F using normal operating procedures.

(1) With one or more auxiliary feedwater pump(s) inoperable take the following actions:

a) With one auxiliary feedwater pump inoperable, restore the pump to operable status within 72 hours or place the reactor in the hot shutdown condition and subsequently cool the RCS to below 350°F using normal operating procedures within the next 12 hours.

b) With two auxiliary feedwater pumps inoperable, place the reactor in hot shutdown and subsequently cool the RCS below 350°F using normal operating procedures within 12 hours.

c) With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to operable status while maintaining power at existing level until at least one auxiliary feedwater pump has been restored to operable status, and then immediately comply with the requirements of 3.4.B(1)(b) above.

C. If, when above 350°F, one or both of the series valves (CT-6 and/or CT-64) in the condensate storage tank discharge line is closed, then:

1. immediately place the auxiliary feedwater pump controls in the manual mode, and

2. within one (1) hour, either the valve(s) shall be reopened or the valves from the alternate city water supply shall be opened and the auxiliary feedwater pump controls restored to the automatic mode.

If these requirements cannot be met, then:

1. maintain the plant in a safe, stable mode which minimizes the potential for a reactor trip, and

2. continue efforts to restore water supply to the auxiliary feedwater system, and

3. notify the NRC within 24 hours regarding the planned corrective action.

Basis

Reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The operability of the twenty main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% Rated Thermal Power coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The total relieving capacity of the twenty main steam safety valves is 15,108,000 lbs/hr which is 114 percent of the total secondary steam flow of 13,310,000 lbs/hr at 100% NSSS Power (3083.4 Mwt). Startup and/or power operation is allowable with main steam safety valves inoperable within the limitations of Table 3.4-1 on the basis of the reduction in secondary system steam flow and thermal power required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \left[\frac{X - Y V}{X} \right] \cdot 109$$

Where:

SP = Reduced reactor trip setpoint in percent of rated thermal power

V = Maximum number of inoperable safety valves per steam line

109 = Power Range Neutron Flux-High Trip Setpoint for (4) loop operation

X = Total relieving capacity of all safety valves per steam line (3,777,000 lbs/hr.)

Y = Maximum relieving capacity of any one safety valve (823,000 lbs/hr.)

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot standby. When the condensate storage supply is exhausted, city water will be used.

The limit on secondary coolant total iodine activity of I-131 and I-133 is based on a postulated release of secondary coolant equivalent to the contents of four steam generators to the atmosphere due to a net load rejection with loss-of-offsite power. The limiting dose for this case would result from radioactive iodine in the secondary coolant. I-131 and I-133 are the dominant isotopes because of their low MPCs in air and because the other, shorter-lived isotopes cannot build up to significant concentrations in the secondary coolant under the limits of primary system leak rate and activity. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets. The inhalation dose at the site boundary is then as follows:

$$\text{Dose(rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot X/Q \cdot DCF$$

where: C = secondary coolant activity (0.15 µCi/cc = 0.15 Ci/m^3)

V = water volume in four steam generators (7416 ft^3 = 210 m^3)

B(t) = breathing rate (3.47×10^{-4} m^3/sec)

X/Q = 7.5×10^{-4} sec/m^3

DCF = 1.00×10^6 rem/Ci Iodine (131 and 133) inhaled

The resultant dose is less than 1.0 rem.

Reference

UFSAR - Chapter 10 and Section 14.1.9

normal detector signal to test at power. Testing of the NIS power range channel requires (1) bypassing the Dropped Rod protection from NIS for the channel being tested, and (2) defeating the ΔT protection channel set that is being fed from the NIS channel, and (3) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position system and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

References

(1) UFSAR Section 7.2

(2) UFSAR Section 14.3

(3) UFSAR Section 14.2.5

(4) Safety Evaluation accompanying the Indian Point Unit No. 2 "Application for Amendment to Operating License," sworn to on May 29, 1979 by Mr. William J. Cahill, Jr. of Consolidated Edison.

TABLE 3.4-1

Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4-Loop Operation

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	85
2	61
3	37

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to plant instrumentation systems.

Objectives

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specifications

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For on-line testing or instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of channels of a particular function in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3.5-2 through 3.5-4.

3.5.4 In the event of sub-system instrumentation channel failure permitted by Specification 3.5.2, Tables 3.5-2 through 3.5-4 need not be observed during the short period of time the operable sub-system channels are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

3.5.5 The cover plate on the rear of the safeguards panel in the control room shall not be removed without authorization from the Watch Supervisor.

3.5.6 When the reactor coolant system is above 350°F, the instrumentation requirements as stated in Table 3.5-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features[1,4].

Safety Injection System Actuation

Protection against a loss-of-coolant or steam break accident is brought about by automatic actuation of the Safety Injection System, which provides emergency cooling and reduction of reactivity.

The loss-of-coolant accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the loss-of-coolant accident by detecting low pressure and generator signals actuating the SIS active phase.

The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steamline-break accident. Therefore, SIS actuation following a steamline break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steamline flow in coincidence with low reactor coolant average temperature or low steamline pressure.

The increase in the extraction of RCS heat following a steamline break results in reactor coolant temperature and pressure reduction. For this reason, protection against a steamline-break accident is also provided by low pressurizer pressure signals actuating safety injection.

Protection is also provided for a steamline break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steamline-break accident.

Containment Spray

The Engineered Safety Features actuation system also initiates containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss-of-coolant or steamline-break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (2.0 psig). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals.

Steamline Isolation

Steamline isolation signals are initiated by the Engineered Safety Features closing all steamline stop valves. In the event of a steamline break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steamlines on high containment pressure (Hi-Hi Level) or high steamline flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steamline isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effect of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steamline break inside the containment by stopping the entry of feedwater.

Setting Limits

1. The Hi Level containment pressure limit is set at 2.0 psig containment pressure. Initiation of Safety Injection protects against loss-of-coolant(2,4) or steamline-break(3,4) accidents as discussed in the safety analysis.

2. The Hi-Hi Level containment pressure limit is set at about 50% of design containment pressure. Initiation of Containment Spray and Steamline Isolation protects against large loss of coolant(2) or steamline-break accidents(3) as discussed in the safety analysis.

3. The pressurizer low-pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis(2).

4. The steamline high differential pressure limit is set well below the differential pressure expected in the event of a large steamline-break accident as shown in the safety analysis[3].

5. The high steamline flow limit is set at approximately 40% of the full steam flow at 0% to 20% load. Between 20% and 100% (full) load, the trip setpoint is ramped linearly with respect to first stage turbine pressure from 40% of the full steam flow to 110% of the full steam flow. These setpoints will initiate safety injection in the case of a large steamline-break accident. Coincident low T_{avg} setting limit for SIS and steamline isolation initiation is set below its hot shutdown value. The coincident steamline pressure setting limit is set below the full load operating pressure. The safety analyses show that these settings provide protection in the event of a large steamline break[3].

Instrument Operating Conditions

During plant operation, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires (1) bypassing the Dropped Rod protection from NIS for the channel being tested, and (2) defeating the ΔT protection channel set that is being fed from the NIS channel, and (3) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position system and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

<u>References</u>

(1) UFSAR Section 7.2

(2) UFSAR Section 14.3

(3) UFSAR Section 14.2.5

(4) Safety Evaluation accompanying the Indian Point Unit No. 2 "Application for Amendment to Operating License," sworn to on May 29, 1979 by Mr. William J. Cahill, Jr. of Consolidated Edison.

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
1.	Manual	2	1	1	0	Maintain hot shutdown
2.	Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
2.a	Nuclear Flux Power Range	4	2	2	1	For zero power physics tests only
3.	Nuclear Flux Intermediate Range	2	1	1*	0	Maintain hot shutdown
4.	Nuclear Flux Source Range	2	1	1**	0	Maintain hot shutdown
5.	Overtemperature delta T	4	2	3	2	Maintain hot shutdown
6.	Overpower delta T	4	2	3	2	Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redundancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
8.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
9.	Pressurizer Hi Water Level	3	2	2	1	Maintain hot shutdown
10.	Low Flow Loop ≥ 75% F.P.	3/loop	2/loop (any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops 10-75% F.P.	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	
11.	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
12.	Undervoltage 6.9 kV Bus	1/bus	2	3	2	Maintain hot shutdown
13.	Low frequency 6.9 kV Bus	1/bus	2	3	2	Maintain hot shutdown***
14.	Quadrant power tilt monitors	2	NA	1	0	Log individual upper and lower ion chamber currents once/shift and after load change > 10%

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redundancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
15.	Turbine trip (overspeed protection)*****	3	2	2	1	Maintain hot shutdown
16.	Control Rod Protection****	3	2	2	1	During RCS cooldown, manually open reactor trip breakers prior to T_{cold} decreasing below 350°F. Maintain reactor trip breakers open during RCS cooldown when T_{cold} is less than 350°F.
17.	Turbine Trip ≥ 35% F.P. A. Low Auto Stop Oil Pressure	3	2	2	1	Maintain reactor power below 35% F.P.
18.	Reactor Trip Logic	2	1	2#	1#	Be in hot shutdown within the next six hours.

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redundancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
19.	Reactor Trip Breakers	2	1	2#	1#	With either diverse trip feature inoperable, or the breaker incapable of tripping for any other reason, restore it to operable conditions or, be in hot shutdown within the next six hours and open both reactor trip breakers. The breaker shall not be bypassed except for the time required for performing maintenance and/or testing to restore it to operability.

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

F.P. = Rated Power

* If two of four power range channels are greater than 10% F.P., channels are not required.

** If one of two intermediate range channels is greater than 10^{-10} amps, channels are not required.

*** 2/4 trips all four reactor coolant pumps.

**** Required only when control rods are positioned in core locations containing LOPAR fuel.

***** This will provide a turbine trip at all power levels and a reactor trip when greater than or equal to 35% F.P.

\# A reactor trip breaker and/or associated logic channel may be bypassed for maintenance or surveillance testing for up to eight hours provided the redundant reactor trip breaker and/or associated logic channel is operable.

Table 3.5-3

Instrumentation Operating Conditions for Engineered Safety Features

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
1	Safety Injection					
a.	Manual	2	1	1	0	Cold shutdown
b.	High Containment Pressure (Hi Level)	3	2	2	1	Cold shutdown
c.	High Differential Pressure Between Steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold shutdown
d.	Pressurizer Low Pressure*	3	2	2	1	Cold shutdown
e.	High Steam Flow in 2/4 Steam Lines Coincident	2/line	1/2 in any 2 lines	1/line in each of 3 lines	2	Cold shutdown
	With Low T_{avg} or Low Steam Line Pressure	4 T_{avg} Signals	2	3	2	
		4 Pres-sure Signals	2	3	2	

* Permissible bypass if reactor coolant pressure less than 2000 psig.

Table 3.5-3

Instrumentation Operating Conditions for Engineered Safety Features

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redundancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
2.	Containment Spray					
a.	Manual	2	1	1	0	Cold shutdown
b.	High Containment Pressure (Hi-Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold shutdown
3.	Loss Of Power					
a.	480V Emergency Bus Undervoltage (Loss of Voltage)	2/bus	1/bus	1/bus	0	Cold shutdown
b.	480V Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	1/bus	0	Cold shutdown
4.	Auxiliary Feedwater					
a.	Steam Gen. Water Level (Low-Low)					
	i. Start Motor-Driven Pumps	3/stm gen.	2 in any stm gen.	2 chan. in each stm gen.	1	Reduce RCS temperature such that T < 350^{o}F

Table 3.5-3

Instrumentation Operating Conditions for Engineered Safety Features

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redundancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
	ii. Start Turbine-Driven Pump	3/stm gen.	2/3 in each of two stm gen.	2 chan. in each stm gen.	1	T < 350^{o}F
b.	S.I. Start Motor-Driven Pumps	(All safety injection initiating functions and requirements)				
c.	Station Blackout Start Motor-Driven and Turbine-Driven Pumps	2	1	1	0	T < 350^{o}F
d.	Trip of Main Feed-water Pumps Start Motor-Driven Pumps	2	1	1	0	Hot shutdown
5.	Overpressure Protection System (OPS)	3	2	2	1	Refer to Specification 3.1.A.4

Table 3.5-4

Instrumentation Operating Conditions for Isolation Functions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
1.	Containment Isolation					
a.	Automatic Safety Injection (Phase A)	See Item No. 1 of Table 3.5-3				Cold shutdown
b.	Containment Pressure (Phase B)	See Item No. 2 of Table 3.5-3				Cold shutdown
c.	Manual					
	Phase A (one out of two)	2	1	1	0	Cold shutdown
	Phase B (one out of two)	2	1	1	0	Cold shutdown
2.	Steam Line Isolation					
a.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3.5-3				Cold shutdown
b.	High Containment Pressure (Hi-Hi Level)	See Item No. 2(b) of Table 3.5-3				Cold shutdown
c.	Manual	1/loop	1/loop	1/loop	0	Cold shutdown

Table 3.5-4

Instrumentation Operating Conditions for Isolation Functions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redundancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
3.	Feedwater Line Isolation					
a.	Safety Injection	See Item No. 1 of Table 3.5-3				
4.	Containment Purge And Pressure Relief Isolation					
a.	Containment Radioactivity High (R-11/R-12)	2	1	*	0	*

* See Specification 3.1.F

Table 3.5-5

Accident Monitoring Instrumentation

Instrument	Minimum No. of Channels Operable*	Applicable Actions
1. Pressurizer Water Level	2	1
2. Reactor Coolant System Subcooling Margin Monitor	1**	1,2
3. PORV Position Indicator (Limit Switch)	1/valve***	1
4. PORV Block Valve Position Indicator (Limit Switch)	1/valve****	1
5. Safety Valve Position Indicator (Acoustic Monitor)	1/valve	1
6. Auxiliary Feedwater Flow Rate	1/S.G.*****	1
7. Wide Range Containment Pressure Monitor (PT-3300, PT-3301)	1	3
8. Plant Vent Noble Gas Effluent Monitor (R-27)	1	3
9. Main Steam Line Radiation Monitor (R-28, R-29, R-30, R-31)	1/steam line	3
10. High Range Containment Radiation Monitor (R-25, R-26)	1	3
11. Containment Hydrogen Concentration Monitor	2	3

Accident Monitoring Instrumentation

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* Encompasses the entire channel from sensor to display where either an indicator, recorder or alarm is acceptable.

** PROTEUS Subcooling margin readout can be used as substitute for the subcooling monitor.

*** Except at times when the associated block valve is closed and de-energized. Acoustic monitoring of PORV position (headered discharge) can be used as a substitute for the PORV Position Indicator-Limit Switches instrument.

**** Except at times when the block valve is closed and de-energized.

***** Steam Generator Level instrumentation can be used as a substitute for auxiliary feedwater flow rate monitoring.

Table 3.5-5

Accident Monitoring Instrumentation

Action Statements

1. With the number of operable accident monitoring instrumentation channels less than the Minimum Channels Operable requirement of Table 3.5-5, restore the inoperable instrument channel(s) to operable status within 7 days and/or recorder(s) within 14 days. If the minimum number of channels required is not restored to meet the above requirements within the time periods specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

   b. If the requirements are not satisfied within an additional 48 hours after hot shutdown, the reactor shall be cooled to below 350°F utilizing normal operating procedures. The cooldown shall start no later than the end of the 48-hour period.

2. If the subcooling margin monitor is inoperable for more than seven (7) days, plant operation may continue for an additional thirty (30) days provided that steam tables are continuously maintained in the control room and the subcooling margin is determined and recorded once a shift.

3. With the number of operable channels less than required by the minimum number of channels requirement, within 7 days either restore the inoperable channel(s) to the minimum operable status or:

   a. Initiate an alternate method of monitoring the appropriate parameter(s), and

   b. Prepare and submit a special report to the Commission pursuant to Specification 6.9.2.h within the next 14 days following the event.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specifications

A. CONTAINMENT INTEGRITY

1. The following requirements shall be satisfied: (a) whenever the reactor is above cold shutdown or (b) whenever the reactor vessel head is less than fully tensioned, and (i) the shutdown margin is <5% $\Delta k/k$, or (ii) the boron concentration within the reactor is less than 2000 ppm.

   a. All non-automatic containment isolation valves which are not required to be open during accident conditions are closed and blind flanges installed where required. Those non-automatic containment isolation valves listed in Table 3.6-1 and any test connection valves which are located between containment isolation valves and which are normally closed with threaded caps or blind flanges installed, may be opened if necessary for plant operation or for testing and only as long as necessary to perform the intended function.

   b. All automatic containment isolation valves are either operable or in the closed position or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

   c. The equipment door is properly closed.

   d. At least one door in each personnel air lock is properly closed.

e. The WC&PPS requirements of Specification 3.3.D are being satisfied.

f. Containment leakage has been verified in accordance with the surveillance requirements of Specification 4.4.

2. The following additional requirements shall be satisfied during power operation:

   a. The automatic containment purge and containment pressure relief isolation valves are set to limit valve disk travel to no greater than 60^{o} open (90^{o} being full open) with stroke times of three seconds or less.

   b. The automatic containment purge and containment pressure relief isolation valves may only be open for safety-related reasons.[1)]

3. Except as specified in 3.a. below, if the above requirements are not satisfied, the condition shall be corrected within 4 hours or the reactor shall be brought to a cold shutdown condition within the next 36 hours, utilizing normal operating procedures.

   a. With one or more isolation valve(s) inoperable:

      1. maintain at least one isolation valve operable in each affected penetration[2)], and

      2. either:

         (a) Restore the inoperable valve(s) to operable status within 4 hours, or

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1) Examples of safety-related reasons include containment pressure control, or to facilitate safety-related surveillance or safety-related maintenance.

2) Not required for penetrations equipped with only one isolation valve.

(b) Isolate each affected penetration within 4 hours by use of at least one deactivated automatic isolation valve secured in the isolation position[3], or

(c) Isolate each affected penetration within 4 hours by use of at least one closed manual valve[3] or blind flange that meets the design criteria for an isolation valve, or

(d) Be in cold shutdown within the following 36 hours, utilizing normal operating procedures.

4. Non-automatic containment isolation valves may be added to plant systems, without prior license amendment to Table 3.6-1, provided that a revision to this Table is included in a subsequent license amendment application.

B. INTERNAL PRESSURE

If the internal pressure exceeds 2 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shut down.

C. CONTAINMENT TEMPERATURE

The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.

Basis

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if a Reactor Coolant System rupture were to occur.

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3) This may be the valve previously maintained operable per 3.a.1 above or the valve initially declared inoperable.

The shutdown margins are selected based on the type of activities that are being carried out. The shutdown margin requirement of Specification 3.8.B.2 when the head is off precludes criticality during refueling. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality at cold shutdown conditions.

Regarding internal pressure limitations, the containment calculated peak accident pressure of 47 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 8 psig[1]. The containment can withstand an internal vacuum of 2.5 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material[3].

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. During periods of normal plant operations requiring containment integrity, valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. The valves to be open intermittently are under administrative control and are open only for as long as necessary to perform their intended function. In all cases, however, the valves listed in Table 3.6-1 are closed during the post-accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

References

(1) UFSAR Section 14.3.5
(2) Deleted
(3) UFSAR Section 5.1.1.1

Non-Automatic Containment Isolation Valves Open Continuously Or Intermittently for Plant Operation

		SWN-44-5-A or B[1]	1814B
3418	851A	SWN-51-5[1]	1814C
3419	850A	SWN-44-1-A or B[1]	
4136	851B	SWN-51-1[1]	5018
			5019
744	850B	SWN-44-2-A or B[1]	5020
		SWN-51-2[1]	
888A	859A	SWN-44-3-A or B[1]	5021
888B	859C	SWN-51-3[1]	5022
958	863		5023
959	3416	SWN-44-4-A or B[1]	5024
990D	3417	SWN-51-4[1]	5025
1870	5459	SWN-71-5-A or B[1]	E-2
743	753H	SWN-71-1-A or B[1]	E-1
732	753G	SWN-71-2-A or B[1]	E-3
885A	SWN-41-5-A or B[1]	SWN-71-3-A or B[1]	E-5
885B	SWN-42-5	SWN-71-4-A or B[1]	MW-17
			MW-17-1
205	SWN-43-5	SA-24	85C
226	SWN-41-1-A or B[1]	SA-24-1	85D
227	SWN-42-1	PCV-1111-1	95C
250A	SWN-43-1	PCV-1111-2	95D
4925	SWN-41-2-A or B[1]	580A	
250B	SWN-42-2	580B	
4926	SWN-43-2	UH-43	
250C	SWN-41-3-A or B[1]	UH-44	
4927	SWN-42-3	990A	
250D	SWN-43-3	990B	
4928	SWN-41-4-A or B[1]	1814A	
869A	SWN-42-4		
878A	SWN-43-4		
869B			

(1) Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

Applies to the availability of electrical power for the operation of plant auxiliaries.

Objective

To define those conditions of electrical power availability necessary (1) to provide for safe reactor operation and (2) to provide for the continuing availability of engineered safety features.

Specifications

A. The reactor shall not be made critical without:

1. two 138 kV lines to Buchanan fully operational,

2. the 6.9 kV buses 5 and 6 energized from the 138 kV source,

3. one 13.8 kV source fully operational and the 13.8/6.9 kV transformer available to supply 6.9 kV power,

4. the four 480-volt buses 2A, 3A, 5A and 6A energized and the bus tie breakers between buses 5A and 2A and between buses 3A and 6A open,

5. three diesel generators operable with onsite supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available onsite other than the normal supply tanks, and

6. station batteries Nos. 21, 22, 23, & 24 and their associated battery chargers and dc distribution systems operable.

B. During power operation, the following components may be inoperable:

1. Power operation may continue for seven days if one diesel is inoperable provided the 138 kV and the 13.8 kV sources of offsite power are available and the remaining diesel generators are tested daily to ensure operability and the engineered safety features associated with these diesel generator buses are operable.

2. Power operation may continue for 24 hours, if the 138 kV or the 13.8 kV source of power is lost, provided the three diesel generators are operable. This operation may be extended beyond 24 hours provided the failure is reported to the NRC within the subsequent 24-hour period with an outline of the plans for restoration of offsite power.

3. If the 138 kV power source is lost, in addition to satisfying the requirements of Specification 3.7.B.2 above, the 6.9 kV bus tie breaker control switches 1-5, 2-5, 3-6, and 4-6 in the CCR shall be placed in the "pull-out" position and tagged to prevent an automatic transfer of the 6.9 kV buses 1, 2, 3 and 4.

4. One battery may be inoperable for 24 hours provided the other batteries and four battery chargers remain operable with one battery charger carrying the dc load of the failed battery's supply system.

5. One battery charger may be inoperable for 24 hours provided the following conditions are satisfied:

    a. The other three battery chargers and their associated batteries are operable; and

    b. The affected battery shall have the Specification 4.6.C.1 surveillance initiated within one hour of the time the battery charger is determined to be inoperable and the surveillance shall be repeated every eight hours thereafter to determine battery operability. This surveillance frequency shall be maintained until the battery is declared inoperable or until the battery charger is declared operable.

C. Gas Turbine Generators:

1. At least one gas turbine generator (GT-1, GT-2 or GT-3) and associated switchgear and breakers shall be operable at all times.

2. A minimum of 54,200 gallons of fuel for the operable gas turbine generator shall be available at all times.

3. If the requirements of 3.7.C.1 or 3.7.C.2 cannot be met, then, within the next seven (7) days, either the inoperable condition shall be corrected or an alternate independent power system shall be established.

4. If the requirements of 3.7.C.3 cannot be satisfied, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.7.C.3 cannot be met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

The requirements of Specification 3.7.A may be modified for an emergency "Black Start" of the unit by using the requirements of either Specification 3.7.D.1 or 3.7.D.2 below:

D.1. a. all 138 kV lines to Buchanan de-energized,

b. the 13.8 kV line de-energized,

c. the 6.9 kV buses 5 and 6 energized from the onsite gas turbine through the 13.8/6.9 kV transformer,

d. the four 480-volt buses 2A, 3A, 5A and 6A energized from the diesels and the tie breakers between buses 5A and 2A and between buses 3A and 6A open,

e. three diesel generators operable with onsite supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available on-site other than the normal supply tanks,

f. station batteries Nos. 21, 22, 23 & 24 and their associated battery chargers and dc distribution systems operable, and

g. the 480-volt tie breakers 52/2A, 52/3A, 52/5A and 52/6A open.

D.2. a. establish 138 kV bus sections at Buchanan with at least 37 MW power (nameplate rating) from any combination of gas turbines at Buchanan and onsite,

b. two 138 kV lines to Buchanan energized from the gas turbines with breakers to Millwood and Orange and Rockland open,

c. the 13.8 kV line to Buchanan operable and the 13.8/6.9 kV transformer available to supply 6.9 kV power,

d. the 6.9 kV buses energized from the 138 kV source,

e. the four 480-volt buses 2A, 3A, 5A and 6A energized and the bus tie breakers between buses 5A and 2A and between buses 3A and 6A open,

f. three diesel generators operable with onsite supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available on-site other than the normal supply tanks, and

g. station batteries Nos. 21, 22, 23 & 24 and their associated battery chargers and dc distribution systems operable.

E. Whenever the reactor is critical, the circuit breaker on the electrical feeder to emergency lighting panel 218 inside containment shall be locked open except when containment access is required.

Basis

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize plant safety. The 480-volt equipment is arranged in four buses. The 6.9 kv equipment is supplied from six buses.

In addition to the unit transformer, three separate sources supply station service power to the plant(1).

The plant auxiliary equipment is arranged electrically so that multiple items receive their power from different sources. The charging pumps are supplied from the 480-volt buses Nos. 3A, 5A, and 6A. The five containment fans are divided among the 480-volt buses. The two residual heat pumps are on separate 480-volt buses. Valves are supplied from separate motor control centers.

The station auxiliary transformer or a gas turbine is capable of providing sufficient power for plant startup. The station auxiliary transformer can supply the required plant auxiliary power during normal operation.

The bus arrangements specified for operation ensure that power is available to an adequate number of safeguards auxiliaries. With additional switching, more equipment could be out of service without infringing on safety.

Two diesel generators have sufficient capacity to start and run, at design load, the minimum required engineered safeguards equipment(1). The minimum diesel fuel oil inventory at all times is maintained to assure the operation of two diesels carrying the load of the minimum required engineered safeguards equipment for at least eighty hours(2). Additional fuel oil suitable for use in the diesel generators will be stored onsite. The minimum storage of 22,000 gallons will assure operation of two diesels for ninety hours at the minimum load for engineered safeguards. Commercial oil supplies and trucking facilities exist to assure deliveries within one day's notice.

One battery charger shall be in service on each battery so that the batteries will always be at full charge in anticipation of a loss-of-ac power incident. This ensures that adequate dc power will be available for starting the emergency diesel generators and other emergency uses.

The plant can be safely shut down without the use of offsite power since all vital loads (safety systems, instruments, etc.) can be supplied from the emergency diesel generators.

Any two of three diesel generators, the station auxiliary transformer or the separate 13.8 to 6.9 kV transformer are each capable of supplying the minimum safeguards loads and therefore provide separate sources of power immediately available for operation of these loads. Thus, the power supply system meets the single failure criteria required of the safety systems.

Three (3) gas turbine generators are directly available to the Indian Point site. One is located onsite (GT-1) and two additional units are located at the adjacent Buchanan Substation (GT-2 and GT-3). One gas turbine generator is more than adequate to provide an additional contingency of backup electrical power for maintaining the plant in a safe shutdown condition. The specified gas turbine generator minimum fuel inventory of 54,200 gallons assures that one gas turbine generator will be capable of supplying more than the maximum electrical load for the Indian Point Unit No. 2 alternate safe shutdown power supply system (i.e., 750 kW) for at least three (3) days. Commercial oil supplies and trucking facilities exist to assure deliveries of additional fuel oil within one day's notice.

Conditions of a system-wide blackout could result in a unit trip. Since normal offsite power supplies as required in Specification 3.7.A are not available for startup, it is desirable to be able to blackstart this unit with onsite power supplies as a first step in restoring the system to an operable status and restoring power to customers for essential service. Specification 3.7.D.1 provides for startup using the onsite gas turbine to supply the 6.9 kV loads and the diesels to supply the 480-volt loads. Tie breakers between the 6.9 kV and 480-volt systems are open so that the diesels would not be jeopardized in the event of any incident and would be able to continue to supply 480-volt safeguards power. The scheme consists of starting two reactor coolant pumps, one condensate pump, 2 circulating water pumps and necessary auxiliaries to bring the unit up to approximately 10% power. At this point, loads can be assumed by the main generator and power supplied to the system in an orderly and routine manner.

Specification 3.7.D.2 is identical with normal start-up requirements as in Specification 3.7.A except that offsite power is supplied exclusively from gas turbines with a minimum total power of 37 MW (nameplate rating), which is sufficient to carry out normal plant startup.

As a result of an investigation of the effect components, that might become submerged following a LOCA, may have on ECCS, containment isolation, and other safety-related functions, a fuse and a locked-open circuit breaker were provided on the electrical feeder to emergency lighting panel 218 inside containment. With the circuit breaker in the open position, containment electrical penetration H-70 is de-energized during the accident condition. Personnel access to containment may be required during power operation. Since it is highly improbable that a LOCA would occur during this short period of time, the circuit breaker may be closed during that time to provide emergency lighting inside containment for personnel safety.

When the 138 kV source of offsite power is out of service, the automatic transfer of 6.9 kV Buses 1, 2, 3 and 4 to offsite power after a unit trip could result in overloading of the 20 MVA 13.8 kV/6.9 kV auto-transformer. Accordingly, the intent of Specification 3.7.B.3 is to prevent the automatic transfer when only the 13.8 kV source of offsite power is available. However, this specification is not intended to preclude subsequent manual operations or bus transfers once sufficient loads have been stripped to assure that the 20 MVA auto-transformer will not be overloaded by these manual actions.

References

(1) UFSAR Section 8.2.1
(2) UFSAR Section 8.2.3

3.8 REFUELING, FUEL STORAGE AND OPERATIONS WITH THE REACTOR VESSEL HEAD BOLTS LESS THAN FULLY TENSIONED

Specifications

A. The following conditions shall be satisfied when fuel is in the reactor vessel and the reactor vessel head bolts are less than fully tensioned:

1. Prior to initial movement of the reactor vessel head, the containment purge supply, exhaust and pressure relief isolation valves, including the radiation monitors which initiate isolation, shall be tested and verified to be operable or the inoperable isolation valves locked closed in accordance with Specification 3.8.B.8.

2. The core subcritical neutron flux shall be continuously monitored by two source range monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed (excluding the movement of neutron source bearing assemblies). When core geometry is not being changed, at least one source range neutron flux monitor shall be in service. With both of the required monitors inoperable or not operating, boron concentration of the reactor coolant system shall be determined at least once per 12 hours.

3. At least one residual heat removal (RHR) pump and heat exchanger shall be operable and in operation when water level is greater than or equal to 23 feet (El. 92'0") above the top of the reactor vessel flange.

4. When water level is less than 23 feet above the top of the reactor vessel flange, both RHR pumps and RHR heat exchangers shall be operable with at least one of each in operation.

5. If the requirements of Specification 3.8.A.3 or 3.8.A.4 cannot be satisfied, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR pump(s) and heat exchanger(s) to operable status.

6. The requirements for RHR pump and heat exchanger operability/operation in Specifications 3.8.A.3 and 3.8.A.4 may be suspended during maintenance, modification, testing, inspection, repair or the performance of core component movement in the vicinity of the reactor pressure vessel hot legs. During operation under the provisions of this specification, an alternate means of decay heat removal shall be available when the required number of RHR pump(s) and heat exchanger(s) are not operable. With no RHR pump(s) and heat exchanger(s) operating, the RCS temperature and the source range detectors shall be monitored hourly.

7. The reactor T_{avg} shall be less than or equal to $140^{o}F$.

8. Specification 3.6.A.1 shall be adhered to for reactor subcriticality and containment integrity.

B. With fuel in the reactor vessel and when:

i) the reactor vessel head is being moved, or

ii) the upper internals are being moved, or

iii) loading and unloading fuel from the reactor, or

iv) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed,

the following specifications (1) through (12) shall be satisfied:

1. Specification 3.8.A above shall be met.

2. The minimum boron concentration shall be the more restrictive of either $\geq$2000ppm or that which is sufficient to provide a shutdown margin $\geq$5% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

3. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

4. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 174 hours.

5. A dead-load test shall be successfully performed on the spent fuel pit bridge refueling crane before fuel movement begins. The load assumed by the refueling crane for this event must be equal to or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A thorough visual inspection of the refueling crane shall be made after the dead-load test and prior to fuel handling.

6. The fuel storage building charcoal filtration system must be operating whenever spent fuel movement is taking place within the spent fuel storage areas unless the spent fuel has had a continuous 35-day decay period.

7. Radiation levels in the spent fuel storage area shall be monitored continuously whenever spent fuel movement is taking place in that area.

8. The equipment door, or a closure plate that restricts direct air flow from the containment, and at least one personnel door in the equipment door or closure plate and in the personnel air lock shall be properly closed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

9. Radiation levels in containment shall be monitored continuously.

10. A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place.

11. The minimum water level above the top of the reactor pressure vessel flange shall be at least 23 feet (El. 92'0") whenever movement of spent fuel is taking place inside the containment.

12. If any of the conditions specified above cannot be met, suspend all operations under this specification (3.8.B). Suspension of operations shall not preclude completion of movement of the above components to a safe conservative position.

C. The following conditions are applicable to the spent fuel pit any time it contains irradiated fuel:

1. The spent fuel cask shall not be moved over <u>any</u> region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable. Furthermore, any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall not be moved on or above El. 95′ in the Fuel Storage Building. Additionally, loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool shall not be moved over spent fuel in the spent fuel pit. The weight of installed crane systems shall not be considered part of these loads.

2. The spent fuel storage pit water level shall be maintained at an elevation of at least 93′2″. In the event the level decreases below this value, all movement of fuel assemblies in the spent fuel pool storage pit and crane operations with loads over spent fuel in the spent fuel pit shall cease and water level shall be restored to within its limit within 4 hours.

D. The following conditions are applicable to the spent fuel pit anytime it contains fuel:

1. The spent fuel storage racks are categorized as either Region I or Region II as specified in Figure 3.8-2. Fuel assemblies to be stored in the spent fuel storage racks are categorized as either Category A, B or C based on burnup and enrichment limits as specified in Figure 3.8-3. The storage of Category A fuel assemblies within the spent fuel storage racks is unrestricted. Category B fuel assemblies shall only be stored in Region I or in a Region II spent fuel rack cell with one cell wall adjacent to a non-fuel area (a non-fuel area is the cask area or the area on the outside of a rack next to a wall). Category C fuel assemblies

shall be stored only in Region I. The one exception to this shall be fuel assembly F-65 which shall be stored in Region I or in a Region II spent fuel rack cell with two cell walls adjacent to non-fuel areas.

In the event any fuel assembly is found to be stored in a configuration other than specified, immediate action shall be initiated to:

a. Verify the spent fuel storage pit boron concentration meets the requirements of Specification 3.8.D.2, and

b. Return the stored fuel assembly to the specified configuration.

2. At all times the spent fuel storage pit boron concentration shall be at least 1500 ppm. With the boron concentration less than this value, all fuel movement within the spent fuel storage pit shall cease and immediate action shall be initiated to restore the boron concentration to at least the minimum specified.

3. During operations described in Specification 3.8.B, the spent fuel storage pit boron concentration shall be at least equal to that required in Specification 3.8.B.2. With the boron concentration less than the specified value either:

a. Isolate the spent fuel storage pit from the refueling cavity, or

b. Take actions required by Specification 3.8.B.12.

E. Specification 3.0.1 is not applicable to the requirements of Specification 3.8.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above-specified precautions, and the design of the fuel-handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety[1]. Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin requirements will keep the core subcritical. During refueling, the reactor refueling cavity is filled with borated water. The minimum boron concentration of this water is the more restrictive of either 2000 ppm or else sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold shutdown condition with all rods inserted. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. Periodic checks of refueling water boron concentration ensure the proper shutdown margin. The specifications allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 174 hour decay time following plant shutdown and the 23 feet of water above the top of the reactor vessel flanges are consistent with the assumptions used in the dose calculations for fuel-handling accidents both inside and outside of the containment. The analysis of the fuel handling accident inside of the containment is based on an atmospheric dispersion factor (X/Q) of 5.1×10^{-4} sec/m^3 and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

The spent fuel storage pit water level requirement in Specification 3.8.C.2 provides approximately 24 feet of water above fuel assemblies stored in the spent fuel storage racks.

The fuel enrichment and burnup limits in Specification 3.8.D.1 and the boron requirements in Specification 3.8.D.2 assure the limits assumed in the spent fuel storage safety analysis will not be exceeded.

The requirement that at least one RHR pump and heat exchanger be in operation ensures that sufficient cooling capacity is available to maintain reactor coolant temperature below 140°F, and sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR pumps and heat exchangers operable when there is less than 23 feet of water above the vessel flange ensures that a single failure will not result in a complete loss of residual heat removal capability. With the head removed and at least 23 feet of water above the flange, a large heat sink is available for core cooling, thus allowing adequate time to initiate actions to cool the core in the event of a single failure.

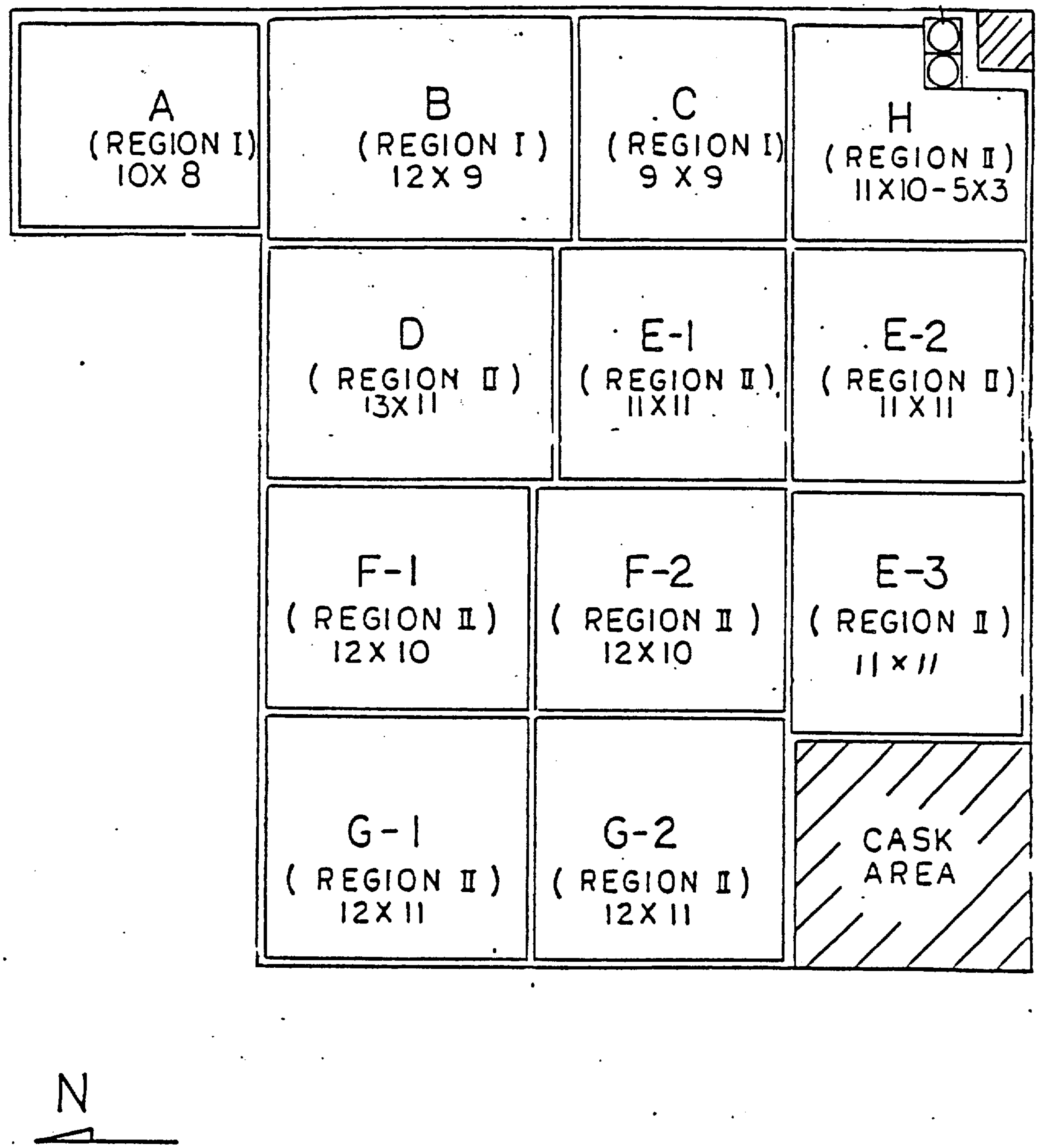
The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the refueling operation during changes in core geometry.

References

(1) FSAR Section 9.5.2

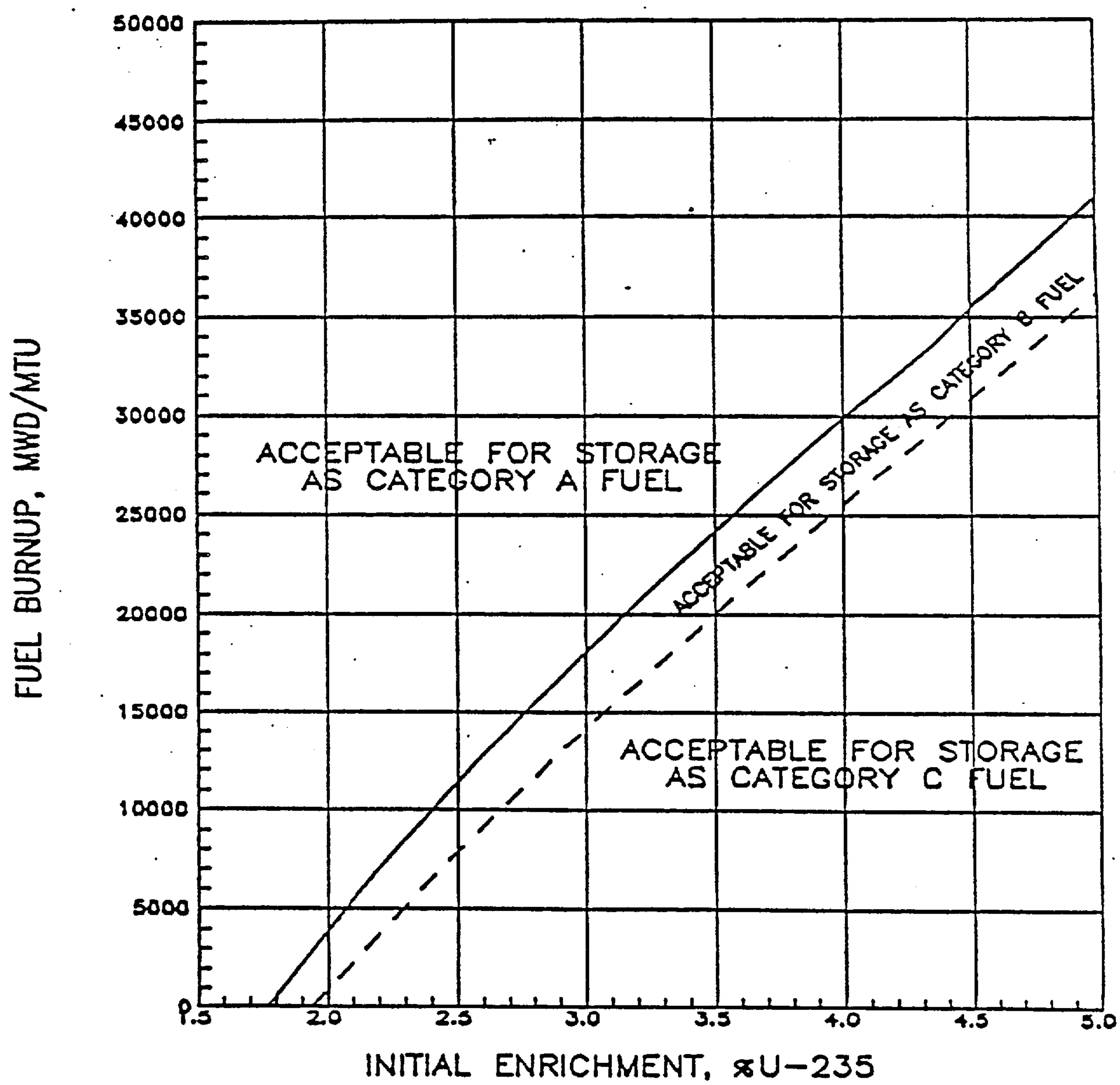
DELETED

Figure 3.8-1



Spent Fuel Storage Rack Layout

Figure 3.8-2



Limiting Fuel Burnup
Versus
Initial Enrichment

Figure 3.8-3

3.9 RADIOACTIVE EFFLUENTS

Applicability

This specification applies to the control of liquid, gaseous and solid radioactive wastes from the facility.

Objective

To define limits and conditions for the controlled release of radioactive materials to the environs such that these releases are as low as reasonably achievable (ALARA) and within allowable regulatory limits.

Specifications

A. RADIOACTIVE LIQUID EFFLUENTS

1. Liquid Effluent Concentration

   a. The concentration of radioactive material released in liquid effluents to unrestricted areas (see Figure 5.1-1) shall be limited to the concentrations specified in 10CFR 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 be limited to 2×10^{-4} microcuries/ml.

   b. With the concentration of radioactive material released in liquid effluents to unrestricted areas exceeding the above limits, without delay restore the concentration to within the the above limits.

2. Liquid Effluent Instrumentation

a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.9-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.A.1.a are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

b. 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 setpoint so it is acceptably conservative.

c. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table 3.9-1. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. Liquid Effluent Dose

a. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from each reactor unit to unrestricted areas (see Figure 5.1-1) shall be limited:

(i) 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

(ii) 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.

b. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the causes(s) for exceeding the limit(s) and defines both the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

4. Liquid Waste Treatment

a. The applicable portions of the liquid radwaste treatment system shall be used as needed to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent, from each reactor unit, to unrestricted areas (see Figure 5.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31-day period.

b. With radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that includes the following information:

(i) Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment of subsystems, and the reason for the inoperability.

(ii) Action(s) taken to restore the inoperable equipment to operable status, and

(iii) Summary description of action(s) taken to prevent a recurrence.

5. Liquid Holdup Tanks

a. The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

   a. Refueling Water Storage Tank
   b. Primary Water Storage Tank
   c. 13, 14 Waste Distillate Storage Tanks
   d. 21, 22, 23 Monitor tanks
   e. Outside temporary tank

b. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, take action within 48 hours to reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

B. RADIOACTIVE GASEOUS EFFLUENTS

1. Gaseous Effluent Dose Rate

a. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary (see Figure 5.1-1) shall be limited to the following:

(i) for noble gases: 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

(ii) for iodine-131, for tritium and for all radionuclides in particulate form with half lives greater than 8 days: less than or equal to 1500 mrem/yr to any organ.

b. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

2. Gaseous Effluent Instrumentation

a. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.9-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.B.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

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

c. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels operable, take the action shown in Table 3.9-2. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. Noble Gases

a. The dose due to noble gases released in gaseous effluents from each reactor unit to areas at and beyond the site boundary (see Figure 5.1-1) shall be limited to the following:

(i) during any calendar quarter: less than or equal to 5 mrem to the whole body from gamma radiation and less than or equal to 10 mrem to the skin from beta radiation, and

(ii) during any calendar year: less than or equal to 10 mrem to the whole body from gamma radiation and less than or equal to 20 mrem to the skin from beta radiation.

b. With the calculated dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

4. Radioiodines, Radioactive Material in Particulate Form, and Radionuclides Other Than Noble Gases

a. The dose to a member of the public from iodine-131, tritium, and all radionuclides in particulate form with half-lives of more than 8 days, in gaseous effluents released from each reactor unit to areas at and beyond the site boundary (see Figure 5.1-1) shall be limited to the following:

(i) during any calendar quarter: less than or equal to 7.5 mrem to any organ, and

(ii) during any calendar year: less than or equal to 15 mrem to any organ.

b. With the calculated dose from the release of iodine-131, tritium, and radionuclides in particulate form with half-lives of more than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

5. <u>Gaseous Waste Treatment System</u>

a. The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the site boundary (see Figure 5.1-1) would exceed 0.2 mrem for gamma radiation and 0.4 mrem for beta radiation in a 31-day period. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the site boundary (see Figure 5.1-1) would exceed 0.3 mrem to any organ in a 31-day period.

b. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that includes the following information:

(i) explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

(ii) action(s) taken to restore the inoperable equipment to operable status, and

(iii) summary description of action(s) taken to prevent a recurrence.

6. Explosive Gas Mixture

a. The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

b. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

c. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

7. Waste Gas Decay Tanks

a. The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 29,761 curies of noble gases (considered as Xe-133).

b. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

C. <u>URANIUM FUEL CYCLE DOSE COMMITMENT</u>

1. The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

2. With the calculated doses from the releases of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.9.A.3.a(i), 3.9.A.3.a(ii), 3.9.B.3.a(i), 3.9.B.3.a(ii), 3.9.B.4.a(i), and 3.9.B.4.a(ii), calculations should be made, including direct radiation contributions from the reactor units and from outside storage tanks, to determine whether the above limits of Specification 3.9.C have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This special report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation dose to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive materials involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition, resulting in violation of 40 CFR Part 190, has not already been corrected, 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.

D. SOLID RADIOACTIVE WASTE

1. The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

2. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

Basis

It is expected that the release of radioactive materials in liquid and gaseous effluents to unrestricted areas will not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as reasonably achievable (ALARA) in accordance with the requirement of 10 CFR 50.36a. While providing reasonable assurance that the design objectives will be met, these Specifications permit the flexibility of operation, compatible with considerations of health and safety, to ensure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that using this operational flexibility under unusual operation conditions, and exerting every effort to keep levels of radioactive materials in liquid and gaseous wastes as low as reasonably achievable, the annual release will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience, taking into account a combination of variables including defective fuel, primary system leakage, primary to secondary system leakage, steam generator blowdown and the performance of the various waste treatment systems, and are consistent with 10 CFR Part 50.36a.

The Indian Point site is a multiple-unit site. There exist shared radwaste treatment systems and shared effluent release points. Where site limits must be met, the effluents of all the units will be combined to determine site compliance. For instances where unit-specific information may be required for radwaste processed or released via a shared system, the effluents shall be proportioned among the units sharing the system(s) in accordance with the methods and agreements set forth in the ODCM.

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas 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 Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-133 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.

This specification applies to the release of liquid effluents from Indian Point Units Nos. 1 and 2.

The radioactive liquid effluent instrumentation, required operable by Specification 3.9.A.2, is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods set forth in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Table II of Appendix B to 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. The purpose of tank level indicating devices is to assure the detection and control of leaks that, if not controlled, could potentially result in the transport of radioactive materials to unrestricted areas.

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 reasonably achievable". Also, for fresh water sites to unrestricted area with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentration in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I 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 ODCM 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.

In addition to the limiting conditions for operation listed under Specification 3.9.A.3.a, the reporting requirements of Specification 3.9.A.3.b specify that the licensee shall identify the cause whenever the dose from the release of radioactive materials in liquid waste effluent exceeds the technical specification limits and describe the proposed program of action to reduce such releases to design objective levels on a timely basis.

Specification 3.9.A.4 requires that the licensee maintain and operate appropriate equipment installed in the liquid waste systems, when necessary, to provide assurance that the releases of radioactive materials in liquid effluents will be kept "as low as reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I to 10 CFR Part 50 for liquid effluents.

The tanks listed in Specification 3.9.A.5 include outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that, in the event of an uncontrolled release of any such tank's contents, the resulting concentration would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

Specification 3.9.B.1 is provided to ensure that the dose at any time beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area to annual average concentration exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For members of the public, who may at times be within the site boundary, the occupancy of that member of the public will usually be sufficiently low to compensate for any increase in the atmosphere 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 a member of the public 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 dose rate above background via the inhalation pathway to less than or equal to 1500 mrem/year.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

The radioactive gaseous effluent instrumentation, required operable by Specification 3.9.B.2, is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous 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. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design criteria 60, 63 and 64 in Appendix A to 10 CFR Part 50.

Specification 3.9.B.3 is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I to 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 to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of

Appendix I 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 dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous 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 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases form Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

Specification 3.9.B.4 is provided to implement the requirements of Section II.C, III.A and IV.A of Appendix I to 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 to unrestricted areas will be kept "as low as 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 of Appendix I 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 ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials 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 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were (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 milch 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.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

Specification 3.9.B.5 requires that the appropriate portions of these systems be used, when specified, to provide reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

Specification 3.9.B.6 is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

The tanks included in Specification 3.9.B.7 are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another technical specification to a quantity that is less than the quantity that provides assurance that, in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem in an event of 2 hours duration.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurances that, in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

Specification 3.9.C is provided to meet the dose limitation of 40 CFR Part 190 that has been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a special report whenever the calculated doses from plant-generated radioactive effluents and direct radiation exceed 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The special report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR Part 190 limits. For the purposes of the special report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contribution from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR Part 190, the special report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.9.A.1 and 3.9.B.1. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

Specification 3.9.D implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the process control program may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constitutents, and mixing and curing times.

Table 3.9-1

Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Channels Operable*	Action
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	(1)	1
b. Steam Generator Blowdown Effluent Line	(1)	2
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	(1)	3
b. Unit 1 - Secondary Boiler Blowdown Purification System (SBBPS) Effluent	(1)	3
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	(1)	4
b. Steam Generator Blowdown Effluent Line	(1)	4
4. TANK LEVEL INDICATING DEVICES**		
a. 13 Waste Distillate Storage Tank	(1)	5
b. 14 Waste Distillate Storage Tank	(1)	5
c. Primary Water Storage Tank	(1)	5
d. Refueling Water Storage Tank	(1)	5
e. 21 Monitor Tank	(1)	5
f. 22 Monitor Tank	(1)	5
g. 23 Monitor Tank	(1)	5

* During release by this pathway, channels shall be operable and in service during such release on a continuous, uninterrupted basis except that outages are permitted, within the time frame and limitations of the specified action, for the purpose of maintenance or required tests, checks and calibration.

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

Table 3.9-1 (Continued)

Radioactive Liquid Effluent Monitoring Instrumentation

Table Notation

ACTION 1 - With the number of channels operable less than required by the Minimum Channels Operable requirement, effluent releases may continue provided that, prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.10.A, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 2 - With the number of channels operable less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 5×10^{-7} microcurie/ml.

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram dose equivalent I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram dose equivalent I-131.

ACTION 3 - With the number of channels operable less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 5×10^{-7} microcurie/ml.

ACTION 4 - With the number of channels operable less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in situ may be used to estimate flow.

ACTION 5 - With the number of channels operable less than required by the Minimum Channels Operable requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

Table 3.9-2

Radioactive Gaseous Effluent Monitoring Instrumentation

Instrument	Minimum Channels Operable	Applicability	Action
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor Providing Alarm	(1)	**	1
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	(1)	**	4
b. Hydrogen or Oxygen Monitor	(1)	**	4
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	3
4. PLANT VENT			
a. Noble Gas Activity Monitor	(1)	* and **	6
b. Iodine Sampler	(1)	*	5
c. Particulate Sampler	(1)	*	5
d. Flow Rate Monitor	(1)	*	2
e. Sampler Flow Rate Monitor	(1)	*	2
5. STACK VENT - UNIT 1			
a. Noble Gas Activity Monitor	(1)	*	3
b. Iodine Sampler	(1)	*	5
c. Particulate Sampler	(1)	*	5
d. Flow Rate Monitor	(1)	*	2
e. Sampler Flow Rate Monitor	(1)	*	2

* Channels shall be operable and in service on a continuous basis during release via this pathway, except that outages are permitted within the time frame of the specified action for the purpose of maintenance and performance of required tests, checks and calibrations.

** During waste gas holdup system operation (treatment for primary system off-gases.)

Table 3.9-2 (Continued)

Radioactive Gaseous Effluent Monitoring Instrumentation

Table Notation

ACTION 1 - With the number of channels operable less than required by the Minimum Channels Operable requirement, the radioactive content of the receiving gas decay tank shall be determined daily to ensure compliance with 3.9.B.1.a.

ACTION 2 - With the number of channels operable less that required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 3 - With the number of channels operable less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 4 - With the number of channels operable less than required by the Minimum Channels Operable requirement, operation of this system may continue provided grab samples are taken and analyzed (1) every 4 hours during degassing operations, and (2) daily during other operations.

ACTION 5 - With the number of channels operable less than required by the Minimum Channels Operable requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.10-3.

ACTION 6 - With the number of channels operable less than required by the Minimum Channels Operable requirement, the contents of the tank(s) may be released to the environment provided that, prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents from waste gas holdup system.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objectives

1. to ensure core subcriticality after reactor trip,

2. to ensure acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. to limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as that shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low-power physics tests, the hot channel factors defined in the basis must meet the following limits:

(a) $F^N_{\Delta H} \leq 1.62\ [1 + 0.3\ (1-P)]$

(b) For $\leq$ 25% steam generator tube plugging:

$$F_Q(Z) \leq (2.32/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.64) \times K(Z) \text{ for } P \leq .5$$

Where P is the fraction of full power at which the core is operating; K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full-power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

3.10.2.2.1 The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated value equal to the ratio of the F_Q or $F^N_{\Delta H}$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.3 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full-power quarter. The target flux difference must be updated each effective full-power month by linear interpolation using the most recent measured value and a value of approximately 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference (defines the band on axial flux difference).

3.10.2.5 At a power level greater than 90% of rated power,

3.10.2.5.1 If the indicated axial flux difference deviates from its target band, the flux difference shall be returned to its target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

3.10.2.6 At a power level no greater than 90 percent of rated power,

3.10.2.6.1 The indicated axial flux difference may deviate from its ± 5% target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power.

3.10.2.6.2 If Specification 3.10.2.6.1 is violated, then the reactor power shall be reduced immediately to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

3.10.2.6.3 A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.7 At a power level no greater than 50 percent of rated power,

3.10.2.7.1 The indicated axial flux difference may deviate from its target band.

3.10.2.7.2 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level ≤ 90% of rated power.

3.10.2.8 Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

3.10.2.9 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests, when the core is operating above 50% of rated thermal power and the indicated quadrant power tilt ratio exceeds 1.02 but is less than or equal to 1.09, within two hours reduce the quadrant power tilt ratio to within its limit or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint three percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

b. Verify that the quadrant power tilt ratio is within its limit within 24 hours after exceeding the limit or restrict core power level to less than 50% of rated thermal power within the next 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ratio exceeds 1.09 with the core operating above 50% of rated thermal power and

a) there is a simultaneous indication of a misaligned control rod, restrict core power level three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0 or until core power level is less than 50% of rated thermal power. If the quadrant power tilt ratio is not within its limit within 2 hours after exceeding the limit, restrict core power level to less than 50% of rated thermal power within the next 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

-or-

b) there is no simultaneous indication of a misaligned control rod, reduce thermal power to less than 50% of rated thermal power within 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02, except as modified in Specification 3.10.10.

3.10.4 Rod Insertion Limits

3.10.4.1 The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin in Figure 3.10-1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits shown in Figure 3.10-3.

3.10.4.3 Control bank insertion shall be further restricted if:

a. The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown,

b. A rod is inoperable (Specification 3.10.7).

3.10.4.4 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-1 must be maintained except for the low-power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one control rod inserted.

3.10.5 Rod Misalignment Limitations

3.10.5.1.1 If a control rod is misaligned from its bank demand position by more than ±12 steps when indicated control rod position is less than or equal to 210 steps withdrawn, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.1.2 If a control rod is misaligned from its bank demand position by more than +17, -12 steps when indicated control rod position is greater than or equal to 211 steps withdrawn, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.2 If the restrictions of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned control rod is not realigned within 8 hours, the rod shall be declared inoperable.

3.10.6 Inoperable Rod Position Indicator Channels

3.10.6.1 A rod position indicator channel shall be capable of determining control rod position within ±12 steps for indicated control rod position less than or equal to 210 steps withdrawn and +17, -12 steps for indicated control rod position greater than or equal to 211 steps withdrawn, or

a. For operation between 50 percent and 100 percent of rating, the position of the control rod shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first.

b. During operation below 50 percent of rating, no special monitoring is required.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3.10.6.3 If a control rod having a rod position indicator channel out of service is found to be misaligned from Specification 3.10.6.1a, above, then Specification 3.10.5 will be applied.

3.10.7 Inoperable Rod Limitations

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5, or which fails to meet the requirements of Specification 3.10.8.

3.10.7.2 Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

3.10.7.3 If any rod has been declared inoperable, then the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time of each control rod shall be no greater than 2.4 seconds from gripper release to dashpot entry.

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.10 Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

Basis

Design criteria have been chosen for normal operations, for operational transients and for those events analyzed in UFSAR Section 14.1 which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must be greater than the safety limits DNBRs in normal operation or in short-term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss-of-coolant accident analysis based on the ECCS acceptance criteria limit of 2200^{o}F. This is required to meet the initial conditions assumed for a loss-of-coolant accident. To aid in specifying the limits on power distribution the following hot channel-factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^E_Q, Engineering Heat Flux Hot Channel Factor is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

The upper bound envelope of the total peaking factor (F_Q) of Specification 3.10.2.1 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss-of-coolant accident analyses based on the specified F_Q times the normalized envelope of Figure 3.10-2 indicate a peak clad temperature of less than 2200°F for the worst case double-ended cold leg guillotine break[1].

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq$ 1.62/1.08. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value (he has no direct control over $F^N_{\Delta H}$) and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests at least each effective full-power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases, including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 12 steps precludes rod misalignment no greater than 15 inches with consideration of maximum instrumentation error for indicated rod position less than or equal to 210 steps withdrawn.

   For indicated control rod positions greater than or equal to 211 steps withdrawn, an indicated misalignment of +17 steps does not exceed the power peaking factor limits. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that, provided the above conditions (1 through 4) are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low-power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full-power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but, expressed as axial offset, it varies only with burnup.

The technical specifications on power distribution control assure that the total peaking factor upper-bound envelope of specified F_Q times Figure 3.10-2 is not exceeded and xenon distributions are not developed which, at a later time, would cause greater local power peaking even though flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the control rod bank more than 190 steps withdrawn (i.e., normal full-power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full-power at which the core was operating, is the full-power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full-power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure 3.10-6 shows the typical variation of the full-power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part-power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these durations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target bank when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target bank; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 to -14 percent (+11 percent to -11 percent indicated) increasing by ± 1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists, the power level is limited to 90 percent or less depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ±5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full-power condition as possible. This is accomplished by using the boron system to position the control rods to produce the required indicated flux difference.

For Condition II events, the core is protected from overpower and a minimum DNBR of less than the safety limit DNBRs by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation as this phenomenon is caused by some asymmetric perturbation, e.g., rod misalignment or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication system or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously but would build up over several hours, and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During startup and power escalation, however, a large tilt could be indicated. Therefore, the specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition of less than the 2% alarm level.

The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt condition. In the event that this tilt condition cannot be eliminated within the two-hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full-core physics map utilizing the movable detector system. For a tilt condition ≤ 1.09, an additional 22-hour time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of three percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event a tilt condition of ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to less than 50% of rated power. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition.

If a tilt ratio greater than 1.09 occurs, which is not due to a misaligned rod, the reactor power level will be reduced to less than 50% of rated power for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each 1 percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. One percent shutdown is adequate except for steam break analysis, which requires more shutdown if the boron concentration is low. Figure 3.10-1 is drawn accordingly.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the power level from full power to zero power is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end-of-life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low-power and zero-power testing is permitted because of the brief period of the test and because special precautions are taken during these tests.

The rod position indicator channel is sufficiently accurate to detect a rod ±7.5 inches away from its demand position for indicated control rod position less than or equal to 210 steps withdrawn. An indicated misalignment ≤ 12 steps does not exceed the power peaking factor limits. A misaligned rod of + 17 steps allows for an instrumentation error of 12 steps plus 5 steps that are not indicated due to the location relationship of the RPI coil stack and the control rod drive rod for indicated rod position greater than or equal to 211 steps withdrawn. The last five steps of rod travel are not indicated by the RPI because the drive rod and spider assembly have been raised three inches (≈5 steps) from rod bottom. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12-step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

Reference

1. UFSAR Section 14.3

FIGURE 3.10-1.
REQUIRED HOT SHUTDOWN MARGIN
versus
REACTOR COOLANT BORON CONCENTRATION

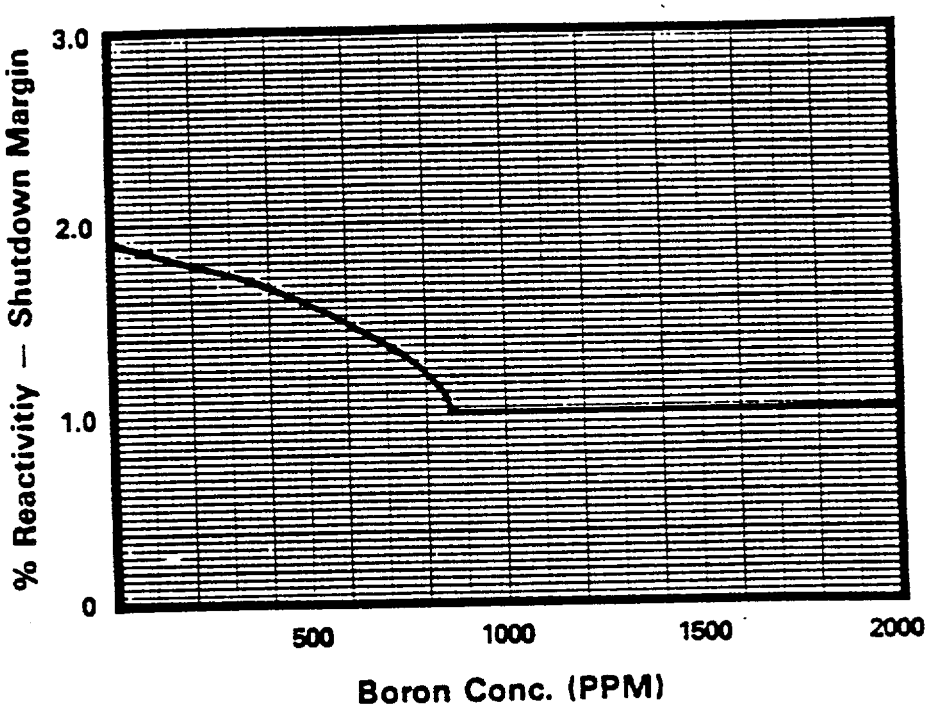


Figure 3.10-2
Hot Channel Factor Normalized Operating Envelope
(For S.G. Tube Plugging Levels up to 25%)

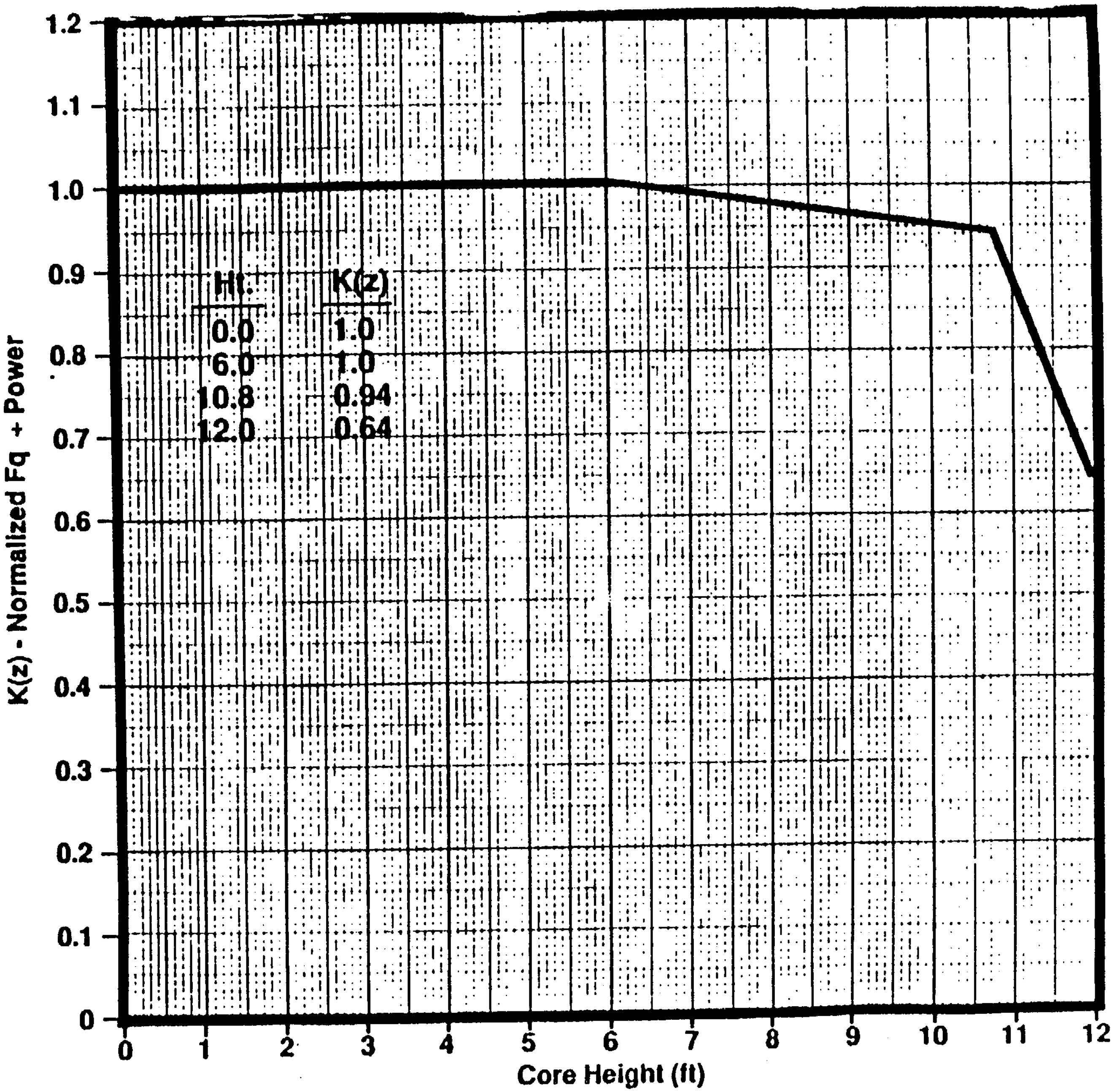


FIGURE 3.10-3.
ROD BANK INSERTION LIMITS
(Four Loop Operation)
100 Step Overlap

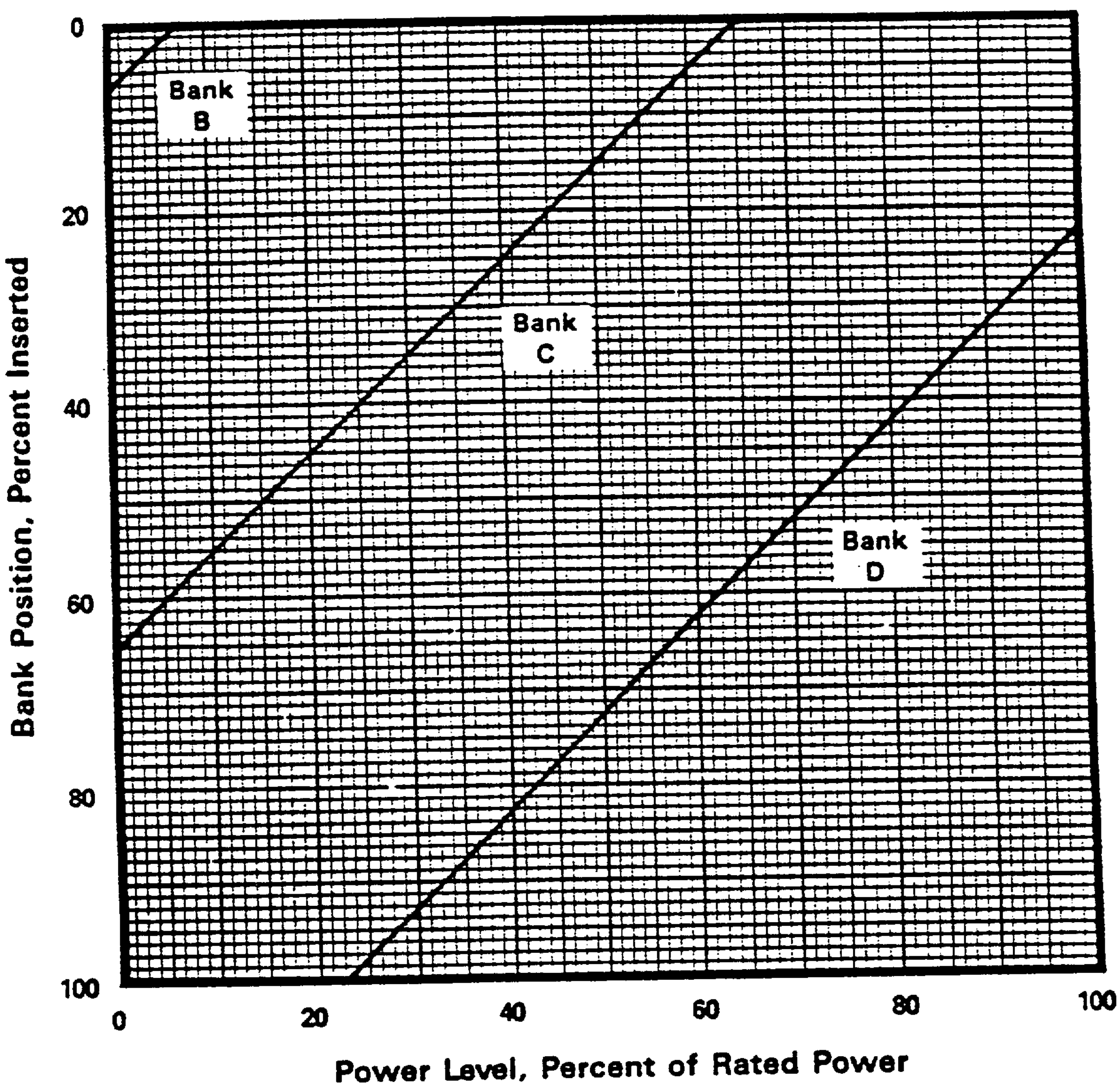


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Figure 3.10-4

Figure 3.10-5
Target Band on Indicated Flux Difference as a Function of Operating Power Level (Typical for BOL)

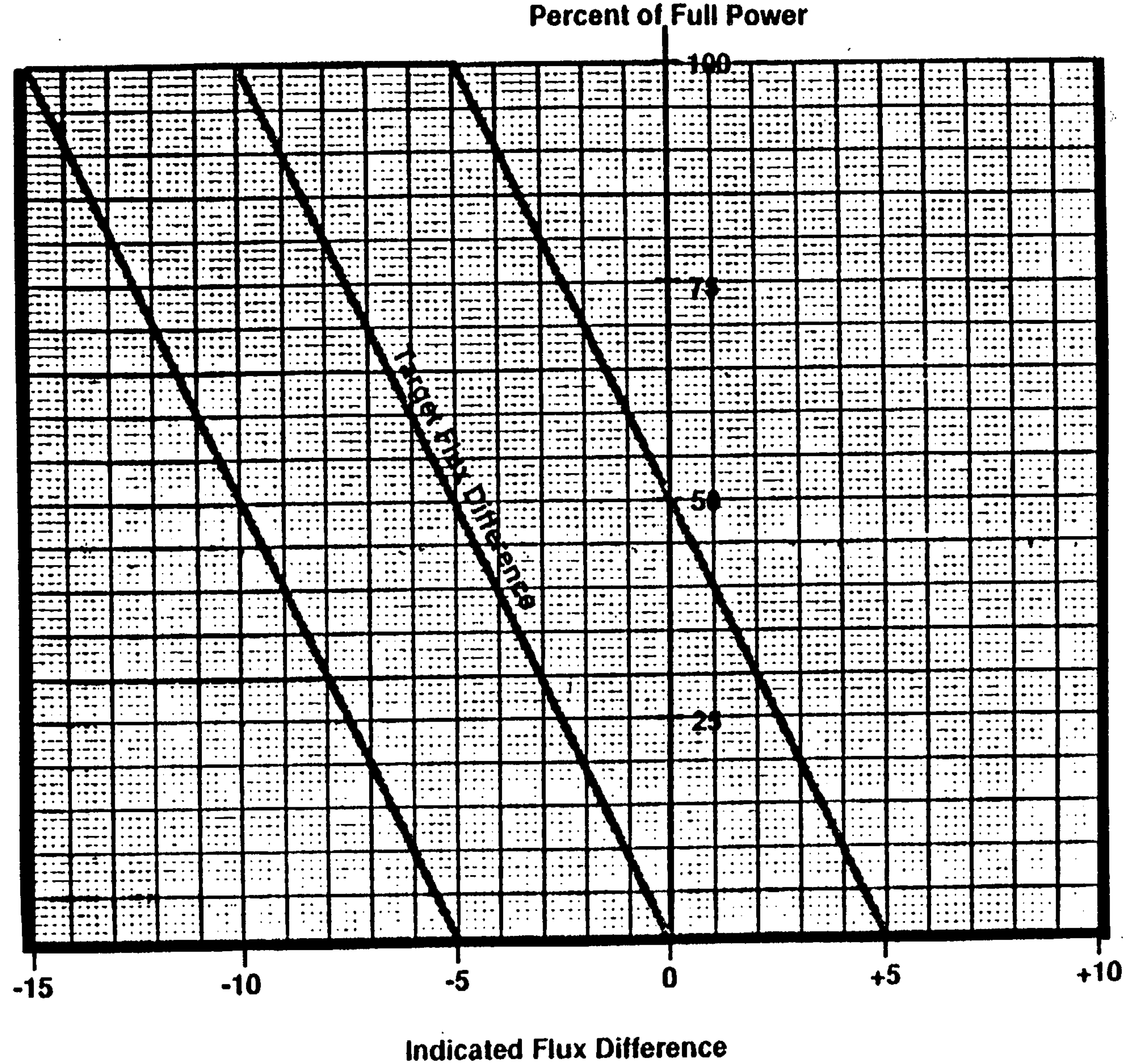
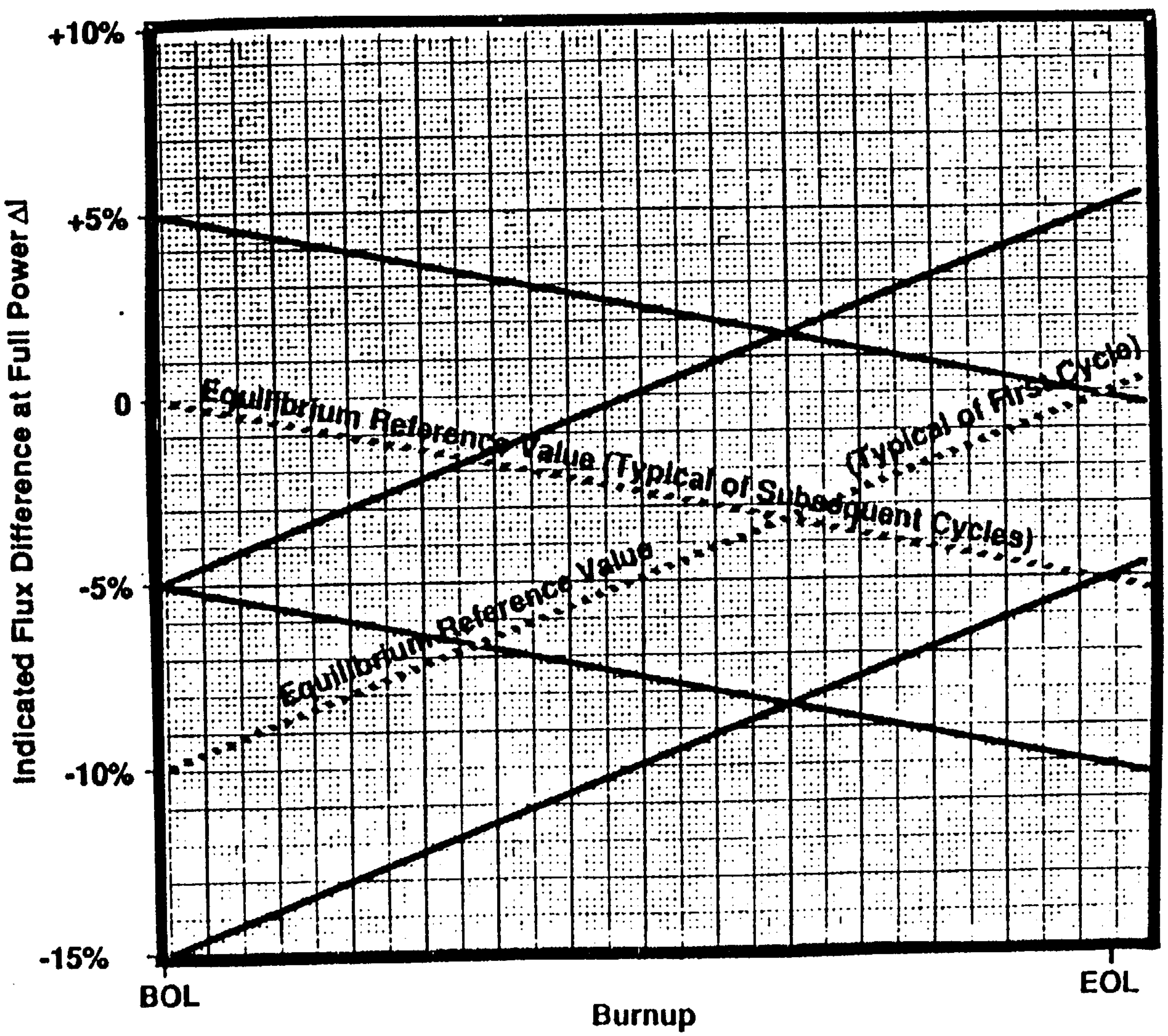


Figure 3.10-6
Permissible Operating Band on Indicated Flux Difference as a Function of Burnup (Typical)



3.11 MOVABLE INCORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the incore instrumentation system, for the recalibration of the excore axial offset detection system.

Specifications

A. A minimum of 2 thimbles per quadrant and sufficient movable in-core detectors shall be operable during re-calibration of the excore axial offset detection system.

B. Power shall be limited to 90% of rated power if re-calibration requirements for the excore axial offset detection system, identified in Table 4.1-1, are not met.

Basis

The Movable Incore Instrumentation System [1] has six drives, six detectors, and 50 thimbles in the core. Each detector can be routed to sixteen or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

To calibrate the excore detectors system, it is only necessary that the Movable Incore System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it will compensate for an error of 10% in the excore protection system. Experience at Beznau No. 1 and R. E. Ginna plants has shown that drift due to changes in the core or instrument channels is very slight. Thus the 10% reduction is considered to be very conservative.

References

(1) UFSAR - Section 7.4

3.12 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the operability of snubbers required for protection of safety-related components.

Objective

To define the time during which reactor operation is permitted after detection of inoperable snubbers.

Specifications

1. All snubbers listed in Table 3.12-1 which are located on systems required for the current mode of operation shall be operable.*

   Snubbers may be added to safety-related systems without prior license amendment to Table 3.12-1 provided that a revision to Table 3.12-1 is included with the next license amendment request.

2. During power operation, the requirements of Specification 3.12.1 may be modified to allow one or more snubbers to be inoperable subject to the following conditions:

   a. The inoperable snubber must be restored to service within 72 hours or the reactor shall be placed in the cold shutdown condition within the succeeding 36 hours.

   b. Either of the following must be performed:

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* Snubber(s) taken out of service for maintenance and testing shall be considered inoperable unless returned to service within 72 hours.

i. An engineering evaluation shall be performed on the supported components within 72 hours of the discovery of the inoperable snubber(s) to determine if the snubber(s) failure has imparted a physical degradation on the supported system. If the supported system is declared inoperable as a result of the evaluation, the appropriate action statement shall be followed.

OR

ii. The supported system shall be declared inoperable within 72 hours of the discovery of the inoperable snubber(s) and appropriate action statements must be followed. If the snubber(s) is repaired or replaced, an engineering evaluation shall be performed on the supported components prior to declaring the system operable.

3. During cold shutdown or refueling, the requirements of Specification 3.12.1 may be modified to allow one or more snubbers to be inoperable subject to the following conditions:

a. The requirements of Specification 3.12.2.b must be met.

b. Snubbers declared inoperable during cold shutdown or refueling shall be made operable or replaced prior to bringing the reactor above cold shutdown.

Basis

Snubbers are required to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping in the event of dynamic loads. It is therefore required that all snubbers required to protect the primary coolant systems or any other safety system or component be operable during reactor operation. Because the snubber protection is required only during low-probability events, a period of 72 hours is allowed for repairs or replacements. Also within that 72-hour period, an engineering evaluation must be performed on the supported system to determine if the snubber(s) failure has imparted a physical degradation on the supported system. If necessary the appropriate action for the system in the Technical Specification shall be taken. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Specification 3.12.3.b prohibits startup if snubbers are known to be inoperable.

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
1	MSR-2V	VC EL 101'-0"	3
1	SR-M4	AFB EL 62'-10"	4
1	SR-M5A	AFB EL 62'-10"	4
1	SR-M5B	AFB EL 62'-10"	4
2	SR-M2	AFB EL 75'-6"	4
2	SR-M3A	AFB EL 75'-6"	4
2	SR-M3B	AFB EL 75'-6"	4
2	SR-M1	AFB EL 75'-6"	4
2	SR-M50	AFB EL 75'-6"	4
2	SR-M51	AFB EL 75'-6"	4
3	MSR-1V	VC EL 101'-0"	3
3	SR-M6	AFB EL 75'-6"	4
3	SR-M7	AFB EL 75'-6"	4
3	SR-M8A (two)	AFB EL 75'-6"	4
3	SR-M8B	AFB EL 75'-6"	4
3	SR-M53	AFB EL 75'-6"	4
4	SR-M9	AFB EL 62'-10"	4
4	SR-M10A	AFB EL 62'-10"	4
4	SR-M10B	AFB EL 62'-10"	4
4	SR-M55	AFB EL 62'-10"	4
4	SR-M56	AFB EL 62'-10"	4
5	SR-B-3	AFB EL 39'-6"	4
6	BF-SR-9	VC EL 59'-6"	3
6	SR-B1	AFB EL 41'-0"	4
7	SR-B7	AFB EL 40'-7"	4

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
8	SR-B5	AFB EL 39'-6"	4
9	SR-55	PAB EL 27'-0"	4
10	SR-65	PAB EL 27'-0"	4
10	SR-807	VC EL 59'-6"	3
10	SR-809	VC EL 59'-6"	3
10	SR-809A	VC EL 59'-6"	3
13	SR-935	VC EL 69'-0"	3
13	SR-936	VC EL 76'-9"	3
13	SR-937	VC EL 84'-0"	3
13	SR-937A	VC EL 84'-0"	3
13	SR-938	VC EL 84'-9"	3
13	SR-939	VC EL 76'-2"	3
13	SR-1027A	VC EL 80'-6"	3
13	SR-1030	VC EL 68'-0"	3
13	SR-1030A	VC EL 68'-0"	3
13	SR-1031	VC EL 76'-0"	3
13	SR-1032	VC EL 76'-8"	3
13	SR-1037	VC EL 83'-7"	3
13	SR-1037A	VC EL 83'-7"	3
13	SR-1051	VC EL 84'-3"	3
13	SR-1052	VC EL 76'-9"	3
13	SR-1053	VC EL 68'-0"	3
13	SR-1060	VC EL 76'-0"	3
13	SR-1079	VC EL 68'-0"	3
13	SR-1080	VC EL 75'-0"	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
13	SR-1099	VC EL 82′-0″	3
13	SR-1100	VC EL 76′-0″	3
13	SR-1103	VC EL 76′-0″	3
13	SR-1104	VC EL 82′-0″	3
13	SR-1105	VC EL 65′-10″	3
13	SR-1106	VC EL 65′-10″	3
13	SR-1124	VC EL 76′-6″	3
14	14-SR-1	VC EL 83′-0″	3
14	SR-925	VC EL 84′-9″	3
14	SR-927	VC EL 81′-6″	3
14	SR-927A	VC EL 81′-6″	3
14	SR-928	VC EL 75′-9″	3
14	SR-928A	VC EL 76′-4″	3
14	SR-969	VC EL 76′-0″	3
14	SR-970	VC EL 75′-0″	3
14	SR-971	VC EL 74′-2″	3
14	SR-1035	VC EL 73′-0″	3
14	SR-1036A	VC EL 76′-0″	3
14	SR-1039A	VC EL 84′-0″	3
14	SR-1040A	VC EL 84′-0″	3
14	SR-1042	VC EL 75′-9″	3
14	SR-1049	VC EL 75′-8″	3
14	SR-1050	VC EL 84′-3″	3
14	SR-1057	VC EL 84′-3″	3
14	SR-1083	VC EL 78′-5″	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
14	SR-1084	VC EL 78′-0"	3
14	SR-1093	VC EL 76′-8"	3
14	SR-1094	VC EL 69′-0"	3
14	SR-1095	VC EL 76′-8"	3
14	SR-1096	VC EL 69′-0"	3
14	SR-954	VC EL 66′-6"	3
14A	SR-1001	VC EL 68′-8"	3
14A	SR-1002	VC EL 69′-0"	3
14A	SR-1075	VC EL 70′-6"	3
14A	SR-1076	VC EL 66′-3"	3
14A	SR-1078	VC EL 70′-3"	3
14A	SR-1120	VC EL 68′-8"	3
14A	SR-1122	VC EL 69′-0"	3
14A	SR-1123	VC EL 68′-8"	3
14A	SR-1077	VC EL 70′-6"	3
16	56-SR-1	VC EL 61′-3"	3
17	17-SR-2	VC EL 58′-0"	3
17	SR-941	VC EL 75′-7"	3
17	SR-941A	VC EL 75′-7"	3
17	SR-1010	VC EL 76′-6"	3
17	SR-1069	VC EL 76′-0"	3
17	SR-1112	VC EL 68′-0"	3
17	SR-1113	VC EL 69′-0"	3
17	SR-1116	VC EL 65′-0"	3
17	SR-1117	VC EL 65′-0"	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
17	SR-1118	VC EL 69'-0"	3
38	38-SR-21	VC EL 60'-0"	3
38	38-SR-22	VC EL 60'-0"	3
38	38-SR-24	VC EL 60'-0"	3
41	SR-952	VC EL 76'-0"	3
41	SR-953	VC EL 68'-0"	3
41	SR-953A	VC EL 65'-0"	3
43	SR-1020A	VC EL 81'-0"	3
43	SR-1024A	VC EL 74'-5"	3
43	SR-1025A	VC EL 69'-11"	3
43	SR-1026	VC EL 68'-9"	3
44	SR-1072	VC EL 68'-3"	3
44	SR-1073	VC EL 68'-7"	3
45	45-SR-9	VC EL 65'-7"	3
45	45-SR-30	PAB EL 64'-0"	4
46	46-SR-2	VC EL 69'-0"	3
46	46-SR-3	VC EL 69'-0"	3
46	46-SR-30	PAB EL 64'-0"	4
47	47-SR-30	PAB EL 64'-0"	4
48	48-SR-30	PAB EL 64'-0"	4
56	56-SR-6	VC EL 55'-6"	3
56	56-SR-12	VC EL 63'-3"	3
56	56-SR-26	VC EL 50'-9"	3
60	SR-73A	PAB EL 71'-0"	4
60	SR-703 (two)	VC EL 55'-0"	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
60	SR-746A	VC EL 64'-6 1/2"	3
60	SR-746B	VC EL 64'-6 1/2"	3
60	SR-746C	VC EL 57'-0"	3
61	SR-887	VC EL 68'-3"	3
62	SR-1	VC EL 70'-6"	3
62	SR-2	VC EL 70'-6"	3
62	SR-3	VC EL 70'-6"	3
62	SR-924	VC EL 70'-0"	3
70	70-SR-3	VC EL 85'-9 1/2"	3
70	70-SR-4	VC EL 66'-10"	3
70	70-SR-5	VC EL 68'-6"	3
70	70-SR-6	VC EL 103'-0"	3
70	70-SR-10	VC EL 123'-9"	3
70	70-SR-11	VC EL 123'-9"	3
70	70-SR-12	VC EL 123'-9"	3
70	70-SR-13	VC EL 123'-9"	3
70	70-SR-14	VC EL 127'-3"	3
70	RCS-5	VC EL 102'-3 3/4"	3
70	RCS-6	VC EL 103'-0"	3
70	70-RCS-5A	VC EL 103'-0"	3
71	71-SR-1	VC EL 80'-0"	3
71	SR-963	VC EL 76'-0"	3
71	SR-964	VC EL 68'-0"	3
71	SR-964A	VC EL 68'-6"	3
71	SR-967A	VC EL 63'-10"	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
72	72-SR-1	VC EL 80′-0"	3
72	SR-1125	VC EL 80′-0"	3
72	SR-1126	VC EL 70′-0"	3
72	SR-1127	VC EL 70′-0"	3
72	SR-1128	VC EL 72′-0"	3
72	SR-1129	VC EL 63′-0"	3
72	SR-1131	VC EL 64′-0"	3
73	SR-1016A	VC EL 76′-7"	3
73	SR-1017	VC EL 69′-1"	3
73	SR-1017A	VC EL 69′-0"	3
73	SR-1017B	VC EL 69′-0"	3
73	SR-1018A	VC EL 69′-0"	3
74	74-SR-1	VC EL 80′-4"	3
74	SR-1085	VC EL 67′-8"	3
74	SR-1086	VC EL 68′-9"	3
74	SR-1087	VC EL 68′-11"	3
74	SR-1087A	VC EL 70′-4"	3
74	SR-1089	VC EL 68′-9"	3
74	SR-1092	VC EL 71′-0"	3
76	76-H-15	VC EL 65′-0"	3
78	78-SR-1	VC EL 70′-6"	3
79	SR-902	VC EL 56′-6"	3
79	SR-907	VC EL 56′-6"	3
79	SR-908	VC EL 56′-6"	3
80	SR-920A	VC EL 58′-8"	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
93	SR-752	VC EL 73'-9"	3
93	SR-752A	VC EL 73'-9"	3
93	SR-753	VC EL 85'-3"	3
94	SR-759	VC EL 85'-3"	3
155	SR-50A	PAB EL 25'-0"	4
163	SR-250	PAB EL 68'-6"	4
163	SR-250A	PAB EL 68'-6"	4
163	163-SR-5	PAB EL 48'-0"	4
250	250-SR-1	PAB EL 110'-0"	4
293	SR-761	VC EL 49'-2"	3
293	SR-763A	VC EL 50'-0"	3
342	342-SR-6	VC EL 103'-0"	3
343	343-SR-5	VC EL 103'-0"	3
344	344-SR-4	VC EL 103'-0"	3
351	PWR-127	VC EL 60'-0"	3
351	PWR-128	VC EL 55'-0"	3
351	PWR-129	VC EL 55'-0"	3
351	351-SR-1	VC EL 55'-0"	3
352	PWR-152	VC EL 66'-6"	3
353	PWR-147A	VC EL 60'-0"	3
353	PWR-148	VC EL 48'-9"	3
353	SR-736	VC EL 47'-3 1/4"	3
353	SR-737	VC EL 48'-9"	3
353	SR-737A	VC EL 58'-6"	3
355	SR-748	VC EL 56'-0"	3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
356	SR-716	VC EL 61'-6"	3
356	SR-718A	VC EL 55'-3"	3
356	SR-720	VC EL 55'-3"	3
358	SR-738B	VC EL 55'-1 3/4"	3
358	SR-738A	VC EL 55'-4 1/8"	3
361	SR-732A	VC EL 65'-0"	3
361	SR-749	VC EL 53'-6"	3
361	SR-749A	VC EL 57'-0"	3
361	SR-749B	VC EL 55'-0"	3
361	SR-749C	VC EL 53'-6"	3
361	SR-755	VC EL 56'-9"	3
361	SR-756	VC EL 71'-6"	3
361	361-SR-10	VC EL 61'-6"	3
518	SR-71A	PAB EL 70'-0"	4
577	577-SR-1	VC EL 65'-1 1/2"	3
577	577-SR-5	VC EL 59'-6"	3
577	577-SR-13	VC EL 54'-11"	3
577	577-SR-15	VC EL 56'-6"	3
577	577-SR-17	VC EL 62'-0"	3
V-2	SR-V20A	AFB EL 55'-2"	4
V-2	SR-V20B	AFB EL 55'-2"	4
V-3	SR-M29	AFB EL 65'-10"	4
V-3	SR-M30	AFB EL 64'-0"	4
V-3	SR-M31	AFB EL 64'-0"	4
V-3	SR-M33	AFB EL 83'-6"	4

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

Line No.	Snubber No.	Location (Approx.)	Category
V-4	SR-M25	AFB EL 78′-6"	4
V-4	SR-M27	AFB EL 76′-0"	4
V-4	SR-M52	AFB EL 76′-0"	4
V-5	SR-M34	AFB EL 76′-3"	4
V-5	SR-M35	AFB EL 73′-6"	4
V-5	SR-M36	AFB EL 73′-6"	4
V-5	SR-M37	AFB EL 76′-3"	4
V-5	SR-M54	AFB EL 76′-3"	4
V-6	SR-M39	AFB EL 64′-2"	4
V-6	SR-M40	AFB EL 64′-2"	4
V-6	SR-M41	AFB EL 64′-1/2"	4
MS-3	SR-499	AFB EL 68′-0"	4
MS-3	SR-501	AFB EL 68′-0"	4
MS-3	SR-503	AFB EL 66′-8"	4
MS-3	MS-SR-129	AFB EL 55′-10"	4
Steam Gen.			
#21	SG21-1 thru SG21-4	VC EL 94′	2, 3
#21	SG21-5 and SG21-6	VC EL 46′	2, 3
#22	SG22-1 thru SG22-4	VC EL 94′	2, 3
#22	SG22-5 and SG22-6	VC EL 46′	2, 3
#23	SG23-1 thru SG23-4	VC EL 94′	2, 3
#23	SG23-5 and SG23-6	VC EL 46′	2, 3
#24	SG24-1 thru SG24-4	VC EL 94′	2, 3
#24	SG24-5 and SG24-6	VC EL 46′	2, 3

Table 3.12-1

Safety-Related Shock Suppressors (Snubbers)

NOTES:

(1) Locations:

AFB	- Aux. Boiler Feed Pump Bldg. and Pipe Bridge Area
PAB	- Primary Auxiliary Building
VC	- Containment Building
SG	- Steam Generator

(2) Categories:

1. Snubber in high radiation area during shutdown.*
2. Snubber especially difficult to remove because of size and/or location.
3. Snubber inaccessible during normal operation because of high radiation and/or temperature environment.*
4. Snubber accessible during normal operation.*

* Modification to this Table due to changes in high radiation areas may be made without prior license amendment provided that a revision to this Table is included with the next license amendment request.

3.13 FIRE PROTECTION AND DETECTION SYSTEMS

Applicability

This specification applies to the operability of fire protection and detection systems provided for protection of safe shutdown systems.

Objective

To assure the operability of fire protection and detection systems.

Specifications

A. HIGH-PRESSURE WATER FIRE PROTECTION SYSTEM

1. The high-pressure water fire protection system shall have:

    a. two (2) main motor-driven fire pumps and one diesel-driven fire pump operable and properly aligned to the high-pressure fire header,

    b. a minimum available water volume of 360,000 gallons contained in the City Water Tank and 300,000 gallons contained in the Fire Water Tank for fire protection purposes, and

    c. all piping and valves necessary for proper functioning of any portion of the system required for protection of safe shutdown systems operable.

2. The requirements of Specification 3.13.A.1 may be modified to allow any one of the following conditions to exist at any one time. If the inoperable equipment is not restored to operable status within the time period specified, then, in lieu of any other report required by 10 CFR 50.73, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b within the next thirty (30) days outlining the plans and procedures to be used for restoring the

inoperable equipment to operable status or for providing an alternate pumping capability or water supply.

a. One or both motor-driven fire pumps and/or one water supply may be out of service provided the inoperable equipment is restored to operable status within seven (7) days.

b. The diesel-driven fire pump and/or one water supply may be out of service provided the inoperable equipment is restored to operable status within seven (7) days.

3. With the high-pressure water fire protection system inoperable in a manner other than permitted by Specification 3.13.A.2:

a. An alternate fire protection system shall be established within 24 hours.

b. In lieu of any other report required by 10 CFR 50.73:

i. The NRC Region I Office shall be notified within 24 hours of identification by telephone and confirm by telegraph, mailgram or facsimile transmission no later than the first working day following the event; and

ii. A special report shall be submitted in accordance with Specification 6.9.2.b within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

c. If the requirement of 3.13.A.3.a cannot be satisfied within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirement of 3.13.A.3.a cannot be satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

B. FIRE PROTECTION SPRAY SYSTEMS

1. The following spray systems shall be operable whenever equipment in the area is required to be operable:

   a. Electrical Tunnel Fire Protection Water Spray System (El-33′ in Control Building to El-68′ in PAB),

   b. Diesel Generator Building Water Spray System (El-67′ in D.G. Building), and

   c. Containment Fan Cooler Charcoal Filter Dousing System (El-68′ in Containment).

2. If the requirements of 3.13.B.1 cannot be satisfied:

   a. Additional equivalent capacity fire hose(s) shall be routed to the affected area(s) from an operable hose station or hydrant within one (1) hour.

   b. The inoperable equipment shall be restored to operable status within 14 days or, in lieu of any other report required by 10 CFR 50.73, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b within the next 30 days outlining the cause of the inoperability and the plans for restoring the equipment to operable status.

C. PENETRATION FIRE BARRIERS

1. The following penetration fire barriers shall be functional at all times:

   a. penetration fire barriers between the central control floor and the cable spreading room,

b. penetration fire barriers between the 480 V switchgear room and the cable spreading room, and

c. penetration fire barrier between the PAB and the electrical tunnel.

2. If the requirements of 3.13.C.1 cannot be satisfied:

a. Within one (1) hour, either a continuous fire watch shall be established on at least one side of the affected penetration(s), or the operability of fire detectors on at least one side of the non-functional fire barrier(s) shall be verified and an hourly fire watch patrol shall be established.

b. The non-functional penetration fire barrier(s) shall be restored to functional status within seven (7) days or, in lieu of any other report required by 10 CFR 50.73, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b within the next 30 days outlining the cause of the malfunction and the plans for restoring the barrier(s) to functional status.

D. FIRE DETECTION SYSTEMS

1. As a minimum, the fire detection instrumentation for each location shown in Table 3.13-1 shall be operable whenever equipment in that location is required to be operable.

2. With the number of operable fire detection instruments less than the minimum required by Table 3.13-1:

a. For instruments outside containment, a fire watch patrol shall be established within 1 hour to inspect the affected location(s) at a frequency of at least once per hour.

b. For instruments inside containment, either a fire watch patrol shall be established to inspect the affected location(s) at least once per eight (8) hours, or the containment air temperature shall be monitored at least once per hour.

c. The minimum operable instrumentation required in Table 3.13-1 shall be restored within 14 days or, in lieu of any other report required by 10 CFR 50.73, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b within the next 30 days outlining the cause of the malfunction and the plans for restoring the instrumentation to operable status.

E. FIRE HOSE STATIONS AND HYDRANTS

1. The fire hose stations and fire hydrants shown in Table 3.13-2 shall be operable whenever safety-related equipment in the areas protected by the hose stations and hydrants is required to be operable.

2. If the requirements of 3.13.E.1 cannot be satisfied:

   a. Additional equivalent capacity fire hose(s) shall be routed to the affected area(s) from an operable hose station or hydrant within one (1) hour.

   b. The inoperable spray system(s) shall be restored to operable status within 14 days or, in lieu of any other report required by 10 CFR 50.73, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b within the next 30 days outlining the cause of inoperability and the plans for restoring the spray system(s) to operable status.

F. CABLE SPREADING ROOM HALON SYSTEM

1. The Cable Spreading Room Halon System shall be operable at all times with the halon storage tanks having an equivalent of at least 95% of full charge weight and an equivalent of at least 90% of full charge pressure at standard temperature and pressure (STP) conditions.

2. If the requirements of 3.13.F.1 cannot be satisfied:

a. A continuous fire watch with backup fire protection equipment shall be established within 1 hour for the Cable Spreading Room.

b. The Cable Spreading Room Halon System shall be restored to operable status within 14 days or, in lieu of any other report required by 10 CFR 50.73, a special report shall be prepared and submitted to the Commisson pursuant to Specification 6.9.2.b within the next 30 days outlining the cause of the inoperability and the plans for restoring the Halon System to operable status.

Basis

These specifications are established to assure the operability of fire protection and detection systems provided to protect equipment utilized for safe shutdown of the unit. The fire protection and detection systems are described in Revision 1 to "Review of the Indian Point Station Fire Protection Program," submitted to the NRC by letter dated April 15, 1977, and also in the Fire Protection Safety Evaluation Report issued by the NRC Regulatory Staff in conjunction with Amendment No. 46 to DPR-26 on January 31, 1979.

Table 3.13-1

Fire Detection Instruments

Instrument Location	Minimum Instruments Operable: Heat	Minimum Instruments Operable: Smoke (ionization detectors)
1. Central Control Room (Control Building: El-53′)	N/A	4
2. Cable Spreading Room (Control Building: El-33′)	N/A	7
3. Switchgear Room (Control Building: El-15′)	N/A	7
4. Electrical Tunnel (El-33′ to El-68′)	38*	3
5. Electrical and Piping Tunnel and Piping Penetration Area (PAB and Fan House: El-68′ to El-51′)	N/A	2
6. Electrical Penetration Area (Fan House: El-46′)	N/A	4
7. Diesel Generator Building (El-67′)	11	N/A
8. Boric Acid Transfer Pumps Area (PAB: El-80′)	N/A	1
9. Containment Spray Pump/Primary Water Makeup Pump Area (PAB: El-68′)	N/A	4
10. Containment Fan Cooler Units (Containment: El-68′)	4 per FC Unit	N/A
11. Electrical Penetration Area Outer Annulus (Containment: El-46′)	N/A	3
12. Auxiliary Boiler Feedwater Pump Area (AFB: El-18′)	N/A	2

* Temperature detector/trip devices

Table 3.13-1

Fire Detection Instruments

Instrument Location	Minimum Instruments Operable Heat	Smoke (ionization detectors)
13. Main Corridor (PAB: El-80′)	N/A	5
14. Main Corridor (PAB: El-98′)	N/A	3
15. Component Cooling Pump Area (PAB: El-68′)	N/A	1
16. RHR Pump 21 Room (PAB: El-15′)	N/A	1
17. RHR Pump 22 Room (PAB: El-15′)	N/A	1
18. Safety Injection Pump Area (PAB: El-59′)	N/A	1
19. Charging Pump 21 Room (PAB: El-80′)	N/A	1
20. Charging Pump 22 Room (PAB: El-80′)	N/A	1
21. Charging Pump 23 Room (PAB: El-80′)	N/A	1
22. Reactor Coolant Pumps (Containment: El-93′)	N/A	2 per RCP

Table 3.13-2

Fire Hose Stations and Fire Hydrants

Hose Station or Hydrant	Location
1. Hose Station	Stairwell No. 3 - Control Building: El-15'
2. Hose Station	Stairwell No. 3 - Control Building: El-33'
3. Hose Station	Stairwell No. 3 - Control Building: El-53'
4. Hose Station	Stairwell No. 4 - Control Building: El-53'
5. Hose Station	East End of PAB: El-98'
6. Hose Station	West End of PAB: El-98'
7. Hose Station	East End of PAB: El-80'
8. Hose Station	West End of PAB: El-80'
9. Hose Station	West End of PAB: EL-68'
10. Hose Station	West End of PAB: El-15'
11. Hose Station	Piping Penetration Area - PAB: EL-54'
12. Hose Station	Southeast End of FSB: El-96'
13. Hose Station	Southeast End of FSB: El-140'
14. Hose Station	North Side of Containment: El-95'
15. Hose Station	South Side of Containment: El-95'
16. Hose Station	North Side of Containment: El-46'
17. Hose Station	South Side of Containment: El-46'

Table 3.13-2

Fire Hose Stations and Fire Hydrants

Hose Station or Hydrant		Location
18.	Hydrant No. 25 and Associated Hose House (with fire hose and nozzles to serve the AFW Building).	Yard Area - South of the AFB: El-15'
19.	Hydrant No. 27 and Associated Hose House (with fire hose and nozzles to serve the DG Building).	Yard Area - South of the DGB: EL-67'

3.14 HURRICANE ALERT

Applicability

Applies to a hurricane with winds in excess of 87 knots, when a Hurricane Warning has been issued for any coastal area south of Indian Point or east of Indian Point as far east as New Haven, Connecticut.

Objective

To define actions permitted after receipt of Hurricane Warnings.

Specifications

3.14.a If the National Weather Service issues a Hurricane Warning for a hurricane with wind in excess of 87 knots (approximately 100 mph) within 500 nautical miles of the facility, a prompt report shall be made to the NRC Incident Response Center within 1 hour of receipt of that Hurricane Warning. This notification is in lieu of the reporting requirements of 10 CFR 50.73.

3.14.b If the National Weather Service issues a Hurricane Warning for a hurricane with winds in excess of 87 knots within 320 nautical miles of the facility and a Hurricane Warning is in effect for any coastal area south of Indian Point or any coastal area east of Indian Point as far east as New Haven, Connecticut: the hurricane direction, translational velocity and average wind speed shall be monitored at least every hour and the Unit shall be placed in the hot shutdown condition within four (4) hours. Appropriate action shall be taken to ensure that the plant is in the cold shutdown condition prior to arrival on site of a hurricane with winds in excess of 87 knots.

3.15 METEOROLOGICAL MONITORING SYSTEM

Applicability

This specification applies to the operability of the meteorological monitoring system.

Objective

To define limits and conditions for the measurement of meteorological conditions at the site.

Specifications

A. The meteorological monitoring instrumentation channels shown in Table 3.15-1 shall be operable at all times with indication of the tabulated parameters available in the control room.

B. With one or more of the required meteorological monitoring channels inoperable for more than seven (7) consecutive days, prepare and submit to the Commission within the next 10 days, pursuant to Specification 6.9.2.g, a special report, in lieu of any other report, outlining the cause of the malfunction(s) and the plans for restoring the channel(s) to operable status.

Basis

Operability of the meteorological monitoring system instrumentation ensures that sufficient meteorological data at the site is available for estimating potential radiation doses to the public as a result of routine or accidental releases 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, Rev. 0.

3.16 REACTOR COOLANT SYSTEM VENTS

Applicability

Applies to the operability of the reactor coolant system vents.

Objective

To define those limiting conditions for operation that are necessary to ensure the ability to exhaust noncondensible gases from the primary coolant system.

Specifications

A. Whenever the reactor coolant system is above 350°F, the reactor vessel head vent and at least one of the two pressurizer steam space vents shall be operable with associated valve positions as follows:

1. Reactor Vessel Head Vent:
   (a) HCV-3100 and HCV-3101 closed
   (b) Manual valve 500 open

2. Pressurizer Steam Space Vent No. 1:
   (a) PCV-455C closed
   (b) MOV-535 open or closed

3. Pressurizer Steam Space Vent No. 2:
   (a) PCV-456 closed
   (b) MOV-536 open or closed

In addition, the PORVs (PCV-455C and 456) may be operated as necessary beyond their use as a reactor coolant system vent in accordance with approved plant procedures.

B. With the reactor vessel head vent inoperable or both pressurizer steam space vents inoperable*, startup and/or power operation may continue provided the inoperable vent(s) is (are) maintained closed with power removed from the valve actuator of all the valves in the inoperable vent(s). Restore either the reactor vessel head vent or one of the two pressurizer steam space vents respectively to operable status to meet the requirements of 3.16.A within 30 days, or the reactor shall be placed in the hot shutdown condition within the next 6 hours, and in cold shutdown within the following 30 hours.

C. With all of the above reactor coolant system vents inoperable*, maintain the inoperable vents closed with power removed from the valve actuators of all the valves in the inoperable vents, restore one vent to operable status within 72 hours and apply the requirements of Specification 3.16.B. If this cannot be accomplished, the reactor shall be placed in the hot shutdown condition within the next 6 hours, and in cold shutdown within the following 30 hours.

Basis

Reactor Coolant System Vents are provided to exhaust noncondensible gases from the primary coolant system. The operability of two reactor coolant system vents from the reactor vessel head and the pressurizer steam space ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vents serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

Reference

UFSAR Section 4.2.10

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* The requirements of Specification 3.1.A.4 shall also be adhered to.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Interval Extension

Unless otherwise noted, each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified interval. Excluded from this provision are the following surveillances whose intervals are solely defined by the applicable Technical Specification paragraphs and cannot be extended.

4.2.1 Inservice Testing - Those tests with a current two year interval whose basis is 10 CFR 50, Appendix J.

4.4A Integrated Leakage Rate

4.4B Sensitive Leakage Rate

4.4C Containment Isolation Valves.

Basis

Specification 4.0.1 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 outages. The limitation of Specification 4.0.1 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.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specifications

a. Calibration, testing and checking of analog channels, and testing of logic channels shall be performed as specified in Table 4.1-1.

b. Sampling and equipment tests shall be conducted as specified in Tables 4.1-2 and 4.1-3, respectively.

c. Performance of any surveillance test outlined in these specifications is not immediately required if the plant condition is the same as the condition into which the plant would be placed by an unsatisfactory result of that test. Such tests will be performed before the plant is removed from the subject condition that has precluded the immediate need to run the test. If the test provisions require that a minimum higher system condition must first be established, the test will be performed promptly upon achieving this minimum condition. The following surveillance tests, however, must be performed without the above exception:

- Table 4.1-1 Items 3 and 19
- Table 4.1-2 Items 1, 2, and 10
- Table 4.1-3 Items 2 and 6

Basis

A surveillance test is intended to identify conditions in a plant that would lead to a degradation of reactor safety. Should a test reveal such a condition, the Technical Specifications require that either immediately, or after a specified period of time, the plant be placed in a condition which mitigates or eliminates the consequences of additional related casualties or accidents. If the plant is already in a condition which satisfies the failure criteria of the test, then plant safety is not compromised and performance of the test yields information that is not necessary to determine safety limits or limiting conditions for operation of the plant. The surveillance test need not be performed, therefore, as long as the plant remains in this condition. However, this surveillance test should be performed prior to removing the plant from the subject condition that has precluded the immediate need to run the test. In the situation in which the test provisions specify that the test must be performed at some minimum system condition, this condition will first be achieved and the test will be performed promptly thereafter prior to proceeding to a higher system condition.

a. CHECK

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, the minimum checking frequency of once per shift when the plant is in operation, is deemed adequate for reactor and steam system instrumentation.

b. CALIBRATION

Calibrations are performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once each refueling shutdown for the process system channels is considered acceptable.

c. TESTING

The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hrs. per channel. This is based on operating experience at conventional and nuclear plants. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = \frac{W - Q\left(\frac{W}{N-M+2}\right)}{W} = 1 - \frac{N!}{(N-M+2)!\,(M-1)!}(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999968, assuming a channel failure rate, λ, equal to 2.5×10^{-6} hr^{-1} and a test interval, W, equal to 720 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one month is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M* (3)	M (2)	1) Heat balance calibration 2) Signal to delta T; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset.
2.	Nuclear Intermediate Range	S (1)	N.A.	S/U**(2)	1) Once/shift when in service Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A.	S/U**(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R	M (1)	1) Overtemperature - delta T 2) Overpower - delta T
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure (High & Low)	S	R	M	
8.	6.9 kV Voltage & Frequency	N.A.	R	M	Reactor Protection circuits only
9.	Analog Rod Position	S	R	M	

* By means of the movable incore detector system.

** Prior to each reactor startup if not done previous week.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	W	R	N.A.	Bubbler tube rodded during calibration
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	DELETED				
17.	Volume Control Tank Level	N.A.	R	N.A.	
18a.	Containment Pressure	D	R	M	Wide Range
18b.	Containment Pressure	S	R	M	Narrow Range
18c.	Containment Pressure (PT-3300,PT-3301)	M	R	N.A.	High Range
19.	Process and Area Radiation Monitoring Systems	D	R	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A.	

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
21a.	Containment Sump and Recirculation Sump Level (Discrete)	S	R	R	Discrete Level Indication Systems.
21b.	Containment Sump, Recirculation Sump and Reactor Cavity Level (Continuous)	S	R	R	Continuous Level Indication Systems.
21c.	Reactor Cavity Level Alarm	N.A.	R	R	Level Alarm System
21d.	Containment Sump Discharge Flow	S	R	M	Flow Monitor
21e.	Containment Fan Cooler Condensate Flow	S	R	M*	
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Steam Line Pressure	S	R	M	
24.	Turbine First Stage Pressure	S	R	M	
25.	Reactor Trip Logic Channel Testing	N.A.	N.A.	M#	
26.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	

* Monthly visual inspection of condensate weirs only.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
27.	Turbine Trip a. Low Auto Stop Oil Pressure	N.A.	R	N.A.	
28.	Control Rod Protection (for use with LOPAR fuel)	N.A.	R	*	
29.	Loss of Power a. 480v Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480v Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
	c. 480v Emergency Bus Undervoltage (Alarm)	N.A.	R	M	
30.	Auxiliary Feedwater a. Steam Generator Water Level (Low-Low)	S	R	R	
	b. Low-Low Level AFWS Automatic Actuation Logic	N.A.	N.A.	M	Test one logic channel per month on an alternating basis.
	c. Station Blackout (Undervoltage)	N.A.	R	R	

* Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to T_{cold} decreasing below 350^{o}F and the breakers are maintained opened during RCS cooldown when T_{cold} is less than 350^{o}F.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
	d. Trip of Main Feedwater Pumps	N.A.	N.A.	R	
31.	Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.	
32.	PORV Position Indicator (Limit Switch)	M	R	R	
33.	PORV Block Valve Position Indicator (Limit Switch)	M*	R	R	
34.	Safety Valve Position Indicator (Acoustic Monitor)	M	R	R	
35.	Auxiliary Feedwater Flow Rate	M	R	R	
36.	PORV Actuation/ Reclosure Setpoints	N.A.	R	N.A.	
37.	Overpressure Protection System (OPS)	N.A.	R	**	

* Except when block valve operator is deenergized.

** Within 31 days prior to entering a condition in which OPS is required to be operable and at monthly intervals thereafter when OPS is required to be operable.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
38.	Wide Range Plant Vent Noble Gas Effluent Monitor (R-27)	S	R	N.A.	
39.	Main Steam Line Radiation Monitor (R-28, R-29, R-30, R-31)	S	R	N.A.	
40.	High Range Containment Radiation Monitor (R-25, R-26)	S	R*	N.A.	
41.	Containment Hydrogen Monitor	Q	Q**	N.A.	

* Acceptable criteria for calibration are provided in Table II.F-13 of NUREG-0737.

** Calibration will be performed using calibration span gas.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
42.	Manual Reactor Trip	N.A.	N.A.	R	Includes: 1) Independent verification of reactor trip and bypass breakers undervoltage trip circuit operability up to and including matrix contacts of RT-11/RT-12 from both manual trip initiating devices, 2) independent verification of reactor trip and bypass breaker shunt trip circuit operability through trip actuating devices from both manual trip initiating devices.
43.	Reactor Trip Breaker	N.A.	N.A.	M#	Includes independent verification of undervoltage and shunt trip attachment operability.
44.	Reactor Trip Bypass Breaker	N.A.	N.A.	M#	Includes: 1) Automatic undervoltage trip, 2) Manual shunt trip from either the logic test panel or locally at the switchgear prior to placing breaker into service.
45.	Service Water Inlet Temperature Monitoring Instrumentation	S	R	A	The test shall take place prior to T.S. 3.3.F.b Applicability.

Each train shall be tested at least every 62 days on a staggered test basis (i.e., one train per month).

Table 4.1-2

Frequencies for Sampling Tests

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Gross Activity (1) Radiochemical(2) $\bar{E}$ Determination Tritium Activity F, Cl & O_2	5 days/week (1) Monthly Semi-annually (3) Weekly (1) Weekly	3 days 45 days 30 weeks 10 days 10 days
2.	Reactor Coolant Boron	Boron Concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 days
4.	Boric Acid Tank	Boron Concentration	Twice/week	5 days
5.	DELETED			
6.	Spray Additive Tank	NaOH Concentration	Monthly	45 days
7.	Accumulator	Boron Concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron Concentration	Monthly	45 days
9.	Secondary Coolant	Iodine-131	Weekly (4)	10 days
10.	Containment Iodine-Particulate Monitor or Gas Monitor	Iodine-131 and Particulate Activity or Gross Gaseous Activity	Continuous When Above Cold Shutdown(5)	NA*

Table 4.1-2

Frequencies for Sampling Tests

FOOTNOTES:

* N.A. - Not Applicable

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half-life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) $\bar{E}$ determination will be started when the gross analysis indicates $\geq$ 10 μCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.D.

(4) When the iodine-131 activity exceeds 10% of the limit in Specification 3.4.A, the sampling frequency shall be increased to a minimum of once each day.

(5) Except as indicated in Specification 3.1.F.

Table 4.1-3

Frequencies for Equipment Tests

		Check	Frequency	Maximum Time Between Tests
1.	Control Rods	Rod drop times of all control rods	Each refueling shutdown	*
2.	Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations	*
3.	Pressurizer Safety Valves	Setpoint	Each refueling shutdown	*
4.	Main Steam Safety Valves	Setpoint	Each refueling shutdown	*
5.	Containment Isolation System	Automatic Actuation	Each refueling shutdown	*
6.	Refueling System Interlocks	Functioning	Each refueling shutdown prior to refueling operation	Not Applicable
7.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days
8.	Turbine Steam Stop Control Valves	Closure	Monthly**	45 days**
9.	Cable Tunnel Ventilation Fans	Functioning	Monthly	45 days

* See Specification 1.9.

** This test may be waived during end-of-cycle operation when reactor coolant boron concentration is equal to or less than 150 ppm, due to operational limitations.

4.2 INSERVICE INSPECTION AND TESTING

Applicability

Applies to the inservice inspection of Quality Group* A, B, and C components and the inservice testing of pumps and valves whose function is required for safety.

Objective

To provide assurance of the continued integrity and/or operability of those structures, systems, and components to which this specification is applicable.

Specifications

4.2.1 Inservice Testing

Inservice testing of pumps and valves whose function is required for safety shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50, Section 50.55a(g), except where specific written relief pursuant to 10 CFR 50, Section 50.55a(g)(6)(i) has been granted.

4.2.2 Inservice Inspection

Inservice inspection of Quality Group* A, B, and C components shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as Required by 10 CFR 50, Section 50.55a(g), except where specific written relief pursuant to 10 CFR 50, Section 50.55a(g)(6)(i) has been granted.

* Quality Group classification is in accordance with Revision 3 of Regulatory Guide 1.26.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations schedules are shown in Table 4.2-1.

4.2.4 Reactor Vessel Special Inspection

1. Interval of Inspection:

The reactor vessel shall be examined during the second ten year interval in the area of the vessel weld located approximately 236 inches below the reactor vessel flange at 345^{o} azimuth. This area shall be re-examined during the three successive inspection periods as defined in accordance with IWB-2410 of the 1980 ASME Boiler and Pressure Vessel Code, Section XI, as modified below.

The examination schedule may revert to the original inspection schedule per IWB-2410 if:

(i) The additional examinations reveal that the indications remain essentially unchanged over 3 successive inspections, or

(ii) Any additional examination utilizing-ultrasonic techniques per IWA-2232, or alternative techniques per IWA-2240, as supplemented by prior examination, demonstrate that the reflector meets the acceptance standards of IWB-3510. Such demonstration shall be submitted for NRC review and approval. Upon receipt of NRC concurrence, this special inspection requirement (4.2.4 in its entirety) shall become void.

2. Reporting Requirements:

The reactor vessel inservice inspection program shall be forwarded to NRC 180 days prior to plant shutdown during which the inspection is scheduled to be accomplished. Inspection results shall be forwarded for NRC review and approval 15 days prior to plant startup.

References

(1) Letter from Robert W. Reid of NRC to William J. Cahill of Consolidated Edison dated April 22, 1976

(2) Letter from Robert W. Reid of NRC to William J. Cahill of Consolidated Edison dated November 17, 1976

(3) Letter from William J. Cahill of Consolidated Edison to Robert W. Reid of NRC dated May 27, 1976

Table 4.2-1

Inservice Inspection Requirements for Indian Point No. 2

No. Item	Examination Category	Components and Part to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
4.2.3	N/A	Primary Pump Flywheel	V & UT	See Remarks	The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods.
4.2.4	N/A	Reactor Vessel Special Inspection Area	UT	See Remarks	The reactor vessel shall be examined during the second ten year interval in the area of the vessel weld located approximately 236 inches below the reactor vessel flange at 345^{o} azimuth. This area shall be reexamined during the three successive inspection periods as defined in accordance with IWB-2410 of the 1980 ASME Boiler & Pressure Vessel Code, Section XI as modified below. The examination schedule may revert to the original inspection schedule per IWB-2410 if: (i) The additional examinations reveal that the indications remain unchanged over 3 successive inspections, or

Table 4.2-1

Inservice Inspection Requirements for Indian Point No. 2

No. Item	Examination Category	Components and Part to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
					(ii) Any additional examination utilizing ultrasonic techniques per IWA-2232, or alternative techniques per IWA-2240, as supplemented by prior examination, demonstrate that the reflector meets the acceptance standards of IWB-3510. Such demonstration shall be submitted for NRC review and approval. Upon receipt of NRC concurrence, this special inspection requirement shall become void.

4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specifications

a. When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b. When Reactor Coolant System modification or repairs have been made which involve new strength welds on components, the new welds shall meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

c. The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first fifteen (15) effective full-power years of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressure during cooldown for the leak test temperature shall be in accordance with Figure 3.1.B-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak-tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak-tightness during normal operation.

The inservice leak temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G. This code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

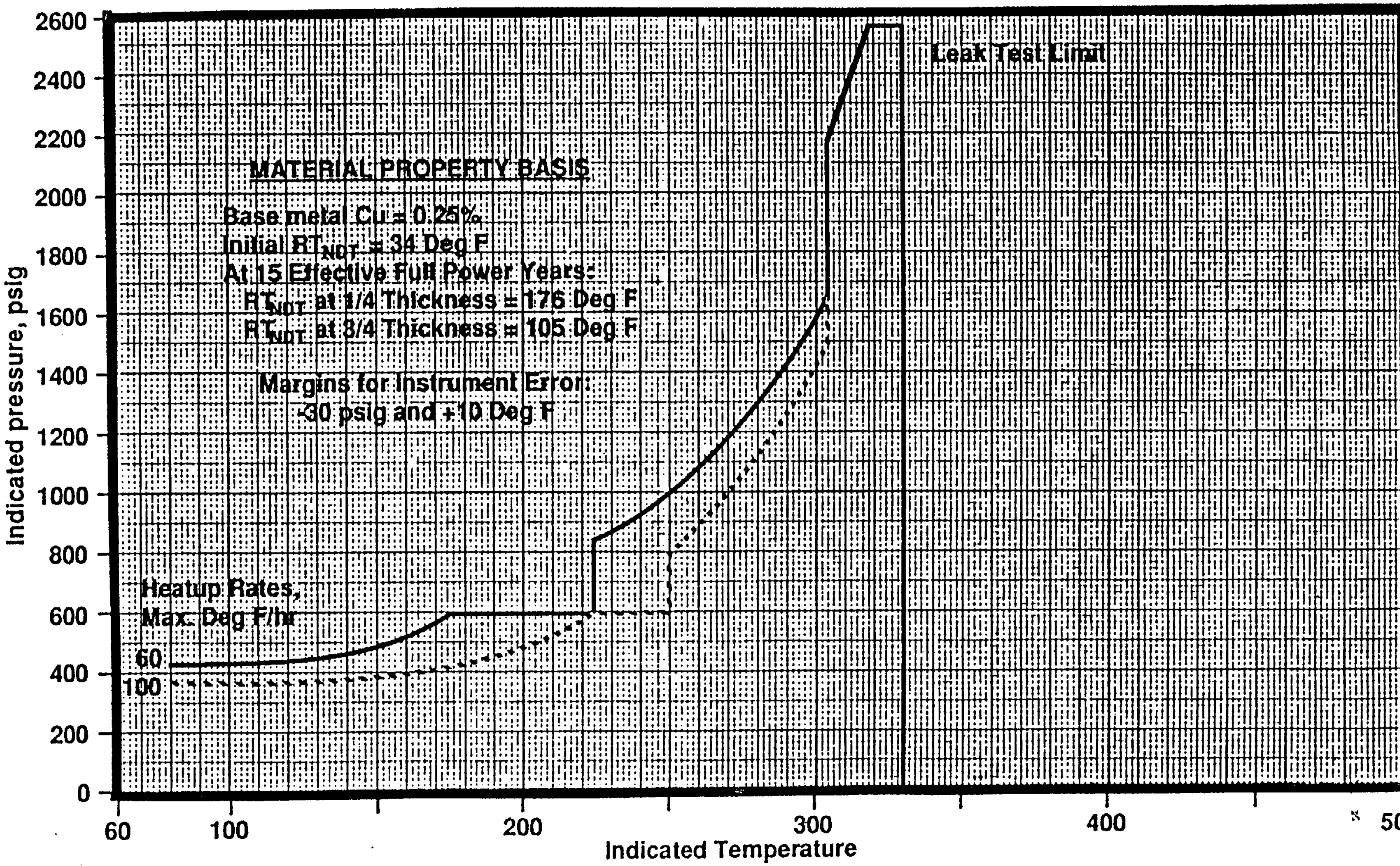
For the first fifteen (15) effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be $176^{o}F$. The minimum inservice leak test temperature requirements for periods up to fifteen (15) effective full-power years are shown on Figure 4.3-1.

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1.B-2 must not be exceeded. Figures 4.3-1 and 3.1.B-2 are recalculated periodically, using methods discussed in WCAP-7924A and results of surveillance specimen testing, as covered in WCAP-7323.

Reference

UFSAR Section 4

Figure 4.3-1
Indian Point Unit No. 2 Vessel leak Test Limitations
Applicable for Periods up to 15 Effective Full Power Years



4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage.

Objective

To verify that potential leakage from the containment is maintained within acceptable values.

Specifications

A. INTEGRATED LEAKAGE RATE

1. Test

a. A full-pressure integrated leakage rate test shall be performed at intervals specified in Specification 4.4.A.3 at the peak accident pressure (P_a) of 47 psig minimum.

b. A test duration of 24 hours, or an NRC approved reduced duration methodology, as described in BN-TOP-1, Revision 1, shall be used. The test shall be extended over a sufficient period of time to verify, by superimposing a known leak rate on the containment, the validity and accuracy of the leakage rate results.

c. A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to performing an integrated leak test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, integrated leakage rate tests shall not be performed until corrective action is taken. Such structural deterioration and corrective actions taken shall be reported as part of the test report.

d. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

2. Acceptance Criteria

The measured leakage rate shall be less than 0.75 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 47 psig and 271^{o}F, which are the peak accident pressure and temperature conditions.

3. Frequency

A set of three leakage rate tests shall be performed (during plant shutdown) at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shut down for the 10-year plant in-service inspection.

B. SENSITIVE LEAKAGE RATE

1. Test

A sensitive leakage rate test shall be conducted with the containment penetrations, weld channels, and certain double-gasketed seals and isolation valve interspaces at a minimum pressure of 47 psig and with the containment building at atmospheric pressure.

2. Acceptance Criteria

The test shall be considered satisfactory if the leak rate for the containment penetrations, weld channel and other pressurized zones is equal to or less than 0.2% of the containment free volume per day.

3. Frequency

A sensitive leakage rate test shall be performed at a frequency of at least every other refueling but in no case at intervals of greater than 3 years.

C. AIR LOCK TESTS

1. The containment air locks shall be tested at a minimum pressure of 47 psig and at a frequency of every 6 months. The acceptance criteria is included in Specification 4.4.D.2.a.

2. Whenever containment integrity is required, verification shall be made of proper repressurization to at least 47 psig of the double-gasket air lock door seal upon closing an air lock door.

D. CONTAINMENT ISOLATION VALVES

1. Tests and Frequency

a. Isolation valves in Table 4.4-1 shall be tested for operability at every refueling but in no case at intervals of greater than 2 years.

b. Isolation valves in Table 4.4-1 which are pressurized by the Weld Channel and Penetration Pressurization System shall be leakage tested as part of the Weld Channel and Penetration Pressurization System Test at every refueling but in no case at intervals of greater than 2 years.

c. Isolation valves in Table 4.4-1 which are pressurized by the Isolation Valve Seal Water System shall be tested at every refueling but in no case at intervals greater than 2 years as part of an overall Isolation Valve Seal Water System Test.

d. Isolation valves in Table 4.4-1 which are not pressurized will be tested at every refueling but in no case at intervals of greater than 2 years.

e. Isolation valves in Table 4.4-1 shall be tested with the medium and at the pressure specified therein.

2. Acceptance Criteria

a. The combined leakage rate for the following shall be less than 0.6 L_a: isolation valves listed in Table 4.4-1 subject to gas or nitrogen pressurization testing, air lock testing as specified in Specification 4.4.C.1, portions of the sensitive leakage rate test described in Specification 4.4.B.1 which pertain to containment penetrations and double-gasketed seals.

b. The leakage rate into containment for the isolation valves sealed with the service water system shall not exceed 0.36 gpm per fan cooler.

c. The leakage rate for the Isolation Valve Seal Water System shall not exceed 14,700 cc/hr.

3. Containment isolation valves may be added to plant systems without prior license amendment to Table 4.4-1 provided that a revision to this table is included in a subsequent license amendment application.

E. CONTAINMENT MODIFICATIONS

Any major modification or replacement of components of the containment performed after the initial pre-operational leakage rate test shall be followed by either an integrated leakage rate test or a local leak detection test and shall meet the appropriate acceptance criteria of Specifications 4.4.A.2, 4.4.B.2, or 4.4.D.2. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

F. REPORT OF TEST RESULTS

Each integrated leakage rate test shall be the subject of a summary technical report to be submitted to the Nuclear Regulatory Commission pursuant to Specification 6.9.2.a and in accordance with the requirements of Appendix J to 10 CFR 50, effective issue date March 16, 1973. Each report shall include leakage test results and a summary analyses of sensitive leak rate, air lock, and containment isolation valve tests performed since the previous integrated leakage rate test.

G. VISUAL INSPECTION

A detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed at each refueling shutdown and prior to any integrated leak test to uncover any evidence of deterioration which may affect either the containment structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

H. RESIDUAL HEAT REMOVAL SYSTEM

1. Test

a. (1) The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified below.

(2) The piping between the residual heat removal pumps suctions and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig at the interval specified below.

b. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

2. Acceptance Criterion

The maximum allowable leakage from the Residual Heat Removal System components located outside of the containment shall not exceed two gallons per hour.

3. Corrective Action

Repairs or isolation shall be made as required to maintain leakage within the acceptance criterion.

4. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at every refueling.

Basis

The containment is designed for a calculated peak accident pressure of 47 psig[1]. While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. With these initial conditions, the peak accident pressure and temperature of the steam-air mixture will not exceed the containment design pressure and temperature of 47 psig and 271°F.

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this preoperational leakage rate test was established as 0.10 weight percent (L_a) per 24 hours at 47 psig and 271°F, which are the peak accident pressure and temperature conditions. This leakage rate is consistent with the construction of the containment[(2)], which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing both the penetrations and the channels over all containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 weight percent per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10 CFR 100 values in the event of the design basis accident[(3)].

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the containment isolation valves are to be closed in the normal manner and without preliminary exercising or adjustments.

The minimum duration of 24 hours for the integrated leakage rate test is established to attain the desired level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation. If an ILRT of a duration less than 24 hours is attempted, the criteria of the Bechtel Topical Report, BN-TOP-1, Revision 1, will be met.

The frequency of the periodic integrated leakage rate test is keyed to the schedule for major shutdowns for inservice inspection and refueling. The specified frequency of periodic integrated leakage rate testing is based on the following major considerations.

The first consideration is the low probability of leaks in the liner because of:

(a) the tests of the leak-tight integrity of the welds during erection,

(b) conformance of the complete containment to a low leakage rate limit at 47 psig or higher during pre-operational testing, and

(c) absence of any significant stresses in the liner during reactor operation.

Secondly, the Weld Channel and Penetration Pressurization System is in service continuously to monitor leakage from potential leak paths such as the containment personnel lock seals and weld channels, containment penetrations, containment liner weld channels, double-gasketed seals and spaces between certain containment isolation valves and personnel door locks. A leak would be expected to build up slowly and would, therefore, be noted before design limits are exceeded. Remedial action can be taken before the limit is reached.

During normal plant operation, containment personnel lock door seals are continuously pressurized after each closure by the Weld Channel and Penetration Pressurization System. Whenever containment integrity is required, verification is made that seals repressurize properly upon closure of an air lock door.

A full pressure test of the air lock will be periodically performed at 6-month intervals to detect any unanticipated leakage.

The containment isolation valve leakage and sensitive leakage rate measurements obtained periodically, periodic inspection of accessible portions of the containment wall to detect possible damage to the liner plates, combined with the leakage monitoring afforded by the Weld Channel and Penetration Pressurization System[4] and IVSWS[5], provide assurance that the containment leakage is within design limits.

The testing of containment isolation valves in Table 4.4-1, either individually or in groups, utilizes the WC & PPS(4) or IVSWS(5) where appropriate and is in accordance with the requirements of Type C tests in Appendix J (issue effective date March 16, 1973) to 10 CFR 50. The specified test pressures are $\geq$ the peak calculated accident pressure. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days. The leakage limit for the Isolation Valve Seal Water System is consistent with the design capacity of the Isolation Valve Seal Water supply tank.

The acceptance criterion of 0.6 L_a for the combined leakage of isolation valves subject to gas or nitrogen pressurization, the air lock, containment penetrations and double-gasketed seals is in accordance with Appendix J (issue effective date March 16, 1973) to 10 CFR 50.

The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. A recirculation system leakage of 2 gal./hr. will limit offsite exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10 CFR 50 and ANSI N45.4-1972 "Leakage Rate Testing of Containment Structures for Nuclear Reactors" (March 16, 1972) for guidance.

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post-accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan-cooler.

References

(1) UFSAR - Section 5

(2) UFSAR - Section 5.1.6

(3) UFSAR - Section 14.3.6

(4) UFSAR - Section 6.6

(5) UFSAR - Section 6.5

Table 4.4-1

Containment Isolation Valves

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
549	PRT to Gas Analyzer	Water(4)	52
548	" " " "	Water(4)	52
518	PRT N_2 Supply	Gas	47
3418	" " "	Gas	47
3419	" " "	Gas	47
4136	" " "	Gas	47
552	PRT Makeup Water	Water(4)	52
519	" " "	Water(4)	52
741A	RHR return to RCS	Water(5)	52(3)
744	" " " "	Nitrogen(4)	47(3)
888A	RHR to S.I. Pumps	Nitrogen(4)	47
888B	" " " "	Nitrogen(4)	47
958	RHR to Sample System	Nitrogen(4)	47
959	" " " "	Nitrogen(4)	47
990D	" " " "	Nitrogen(4)	47
1870	RHR from RCS	Nitrogen(4)	47
743	" " "	Nitrogen(4)	47
732	" " "	Nitrogen(4)	47(3)
885A	Cont. Sump Recirc. Line	Water(5)	52
885B	" " " "	Water(5)	52
201	Letdown Line (CVCS)	Water(4)	52
202	" " "	Water(4)	52
205	Charging Line (CVCS)	Water(4)	52

Table 4.4-1

Containment Isolation Valves

Valve No.	System[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
226	Charging Line (CVCS)	Water[4]	52
227	" " "	Water[4]	52
250A	RCP Seal Water (CVCS)	Water[4]	52
4925	" " " "	Water[4]	52
250B	" " " "	Water[4]	52
4926	" " " "	Water[4]	52
250C	" " " "	Water[4]	52
4927	" " " "	Water[4]	52
250D	" " " "	Water[4]	52
4928	" " " "	Water[4]	52
222	" " " "	Water[4]	52
956E	RCS to Sample System	Water[4]	52
956F	" " " "	Water[4]	52
869A	Cont. Spray System	Water[4]	52
867A	" " "	Gas	47
878A	" " "	Gas	47
869B	" " "	Water[4]	52
867B	" " "	Gas	47
851A	Safety Inj. System	Water[4]	52
850A	" " "	Water[4]	52
851B	" " "	Water[4]	52
850B	" " "	Water[4]	52
859A	S.I. Test Line	Water[4]	52

Table 4.4-1

Containment Isolation Valves

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
859C	S.I. Test Line	Water(4)	52
4312	Acc. & OPS N_2 Supply	Gas	47
863	" " " " "	Gas	47
956G	Acc. to Sample System	Water(4)	52
956H	" " " "	Water(4)	52
1786	RCDT to Vent Header	Water(4)	52
1787	" " " "	Water(4)	52
3416	RCDT N_2 Supply	Gas	47
3417	" " "	Gas	47
5459	" " "	Gas	47
1616	" " "	Gas	47
1788	RCDT to Gas Analyzer	Water(4)	52
1789	" " " "	Water(4)	52
1702	RCDT to WHT (WDS)	Water(4)	52
1705	" " " "	Water(4)	52
797	RCP Comp. Cooling (CCS)	Water(4)	52
784	" " " "	Water(4)	52
FCV-625	" " " "	Water(4)	52
791	Excess Letdown Cool. (CCS)	Water(4)	52
798	" " " "	Water(4)	52
796	" " " "	Water(4)	52
793	" " " "	Water(4)	52
1728	Cont. Sump to WHT (WDS)	Water(4)	52

Table 4.4-1

Containment Isolation Valves

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
1723	Cont. Sump to WHT (WDS)	Water(4)	52
1234	Cont. Air Sample	Gas(7)	47
1235	" " "	Gas(7)	47
1236	" " "	Gas(7)	47
1237	" " "	Gas(7)	47
PCV-1229	Air Ejector to Cont.	Gas(7)	47
PCV-1230	" " " "	Gas(7)	47
PCV-1214	S.G. Blowdown/Sample	Water(4)	52
PCV-1214A	" " "	Water(4)	52
PCV-1215	" " "	Water(4)	52
PCV-1215A	" " "	Water(4)	52
PCV-1216	" " "	Water(4)	52
PCV-1216A	" " "	Water(4)	52
PCV-1217	" " "	Water(4)	52
PCV-1217A	" " "	Water(4)	52
SWN-41-5-A	Cont. Fan Cooler-Ser. Wtr.	Water(6)	52
SWN-41-5-B	" " " " "	Water(6)	52
SWN-43-5	" " " " "	Water(6)	52
SWN-42-5	" " " " "	Water(6)	52
SWN-41-1-A	" " " " "	Water(6)	52
SWN-41-1-B	" " " " "	Water(6)	52
SWN-43-1	" " " " "	Water(6)	52
SWN-42-1	" " " " "	Water(6)	52

Table 4.4-1

Containment Isolation Valves

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
SWN-41-2-A	Cont. Fan Cooler-Ser. Wtr.	Water(6)	52
SWN-41-2-B	" " " " "	Water(6)	52
SWN-43-2	" " " " "	Water(6)	52
SWN-42-2	" " " " "	Water(6)	52
SWN-41-3-A	" " " " "	Water(6)	52
SWN-41-3-B	" " " " "	Water(6)	52
SWN-43-3	" " " " "	Water(6)	52
SWN-42-3	" " " " "	Water(6)	52
SWN-41-4-A	" " " " "	Water(6)	52
SWN-41-4-B	" " " " "	Water(6)	52
SWN-43-4	" " " " "	Water(6)	52
SWN-42-4	" " " " "	Water(6)	52
SWN-44-5-A	" " " " "	Water(6)	52
SWN-44-5-B	" " " " "	Water(6)	52
SWN-51-5	" " " " "	Water(6)	52
SWN-44-1-A	" " " " "	Water(6)	52
SWN-44-1-B	" " " " "	Water(6)	52
SWN-51-1	" " " " "	Water(6)	52
SWN-44-2-A	" " " " "	Water(6)	52
SWN-44-2-B	" " " " "	Water(6)	52
SWN-51-2	" " " " "	Water(6)	52
SWN-44-3-A	" " " " "	Water(6)	52
SWN-44-3-B	" " " " "	Water(6)	52

Table 4.4-1

Containment Isolation Valves

Valve No.	System (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
SWN-51-3	Cont. Fan Cooler-Ser. Wtr.	Water (6)	52
SWN-44-4-A	" " " " "	Water (6)	52
SWN-44-4-B	" " " " "	Water (6)	52
SWN-51-4	" " " " "	Water (6)	52
SWN-71-5-A	" " " " "	Water (6)	52
SWN-71-5-B	" " " " "	Water (6)	52
SWN-71-1-A	" " " " "	Water (6)	52
SWN-71-1-B	" " " " "	Water (6)	52
SWN-71-2-A	" " " " "	Water (6)	52
SWN-71-2-B	" " " " "	Water (6)	52
SWN-71-3-A	" " " " "	Water (6)	52
SWN-71-3-B	" " " " "	Water (6)	52
SWN-71-4-A	" " " " "	Water (6)	52
SWN-71-4-B	" " " " "	Water (6)	52
SA-24	Service Air to Cont.	Water (4)	52
SA-24-1	" " " "	Water (4)	52
580A	Dead Weight Tester	Gas	47
580B	" " "	Gas	47
UH-43	Auxiliary Steam System	Water (4)	52
UH-44	" " "	Water (4)	52
MW-17	City Wtr. to Cont.	Water (4)	52
MW-17-1	" " " "	Water (4)	52
1170	Cont. Purge System	Gas (7)	47

Table 4.4-1

Containment Isolation Valves

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
1171	Cont. Purge System	Gas(7)	47
1172	" " "	Gas(7)	47
1173	" " "	Gas(7)	47
1190	Cont. Pressure Relief	Gas(7)	47
1191	" " "	Gas(7)	47
1192	" " "	Gas(7)	47
990A	Recirc. Pump to Samp. Sys.	Nitrogen(4)	47
990B	" " " " "	Nitrogen(4)	47
956A	Pressurizer to Samp. Sys.	Water(4)	52
956B	" " " "	Water(4)	52
956C	" " " "	Water(4)	52
956D	" " " "	Water(4)	52
1814A	Cont. Pressure Instr.	Gas	47
1814B	" " "	Gas	47
1814C	" " "	Gas	47
5018	Post Acc. Cont. Sampling	Gas	47
5019	" " " "	Gas	47
5020	" " " "	Gas	47
5021	" " " "	Gas	47
5022	" " " "	Gas	47
5023	" " " "	Gas	47
5024	" " " "	Gas	47
5025	" " " "	Gas	47

Table 4.4-1

Containment Isolation Valves

Valve No.	System(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
95B	Equipment Airlock	Gas	47
95C	" "	Gas(7)	47
95D	" "	Gas(7)	47
4399	Sample Return to Cont. Sump.	Water(4)	52
5132	" " "	Water(4)	52

Notes:

1. System in which valve is located.

2. Gas test fluid indicates either nitrogen or air as test medium.

3. Testable only when at cold shutdown.

4. Isolation Valve Seal Water System.

5. Sealed by Residual Heat Removal System fluid.

6. Sealed by Service Water System. Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).

7. Sealed by Weld Channel and Penetration Pressurization System.

4.5 ENGINEERED SAFETY FEATURES

Applicability

Applies to testing of the Safety Injection System, the Containment Spray System, the Hydrogen Recombiner System, and the Air Filtration System.

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required.

Specifications

A. SYSTEM TESTS

1. Safety Injection System

   a. System tests shall be performed at each reactor refueling interval. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350^{o}F, a test safety injection signal will be applied to initiate operation of the system. The safety injection pumps are made inoperable for this test.

   b. The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing; that is, the appropriate pump breakers shall have opened and closed, and the appropriate valves shall have completed their travel.

   c. Conduct a flow test of the high head safety injection system after any modification is made to either its piping and/or valve arrangement.

d. Verify that the mechanical stops on Valves 856 A, C, D and E are set at the position measured and recorded during the most recent ECCS operational flow test or flow tests performed in accordance with (c) above. This surveillance procedure shall be performed following any maintenance on these valves or their associated motor operators and at a convenient outage if the position of the mechanical stops has not been verified in the preceding three months.

B. CONTAINMENT SPRAY SYSTEM

1. System tests shall be performed at each reactor refueling interval. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

2. The spray nozzles shall be tested for proper functioning at least every five years.

3. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

C. HYDROGEN RECOMBINER SYSTEM

1. A complete recombiner system test shall be performed at each normal reactor refueling on each unit. The test shall include verification of ignition and attainment of normal operating temperature.

2. A complete control system test shall be performed at intervals not greater than six months on each unit. The test shall consist of a complete dry run startup using artificially generated signals to simulate light off.

3. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

4. Each recombiner air-supply blower shall be started at least at two-month intervals. Acceptable levels of performance shall be that the blowers start, deliver flow, and operate for at least 15 minutes.

D. CONTAINMENT AIR FILTRATION SYSTEM

Each air filtration unit specified in Specification 3.3.B shall be demonstrated to be operable:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the unit operates for at least 15 minutes.

2. At least once per 18 months, or (1) after any structural maintenance on the HEPA filters or charcoal adsorber housings, or (2) at any time painting, fire or chemical releases could alter filter integrity by:

   a. verifying a system flow rate at ambient conditions of 65,600 cfm ±10% during filtration unit operation when tested in accordance with ANSI N510-1975. Verify that the flow rate through the charcoal adsorbers is $\geq$ 8,000 cfm.

   b. verifying that the HEPA filters and/or charcoal adsorbers satisfy the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, at ambient conditions and at a flow rate of 65,600 cfm ±10% for the HEPA filters.

   c. 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 (except for Position C.6.a(1)) of Regulatory Guide 1.52, Revision 2, March 1978.

3. 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 (except for Position C.6.a(1)) of Regulatory Guide 1.52, Revision 2, March 1978.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the moisture separator and HEPA filters is less than 6 inches Water Gauge while operating the filtration unit at ambient conditions and at a flow rate of 65,600 cfm ±10%.

   b. Verifying that the unit starts automatically on a Safety Injection Test Signal.

5. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the unit at ambient conditions and at a flow rate of 65,600 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank, verify that the flow rate through the charcoal adsorbers is ≥8,000 cfm when the system is operating at ambient conditions and a flow rate of 65,600 cfm ±10% when tested in accordance with ANSI N510-1975.

E. CONTROL ROOM AIR FILTRATION SYSTEM

The control room air filtration system specified in Specification 3.3.H shall be demonstrated to be operable:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

2. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) at any time painting, fire or chemical releases could alter filter integrity by:

   a. verifying a system flow rate, at ambient conditions, of 1840 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

   b. verifying that, with the system operating at ambient conditions and at a flow rate of 1840 CFM ±10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

   c. verifying that the system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at ambient conditions and at a flow rate of 1840 cfm ±10%.

   d. 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.

3. 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 1973, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. At least once per 18 months by:

   a. 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 ambient conditions and at a flow rate of 1840 cfm ±10%.

   b. verifying that, on a Safety Injection Test Signal or a high radiation signal in the control room, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

   c. verifying that the system maintains the control room at a neutral or positive pressure relative to the outside atmosphere during system operation.

5. After each complete or partial replacement of an HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 1840 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 1840 cfm ±10%.

7. Each toxic gas detection system shall be demonstrated operable by performance of a channel check at least once per day, a channel test at least once per 31 days and a channel calibration at least once per 18 months.

F. FUEL STORAGE BUILDING AIR FILTRATION SYSTEM

The fuel storage building air filtration system specified in Specification 3.8 shall be demonstrated operable:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

2. At each refueling shutdown, prior to refueling operations, or (1) after any structural maintainance on the HEPA filter or charcoal adsorber housings, or (2) at any time painting, fire or chemical releases could alter filter integrity by:

   a. verifying a system flow rate at ambient conditions of 20,000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

   b. verifying that the system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at ambient conditions and at a flow rate of 20,000 cfm ±10%.

   c. 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.

3. Prior to handling spent fuel which has decayed for less than 35 days, verify 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. Such an analysis is good for 720 hours of charcoal adsorber operation. After 720 hours of operation, if spent fuel with a decay time of less than 35 days is still being handled, a new sample is required along with a new analysis.

4. At each refueling shutdown, prior to refueling operations by:

   a. 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 ambient conditions and at a flow rate of 20,000 cfm ±10%.

   b. verifying that the system maintains the spent fuel storage pool area at a pressure less than that of the outside atmosphere during system operation.

5. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 20,000 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 20,000 cfm ±10%.

G. <u>POST-ACCIDENT CONTAINMENT VENTING SYSTEM</u>

The post-accident containment venting system shall be demonstrated operable:

1. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) at any time painting, fire or chemical releases could alter filter integrity by:

   a. verifying no flow blockage by passing flow through the filter system.

   b. verifying that the system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at ambient conditions and at a flow rate of 200 cfm ±10%.

   c. 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.

2. 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.

3. At least once per 18 months by:

   a. 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 ambient conditions and at a flow rate of 200 cfm ±10%.

   b. verifying that the system valves can be manually opened.

4. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 200 cfm ±10%.

5. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 200 cfm ±10%.

Basis

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during plant refueling shutdowns, with more frequent component tests, which can be performed during reactor operation.

The refueling systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry[1].

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel input is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupt the logic matrix output to the master relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the logic matrices to the master relay is complete is accomplished by use of an ohm-meter to check continuity.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and, by these means, are continuously monitored for satisfactory performance.

For the four flow distribution valves (856 A, C, D and E), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

The hydrogen recombiner system is an engineered safety feature which would be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be started until approximately 13 days have elapsed following the accident. At this time the hydrogen concentration in the containment will have reached 2% by volume, which is the design concentration for starting the recombiner system. Actual starting of the system will be based upon containment atmosphere sample analysis. The complete functional tests of each unit at refueling shutdown will demonstrate the proper operation of the recombiner system. More frequent tests of the recombiner control system and air-supply blowers will assure operability of the system. The biannual testing of the containment atmosphere sampling system will demonstrate the availability of this system.

The charcoal portion of the in-containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the required periodic visual inspections will verify that this is the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore, at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiency for removal of methyl iodide is obtained(2). Such laboratory charcoal sample testing together with the specified in-place testing of the HEPA filters will provide further assurance that the criteria of 10 CFR 100 continue to be met.

The control room air filtration system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air filtration system is designed to automatically start upon control room isolation. High-efficiency particulate absolute (HEPA) filters are installed upstream of the charcoal adsorbers to prevent clogging of these adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine by control room personnel. The required in-place testing and the laboratory charcoal sample testing of the HEPA filters and charcoal adsorbers will provide assurance that Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50 continues to be met.

The fuel storage building air filtration system is designed to filter the discharge of the fuel storage building atmosphere to the plant vent. This air filtration system is designed to start automatically upon a high radiation signal. Upon initiation, isolation dampers in the ventilation system are designed to close to redirect air flow through the air treatment system. HEPA filters and charcoal adsorbers are installed to reduce potential releases of radioactive material to the atmosphere. Nevertheless, as required by Specification 3.8.B.6, the fuel storage building air filtration system must be operating whenever spent fuel is being moved unless the spent fuel has had a continuous 35-day decay period. The required in-place testing and the laboratory charcoal sample testing of the HEPA filters and charcoal adsorbers will provide added assurance that the criteria of 10 CFR 100 continue to be met.

The post-accident containment venting system may be used in lieu of hydrogen recombiners for removal of combustible hydrogen from the containment building atmosphere following a design basis accident. As was the case for hydrogen recombiner use, this system is not expected to be needed until approximately 13 days have elapsed following the accident. Use of the system will be based upon containment atmosphere sample analysis and availability of the hydrogen recombiners. When in use, HEPA filters and charcoal adsorbers will filter the containment atmosphere discharge prior to release to the plant vent. The required in-place testing and laboratory charcoal sample testing will verify operability of this venting system and provide further assurance that releases to the environment will be minimized.

As indicated for all four of the previously mentioned engineered safety feature (ESF) air filtration systems, high-efficiency particulate absolute (HEPA) filters are installed upstream of the charcoal adsorbers to prevent clogging of these adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The laboratory charcoal sample testing periodically verifies that the charcoal meets the iodine removal efficiency requirements of Regulatory Guide 1.52, Revision 2. Should the charcoal of any of these filtration systems fail to satisfy the specified test acceptance criteria, the charcoal will be replaced with new charcoal which satisfies the requirements for new charcoal outlined in Regulatory Guide 1.52, Revision 2.

References

(1) UFSAR Section 6.2
(2) UFSAR Section 6.4

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability

Applies to periodic testing and surveillance requirements of the emergency power systems.

Objective

To verify that emergency power systems will respond promptly and properly when required.

Specifications

The following tests and surveillances shall be performed as stated:

A. DIESEL GENERATORS

1. Each month, each diesel generator shall be manually started and synchronized to its bus or buses and shall be allowed to assume the normal bus load.

2. At each refueling outage, each diesel generator shall be manually started, synchronized and loaded up to its nameplate rating.

3. At each refueling outage, to assure that each diesel generator will automatically start and assume the required load within 60 seconds after the initial start signal, the following shall be accomplished: by simulating a loss of all normal AC station service power supplies and simultaneously simulating a Safety Injection signal, observations shall verify automatic start of each diesel generator, required bus load shedding and restoration to operation of particular vital equipment. To prevent Safety Injection flow to the core, certain safeguards valves will be closed and made inoperable.

4. Each diesel generator shall be given a thorough inspection at least annually following the manufacturer's recommendations for this class of stand-by service.

The above tests will be considered satisfactory if the required minimum safeguards equipment operated as designed.

B. DIESEL FUEL TANKS

A minimum oil storage of 41,000 gallons will be maintained at the station at all times.

C. STATION BATTERIES (NOS. 21, 22, 23 & 24)

1. Every month, the voltage of each cell, the specific gravity and temperature of a pilot cell in each battery and each battery voltage shall be measured and recorded.

2. Every 3 months, each battery shall be subjected to a 24-hour equalizing charge, and the specific gravity of each cell, the temperature reading of every fifth cell, the height of electrolyte, and the amount of water added shall be measured and recorded.

3. Each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

4. At each refueling interval, each battery shall be subjected to a load test and a visual inspection of the plates.

D. GAS TURBINE GENERATORS

1. At monthly intervals, at least one gas turbine generator shall be started and synchronized to the power distribution system for a minimum of thirty (30) minutes with a minimum electrical output of 750 kW.

E. GAS TURBINE FUEL SUPPLY

1. At weekly intervals, the minimum gas turbine fuel volume shall be verified to be available and shall be documented in the plant log.

Basis

The tests specified in Specifications 4.6.A, 4.6.B and 4.6.C are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency diesel generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of all normal 480v ac station service power.

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply is continuously monitored. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

Each diesel generator has a continuous rating of 1750 kW with a 2000 hr rating of 2000 kW. Two diesels operating at their continuous rating can power the minimum safeguards loads. A minimum oil storage of 41,000 gallons will provide for operation of the minimum required engineered safeguards on emergency diesel power for a period of 168 hours.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails. The periodic equalizing charge will ensure that the ampere-hour capability of the batteries is maintained.

The refueling interval load test for each battery, together with the visual inspection of the plates, will assure the continued integrity of the batteries. The batteries are of the type that can be visually inspected, and this method of assuring the continued integrity of the battery is proven standard power plant practice.

The tests specified in Specifications 4.6.D and 4.6.E are designed to assure that at least one gas turbine generator will be available to provide power for operation of equipment if required. Since the Indian Point Unit No. 2 alternate safe shutdown power supply system demands a maximum electrical load of approximately 750 kW, the required minimum test load will demonstrate adequate capability. In addition, the minimum gas turbine fuel oil storage volume of 54,200 gallons will conservatively assure at least three (3) days of operation of a gas turbine generator.

The specified test frequencies for the gas turbine generator(s) and associated fuel supply will be adequate to identify and correct any mechanical or electrical deficiency before it can result in a component malfunction or failure.

Reference

UFSAR Section 8.2

4.7 MAIN STEAM STOP VALVES

Applicability

Applies to periodic testing of the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal.

Specification

The main steam stop valves shall be tested at refueling intervals with the reactor at cold shutdown. Closure time of five seconds or less shall be verified.

Basis

The main steam stop valves serve to limit an excessive Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam break incident[1]. Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis[2].

References

(1) UFSAR - Section 10.4
(2) UFSAR - Section 14.2.5

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.

Specifications

A. The following surveillance tests shall be performed at refueling intervals:

1. Verification of proper operation of auxiliary feedwater system components and initiating logic upon receipt of test signals for each mode of automatic initiation.

2. Verification of the capability of each auxiliary feedwater pump to deliver full flow to the steam generators.

B. The above tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Basis

The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements. Testing of the auxiliary feedwater system will verify its operability. These specifications establish those surveillance tests to be performed at refueling intervals to verify operability of both the automatic initiation circuitry and the individual components necessary for proper functioning of the auxiliary feedwater system. This testing will verify proper component actuation upon receipt of all required automatic initiation signals and will verify that adequate system flow rates and pressures are obtained with proper valve positioning and pump full-flow operation. Both control room instrumentation and visual observation of the equipment will be used to verify proper component operation.

The periodic "operational readiness" testing required by the ASME Code Section XI for pumps and valves in the auxiliary feedwater system is conducted as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program and is therefore not included in these specifications.

References

UFSAR - Sections 10.4, 14.1.9, 14.1.12 and 14.4.6

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burn-up, actual boron concentration of the coolant shall be periodically compared with the predicted value.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization shall be completed early in core life. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully-withdrawn position is always maintained.

4.10 RADIOACTIVE EFFLUENTS

Applicability

The specification applies to the surveillance requirements associated with facility radioactive effluents.

Objective

To ensure that the releases of radioactive materials to the environs are maintained as low as reasonably achievable (ALARA) and within allowable regulatory limits.

Specifications

A. RADIOACTIVE LIQUID EFFLUENTS

Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.10-1.

The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to ensure that the concentrations at the point of release are maintained within the limits of Specification 3.9.A.1.

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable by performance of the channel check, source check, channel calibration and channel function test operations at the frequencies shown in Table 4.10-2.

Cumulative dose contribution from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

Doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

The quantity of radioactive material contained in each of the tanks listed in Specification 3.9.A.5 shall be determined to be within the limit of Specification 3.9.A.5.a by analyzing a representative sample of the tank's contents at least once per month when radioactive materials are being added to the tank.

B. RADIOACTIVE GASEOUS EFFLUENTS

1. The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.9.B.1.a(i) in accordance with the methodology and parameters in the ODCM.

2. The dose rate due to iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.9.B.1.a(ii) in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.10-3.

3. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated operable by performance of the channel check, source check, channel calibration and channel functional test operations at the frequencies shown in Table 4.10-4.

4. Cumulative dose contribution for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

5. Cumulative dose contribution for the current calendar quarter and current calendar year for iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

6. Doses due to gaseous releases from each reactor unit to areas at and beyond the site boundary shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

7. The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the limits of Specification 3.9.B.6 by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required operable by Table 3.9.2 of Specification 3.9.B.2.a.

8. The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limit of Specification 3.9.B.7 at least once per 24 hours when radioactive materials are being added to the tank.

C. URANIUM FUEL CYCLE DOSE COMMITMENT

1. Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.10.A.4, 4.10.B.4, and 4.10.B.5, and in accordance with the methodology and parameters in the ODCM.

2. Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.9.C.2.

D. SOLID RADIOACTIVE WASTE

1. The process control program shall be used to verify the solidification of test specimens representative of wet waste.

2. For each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period, the following information shall be recorded:

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate),

c. principal radionuclides (specify whether determined by measurement or estimate),

d. source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent or absorbent (e.g., cement, urea formaldehyde).

E. ROUTINE REPORTING REQUIREMENTS

1. The results of the surveillance performed pursuant to Specifications 4.10.A and 4.10.B above will be included in the Semiannual Radioactive Effluent Release Report to the extent required by Specification 6.9.1.6. In addition, this report shall include the information recorded pursuant to Specification 4.10.D above during the report period.

Basis

The sampling and monitoring requirements specified in Specification 4.10.A and 4.10.B provide assurance that radioactive materials released in liquid and gaseous wastes are properly controlled and monitored in conformance with the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50. These requirements provide the data for the licensee and the Commission to evaluate the performance of the plants relative to radioactive waste effluents released to the environment.

Specification 4.10.B excludes monitoring the turbine building ventilation exhaust since this release is expected to be a negligible release point. Many PWR reactors do not have turbine building enclosures. To be consistent in the requirement for all PWR reactors, the monitoring of gaseous releases from turbine building is not required.

Specification 4.10.D specifies the appropriate information to be recorded and maintained regarding offsite shipments of solid radioactive waste.

As referenced in Specification 4.10.E, reports on the quantities of liquid, gaseous and solid radioactive materials released from the site are furnished to the Commission in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6 and in conformance with USNRC Regulatory Guide 1.21, Revision 1. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

Table 4.10-1

Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a,g,c] (μCi/ml)
A. Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters Mo-99, Ce-144 I-131	$5x10^{-7}$ $5x10^{-6}$ $1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P Each Batch	M Composite[d]	H-3 Gross Alpha	$1x10^{-5}$ $1x10^{-7}$
	P Each Batch	Q Composite[d]	Sr-89, Sr-90 Fe-55	$5x10^{-8}$ $1x10^{-6}$
B. Continuous Releases[e]	Composite[d]	W Composite[d]	Principal Gamma Emitters[c] Mo-99, Ce-144 I-131	$5x10^{-7}$ $5x10^{-6}$ $1x10^{-6}$
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Composite[d]	M Composite[d]	H-3 Gross Alpha	$1x10^{-5}$ $1x10^{-7}$
	Composite[d]	Q Composite[d]	Sr-89, Sr-90 Fe-55	$5x10^{-8}$ $1x10^{-6}$

Table 4.10-1

Radioactive Liquid Waste Sampling and Analysis Program

Table Notation

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above systems background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal. The LLD shall be determined in accordance with the methodology and parameters in the ODCM. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling. (Steam Generators may be considered a batch release for reporting purposes during shutdown condition.)

c. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, and Ce-141. Other gamma peaks that are identified, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6.

d. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

f. When operational or other limitations preclude specific gamma radionuclide analysis in batch releases, the provisions of Regulatory Guide 1.21 (Revision 1), Appendix A, Section C.4 and Appendix A, Section B shall be followed.

For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased in inverse proportion to the magnitude of the gamma yield (i.e., $5 \times 10^{-7}/I$, where I is the photon abundance expressed as a decimal fraction).

Table 4.10-2

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. Liquid Radwaste Effluent Line	D*	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line	D*	M	R(3)	Q(1)
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line	D*	M	R(3)	Q(2)
b. Unit 1 Secondary Boiler Blowdown Effluent Line	D*	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENTS DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D(4)	N.A.	R	Q
4. TANK LEVEL INDICATING DEVICES***				
a. 13 Waste Distillate Storage Tank	D**	N.A.	R	Q
b. 14 Waste Distillate Storage Tank	D**	N.A.	R	Q
c. Primary Water Storage Tank	D**	N.A.	R	Q
d. Refueling Water Storage Tank	D**	N.A.	R	Q
e. 21 Monitor Tank	D**	N.A.	R	Q
f. 22 Monitor Tank	D**	N.A.	R	Q
g. 23 Monitor Tank	D**	N.A.	R	Q

* During releases via this pathway

** During liquid additions to the tank

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

Table 4.10-2

Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Table Notation

(1) The channel functional test shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if the following conditions exist:

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

(2) The channel functional 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 setpoint.

2. Instrument controls not set in operate mode.

(3) Radioactive calibration standards used for channel calibrations shall be analyzed with instrumentation which is calibrated with NBS traceable standards. (Standards from suppliers who participate in measurement assurance activities with NBS are acceptable.)

(4) Channel check shall consist of verifying indication of flow during periods of release. Channel check shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

Table 4.10-3

Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Releases Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment Purge	P Each Purge Grab Sample	P Each Purge	Principal Gamma Emitters[b]	1×10^{-4}
C. Condenser Air Ejector	M	M	Principal Gamma Emitters[b]	1×10^{-4}
D. Plant Vent	M[c] Grab Sample	M[c]	Principal Gamma Emitters[b]	1×10^{-4}
	Continuous[f]	W[e]	H-3	1×10^{-6}
	Continuous[f]	W[g] Charcoal Sample	I-131	1×10^{-12}
	Continuous[f]	W Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11} 1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6} d

Table 4.10-3

Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Releases Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (µCi/ml)
E. Stack Vent	M Grab Sample	M	Principal Gamma Emitters[b]	$1x10^{-4}$
	Continuous[f]	W[e]	H-3	$1x10^{-6}$
	Continuous[f]	W Charcoal Sample	I-131	$1x10^{-12}$
	Continuous[f]	W Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	$1x10^{-11}$
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$ d

Table 4.10-3

Radioactive Gaseous Waste Sampling and Analysis Program

Table Notation

a The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above systems background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal. The LLD shall be determined in accordance with the methodology and parameters in the ODCM.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

b The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6.

c If discharging via this pathway, sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 15 percent of rated power within one hour unless (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased by more than a factor of 3, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

d Radiation monitor sensitivity.

e Grab sample can be used as alternative to continuous sampling.

f The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.9.B.1.a, 3.9.B.3.a and 3.9.B.4.a.

Table 4.10-3

Radioactive Gaseous Waste Sampling and Analysis Program

Table Notation

g Continuous samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or thermal power change exceeding 15 percent of rated power in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

This requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased by more than a factor of 3, or (2) the noble gas monitor shows that effluent activity has not increased by more than a factor of 3.

Table 4.10-4

Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test	Modes In Which Surveillance Required
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Providing Alarm	D	M	$R^{(3)}$	$Q^{(2)}$	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	$Q^{(4)}$	M	**
b. Hydrogen or Oxygen Monitor	D	N.A.	$Q^{(5)}$	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity	D	M	$R^{(3)}$	$Q^{(2)}$	*
4. PLANT VENT					
a. Noble Gas Activity Monitor	D	M	$R^{(3)}$	$Q^{(2)}$	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
5. STACK VENT					
a. Noble Gas Activity Monitor	D	P	$R^{(3)}$	$Q^{(1)}$	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

Table 4.10-4

Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Table Notation

* Surveillance is required at all times except when monitor has been removed from service in accordance with Table 3.9-2.

** During waste gas holdup system operation (treatment for primary system off-gasses).

(1) The channel functional test shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the following conditions exist:

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

(2) The channel functional 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 setpoint.
2. Instrument controls not set in operate mode.

(3) Radioactive Calibration Standards used for channel calibrations shall be traceable to the National Bureau of Standards or an aliquot of calibration gas shall be analyzed with instrumentation which is calibrated with NBS traceable standards (standards from suppliers who participate in measurement assurance activities with NBS are acceptable).

(4) The channel calibration shall include the use of standard gas samples containing:

1. less than or equal to two volume percent hydrogen, and
2. greater than or equal to four volume percent hydrogen.

(5) The channel calibration shall include the use of standard gas samples containing:

1. less than or equal to two volume percent oxygen, and
2. greater than or equal to two volume percent oxygen.

4.11 RADIOLOGICAL ENVIRONMENTAL MONITORING

Applicability

Applies to routine testing of the radioactivity in the plant environs and is applicable to the entire Indian Point site.

Objective

The overall objectives of the radiological environmental monitoring program are:

(1) to establish a sampling schedule for the entire Indian Point site, which will recognize changes in radioactivity in the environs of the plants,

(2) to assure that the effluent releases are kept as low as reasonably achievable (ALARA) and within allowable limits in accordance with 10 CFR 20, and

(3) to verify projected and anticipated radioactivity concentrations in the environment and related exposures from releases of radioactive materials from Indian Point Units 1, 2 and 3.

Specifications

A. MONITORING PROGRAM

1. As a minimum, radiological environmental monitoring samples shall be collected pursuant to Table 4.11-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 4.11-1 and the detection capabilities required by Table 4.11-3.

2. With the radiological environmental monitoring program not being conducted as specified in Table 4.11-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.5, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

3. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 4.11-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a member of the public is less than the calendar year limits of Specifications 3.9.A.3.a, 3.9.B.3.a, and 3.9.B.4.a. When more than one of the radionuclides in Table 4.11-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 4.11-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to a member of the public is equal to or greater than the calendar year limits of Specifications 3.9.A.3.a, 3.9.B.3.a and 3.9.B.4.a. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

4. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 4.11-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the

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* The methodology and parameters used to estimate the potential annual dose to a member of the public shall be indicated in this report.

monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.6, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

B. LAND USE CENSUS

1. A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location, in each of the 16 meteorological sectors, of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations, in each of the 16 meteorological sectors, of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.)

2. The land use census shall be conducted during the growing seasons at least once per calendar year using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

3. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.10.B.4, in lieu of a Licensee Event Report,

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* Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.11-1.4c shall be followed, including analysis of control samples.

identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.6.

4. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) a factor of 2 greater than at a location from which samples are currently being obtained in accordance with Specification 4.11.A, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.6, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

C. INTERLABORATORY COMPARISON PROGRAM

1. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

2. With analyses not being performed as required in Specification 4.11.C.1 above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

3. The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

D. ROUTINE REPORTING REQUIREMENTS

1. A summary of the results of the monitoring program in Specification 4.11.A, the results the land use census in Specification in 4.11.B, and the results of analysis performed as part of an Interlaboratory Comparison Program in Specification in 4.11.C shall all be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

Basis

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. Program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.11-3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Specification 4.11.B is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

Table 4.11-1

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	40 routine monitoring stations (DR1-DR40) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the site boundary (DR1-DR16); an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR17-DR32); the balance of the stations (DR33-DR40) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly

Table 4.11-1

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 site boundary locations, in different sectors, of the highest calculated annual average groundlevel D/Q. 1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-level D/Q. 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]	Continuous sampler operation with sample collection weekly, or more frequently by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change,[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
3. WATERBORNE			
a. Surface[f]	1 sample upstream (Wa1) 1 sample downstream (Wa2)	Composite sample 1-month period[g]	Gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.

Table 4.11-1

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
b. Drinking	1 sample (Wb1) of the nearest surface drinking water supply	Grab Sample - monthly	Gross beta and gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
c. Sediment from Shoreline	2 samples (Wc1-Wc2) 1 sample (Wc1) from downstream area with existing or potential recreational value. 1 control sample (Wc2) from an upstream area.	2 annually at least 90 days apart	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations (Ia1-Ia3) within 5 km distance having the highest dose potential. If there are none, then 1 sample from milking animals in each of 3 areas (Ia1-Ia3) between 5 to 8 km distance, if available, where doses are calculated to be greater than 1 mrem per yr[h]	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic[e] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
	1 sample from milking animals at a control location (Ia4), 15-30 km distant and in the least prevalent wind direction.	Concurrently with indicator locations	

Table 4.11-1

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
b. Fish and Invertebrates	1 sample of each of 2 commercially and/or recreationally important species in vicinity of plant discharge area (Ib1). 1 sample of same species, if available in areas not influenced by plant discharge (Ib2).	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis[e] on edible portions.
c. Food Products	Samples of 3 different kinds of broad leaf vegetation (edible or inedible) grown nearest each of two different offsite locations of highest predicted annual average groundlevel D/Q if milk sampling is not performed (Ic1-Ic2).	Monthly when available	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed (Ic3).	Monthly when available	Gamma isotopic[e] and I-131 analysis.

Table 4.11-1

Radiological Environmental Monitoring Program

Table Notation

a The code letters in parenthesis, e.g. DR1, A1 define generic sample locations. Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 4.11-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.6, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

b One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation.

c The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

Table 4.11-1

Radiological Environmental Monitoring Program

Table Notation (continued)

d Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the previous calendar year mean of control samples, gamma isotopic analyses shall be performed on the individual samples.

e Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

f "Upstream" samples shall be taken near the intake structures as described in the ODCM. "Downstream" samples shall be taken from the mixing zone at the diffuser of the discharge canal.

g A composite sample is one in which the quantity (aliquot) of liquid sample shall be collected at time intervals that are very short (e.g. hourly) relative to the compositing period (e.g. monthly) in order to assure obtaining a representative sample.

h The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

Table 4.11-2

Reporting Levels for Radioactivity Concentrations in Environmental Samples

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

* For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

Table 4.11-3

Detection Capabilities for Environmental Sample Analysis[a]

Lower Limit of Detection (LLD)[b,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

* If no drinking water pathway exists, a value of 3000 pCi/l may be used.

Table 4.11-3

Detection Capabilities for Environmental Sample Analysis[a]

Table Notation

a This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

b Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

c The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

d. LLD for drinking water samples. If no drinking water pathway exists the LLD of gamma isotopic analysis may be used.

4.12 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the inspection and testing of all hydraulic snubbers listed in Table 3.12-1.

Objective

To verify that snubbers will perform their design functions in the event of a seismic or other transient dynamic event.

Specifications

The following surveillance requirements apply to those snubbers listed in Table 3.12-1.

A. VISUAL INSPECTION

Snubbers whose seal material has been demonstrated by operating experience, laboratory testing, or analysis to be compatible with the operating environment shall be visually inspected to verify operability in accordance with the following schedule:

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

* The provision of Section 1.10 of the Technical Specifications is not applicable.

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

Snubbers are categorized in Table 3.12-1 as accessible or inaccessible during reactor operation. These two groups may be inspected independently according to the above schedule except as noted below.

If snubber inoperability is identified due to excessive fluid leakage from the external tubing associated with the twenty-four snubbers installed at the steam generators, this group of snubbers may be inspected independently according to the above schedule.

Visual inspection shall verify that (1) there is no visual indication of damage or impaired operability, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, the snubber has freedom of movement and is not frozen. Snubbers which appear inoperable as a result of visual inspection may be determined operable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and (2) the affected snubber is functionally tested in the as-found condition and determined operable per Specification 4.12.C, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable, and cannot be determined operable via functional testing for the purpose of establishing the next visual inspection period unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

B. FUNCTIONAL TESTING

1. Once each refueling outage, with the provisions of Technical Specification 1.10 applicable, a representative sample of 10% of all the safety-related hydraulic snubbers shall be functionally tested for

operability, including verification of proper piston movement, lock-up rate and bleed. For each hydraulic snubber found inoperable, an additional 10% of the total installed of that type of hydraulic snubber shall be functionally tested. This additional testing will continue until no failures are found or until all snubbers of the same type have been functionally tested.

At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. the first snubber away from each reactor vessel nozzle,

2. snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.), and

3. snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers identified as "Especially Difficult to Remove" or in "High-Radiation Zones During Shutdown" shall also be included in the representative samples.* Table 3.12-1 shall be used as the basis for the sampling plan.

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

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* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions.

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

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

C. FUNCTIONAL TEST ACCEPTANCE CRITERIA

The snubber functional test shall verify that:

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

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

D. RECORD OF SNUBBER SERVICE LIFE

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.n. Concurrently with the first visual inspection and at least once during every refueling outage, the installation and maintenance records for each snubber listed in Table 3.12-1 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement, or reconditioning shall be indicated in the records.

Basis

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible and verified operable by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, and are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. Ten percent of the installed hydraulic snubbers represents an adequate sample for such tests. Selection of a representative sample of hydraulic snubbers provides a confidence level within acceptable limits that these supports will be in an operable condition. Observed failures of these sample snubbers shall require functional testing of additional units of the same type.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high-radiation area, in high-temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide a statistical basis for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operations.

Reference

(1) Report: H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974; Subject: Hydraulic Shock Sway Arrestors

4.13 STEAM GENERATOR TUBE INSERVICE SURVEILLANCE

Applicability

Applies to inservice surveillance of the steam generator tubes.

Objective

To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.

Specifications

Steam generator tubes shall be determined operable by the following inspection program and corrective measures.

A. INSPECTION REQUIREMENTS

1. Definitions

   a. Imperfection is a deviation from the dimension, finish, or contour required by drawing or specification.

   b. Deformation is a deviation from the initial circular cross-section of the tubing. Deformation includes the deviation from the initial circular cross-section known as denting.

   c. Degradation means service-induced cracking, wastage, pitting, wear or corrosion (i.e., service-induced imperfections).

   d. Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

   e. % Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

f. Defect is a degradation of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

g. Plugging Limit is the degradation depth at or beyond which the tube must be plugged.

h. Hot-Leg Tube Examination is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

i. Cold-Leg Tube Examination is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

2. Extent and Frequency of Examination

a. Subject to the conditions of Specification 4.13.C.5 and/or 4.13.C.6, steam generator examinations shall be conducted not later than after sixteen equivalent months of operation (i.e., operation with a primary coolant temperature greater than 350°F) or not later than twenty calendar months from the date of restart after the previous examination, whichever comes first.

b. Scheduled examinations shall include each of the four steam generators in service.

c. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

d. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

e. In case of an unscheduled steam generator examination, the profilometry tensile strain criterion shall be the same as contained in the approved program for the last scheduled steam generator inspection.

3. Basic Sample Selection and Examination

   a. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

   b. At least 25% of the tubes inspected in Specification 4.13.A.3.a above shall be subjected to a cold-leg examination.

   c. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

   d. Examination for deformation ("dents") shall be either by eddy current or by profilometry.

   e. Examination for degradation other than deformation shall be by eddy current techniques, using a 700-mil diameter probe. If the 700-mil diameter probe cannot pass through the tube, a 610-mil diameter probe shall be used. For examination of the U-bends and cold-legs of tubes in rows 2 through 5, a 540-mil diameter probe may be used, provided it is justified by profilometry measurement within the tensile strain criterion.

4. Additional Examination Criteria

   1. Degradation Not Caused by Denting

      a. If 5% of more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 4.13-1.

b. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

c. The second and third sample inspections in Table 4.13-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

2. Degradation Caused by Denting

a. Additional examinations, for degradation caused by denting, shall be performed as described in the most recent steam generator examination program approved by the NRC.

B. ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

1. Tubes shall be considered acceptable for continued service if:

a. depth of degradation is less than 40% of the tube wall thickness, and

b. the tube will permit passage of a 0.540" diameter probe and the strain in the tube wall (if measured) is less than the tensile strain criterion as specified in the approved examination program, or the tube will permit passage of a 0.610" diameter probe, in the absence of strain measurement.

2. Tubes that are not considered acceptable for continued service shall be plugged.

C. REPORTS AND REVIEW AND APPROVAL OF RESULTS

1. The proposed steam generator examination program shall be submitted for NRC staff review and concurrence at least 60 days prior to each scheduled examination.

2. The results of each steam generator examination shall be submitted to NRC within 45 days after the completion of the examination. A significant increase in the rate of denting or significant change in steam generator condition shall be reportable immediately.

3. An evaluation which addresses the long term integrity of small radius U-bends beyond row 1 shall be submitted within 60 days of any finding of significant hourglassing (closure) of the upper support plate flow slots.

4. Restart after the scheduled steam generator examination need not be subject to NRC approval.

5. In any event, NRC staff approval shall be obtained for operating for a period longer than eight equivalent months of operation or one calendar year from the date of restart after examination.

6. In the event of an unscheduled steam generator examination, NRC staff approval shall be obtained in order for the examination to serve as a basis for operation for an additional eight months equivalent operation from the date of the examination.

Basis

Inservice examination of steam generator tubing is essential if there is evidence of mechanical damage or progressive deterioration in order to assure continued integrity of the tubing. Inservice examination of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube examination was performed on each tube in each steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. No significant baseline imperfections were identified. In addition, prior to the discontinuance of phosphate treatment and the institution of all-volatile treatment (AVT), a baseline inspection was conducted in March, 1975 before the resumption of power operation.

Wastage-type defects are unlikely with the all-volatile treatment (AVT) of secondary coolant; however, even if this type of defect occurs, the steam generator tube examination will identify tubes with significant degradation from this effect.

The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness of not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents. An allowance of 10% for tube degradation that may occur between inservice tube examinations added to the 40% degradation depth provided in the acceptance criteria provides an adequate margin to assure that tubes considered acceptable for continued operation will not have a minimum tube wall thickness of less than the acceptable 50% of normal tube wall thickness (i.e. 0.025 inch) during the service life-time of the tubes. Steam generator tube examinations of other operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

Examination of samples of tubes and support plates removed from steam generators have revealed that "denting" is caused by the accretion of steel corrosion products in the tube/support plate annuli. As these corrosion products are more voluminous than the support plate material from which they are derived, a compressive force is exerted on the tubes in the plane of the support plates, resulting in deformation of the tubes. If the deformation results in an ovalization of the tubes, the resulting strain is low and there is no risk of development of stress corrosion cracking in the tubes. However, if the deformation results in an irregular tube shape, the resulting strain may be high enough for the tube to become susceptible to stress corrosion cracking inservice, and it should be preventively repaired. Beginning with the steam generator examination to be conducted during the Cycle 5/6 Refueling Outage, the tensile strain criterion for profilometry shall be 25%. The 25% strain criterion is based on a review of data currently available from operating steam generators, and will be revised as necessary as more experience is gained with the evaluation of this measurement. In the future, this criterion may be revised, either higher or lower, based on steam generator examination results. The profilometry criterion to be used for any steam generator examination shall be established in the most recent program approved by NRC.

A first report on the R&D work leading to the development of profilometry, entitled "Profilometry of Steam Generator Tubes" dated August, 1980, was forwarded to the NRC by Con Edison. Additional R&D work has improved the accuracy of the profilometer and the calculation of strain in a deformed tube.

Before the development of profilometry, a minor diameter of 0.610" was established as the criterion for continuing a tube inservice. This criterion was used successfully for several years at Indian Point Unit 2 and at other plants, and appears to be sufficiently conservative so that it can be continued in the absence of more accurate strain determination by means of profilometry.

This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, dated July 1975.

Tabl -1

Steam Generator Tube Inspection

First Sample Inspection		Second Sample Inspection		Third Sample Inspection		
Minimum Size	Result	Action	Result	Action	Result	Action
	C-1 --	- - - - - - - -	- - - -	- - - - - - - - -	- - >	Go to power.
12% tubes per steam generator hot leg plus 3% tubes per steam generator cold leg	C-2	Plug defective tubes. Inspect additional 6% tubes in this S.G.	C-1 --	- - - - - - - - -	- - >	Go to power.
			C-2	Plug defective tubes. Inspect additional 12% tubes in this S.G	C-1->	Go to power.
					C-2->	Plug defective tubes. Go to power.
					C-3->	Go to first sample. C-3 action.
			C-3	Go to first sample. C-3 action.		
	C-3	Inspect all tubes this S.G. Plug defective tubes. Inspect 6% tubes in each other S.G. if not included in the examination program.	All other S.G.s C-1 --	- - - - - - - - - - -	- - >	Go to power.
			Some S.G.s C-2 but no add'l C-3	Go to second sample. C-2 action.		
			Add'l S.G. C-3	Inspect all tubes in all S.G.s. Plug defective tubes.	- - >	Report to NRC. NRC approval req'd prior to startup.

Table 4.13-1

Steam Generator Tube Inspection

Category C-1	Less than 5% of the total tubes inspected are degraded tubes and none of them is defective.
Category C-2	One or more of the total tubes inspected is defective but no more than 1% of the tubes inspected or less than 10% of the tubes inspected are degraded tubes.
Category C-3	More than 10% of the total inspected are degraded or more than 1% of the tubes inspected are defective.

4.14 FIRE PROTECTION AND DETECTION SYSTEMS

Applicability

This specification applies to the surveillance requirements of fire protection and detection systems provided for protection of safe shutdown systems.

Objective

To verify the operability of fire protection and detection systems.

Specifications

A. HIGH-PRESSURE WATER FIRE PROTECTION SYSTEM TESTING

1. Testing Requirements

Item	Frequency
a. City Water Tank and Fire Water Tank Minimum Water Volume	once/week
b. Diesel-Pump Starting Battery Bank Operability Verify that the electrolyte level of each battery is above the plates, and the overall battery bank voltage is $\geq$ 24 volts.	once/week
c. Main Fire Pump Operability Each pump operating for at least 15 minutes.	once/month
d. Diesel Engine Operability The diesel starts and operates for at least 30 minutes.	once/month

e. <u>Diesel Fire Pump Fuel Supply</u> — once/month

Verify that the diesel-driven fire pump fuel storage tank contains at least 50 gallons of fuel.

f. <u>Valve Position Check</u> — once/month

Verification that each valve (manual, power-operated or automatic) in the flow path necessary for proper functioning of any portion of this system required for protection of safe shutdown systems is in its correct position. If the valve has an installed monitoring system, the valve position can be checked via that monitoring system.

g. <u>Valve Cycling Test</u>

Exercise each valve necessary for proper functioning of any portion of this system required for protection of safe shutdown systems through at least one complete cycle:

(i) Valves testable with plant on-line. — once/12 months

(ii) Valves not testable with plant on-line. — once/18 months

h. <u>System Functional Test</u> — once/18 months

Verification of proper automatic actuation of this system throughout its operating sequence.

i. <u>Main Fire Pump Capacity and System Flow Checks</u> — once/18 months

The motor-driven pumps shall be verified to have a capacity of at least 1500 gpm each at a net pressure of ≥ 93 psig. The diesel-driven pump shall be verified to have a capacity of at least 2500 gpm with a discharge pressure of ≥ 109 psig.

j. <u>Diesel Engine Inspection</u> — once/18 months

Subject the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

k. <u>Diesel Engine Functional Test</u> — once/18 months

Verification that the diesel starts on the auto-start signal and operates for at least 30 minutes while loaded with the fire pump.

l. <u>Diesel Engine Battery Inspection</u> — once/18 months

Verification that the batteries and battery racks show no visual indication of physical damage or deterioration, and that the battery-to-battery terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

m. <u>System Flow Test</u> — once/3 years

Performance of a flow test in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association for any portion of this system required for protection of safe shutdown systems.

B. ELECTRICAL TUNNEL, DIESEL GENERATOR BUILDING AND CONTAINMENT FAN COOLER FIRE PROTECTION SPRAY SYSTEMS TESTING

1. Testing Requirements:

Items	Frequency
a. Valve Cycling Test Exercise each valve necessary for proper functioning of any portion of this system required for protection of safe shutdown systems through at least one complete cycle:	
(i) Valves testable with plant on-line.	once/12 months
(ii) Valves not testable with plant on-line.	once/18 months
b. System Functional Test Includes simulated automatic actuation of spray system and verification that automatic valves in the flow path actuate to their correct position.	once/18 months
c. Spray Header Visual Inspection To verify integrity.	once/18 months

d. Visual Inspection of Each Spray Nozzle — once/18 months

To verify no blockage.

e. Air Flow Test — once/3 years

Perform air flow test through each spray header and verify each spray nozzle is unobstructed.

2. The requirements of Specification 4.14.B.1 shall not apply to self-actuated type spray nozzles which are capable of only one actuation and cannot be periodically cycled or tested. These self-actuated spray nozzles shall be visually inspected at least once per 18 months to verify that no nozzle damage exists and that the nozzles are unobstructed.

C. PENETRATION FIRE BARRIER INSPECTIONS

1. The penetration fire barriers listed in Specification 3.13.C.1 shall be verified to be functional by visual inspection:

   a. At least once per 18 months.

   b. Prior to declaring a fire penetration barrier functional following repairs or maintenance.

D. FIRE DETECTION SYSTEMS TESTING

1. The operability of the fire detection instruments utilized in satisfying the requirements of Specification 3.13.D.1, including the actuation of appropriate alarms (Channel Functional Test), shall be verified as follows:

Item	Frequency
a. Smoke Detectors	
(i) Those testable during plant operation (i.e., all except items 11 and 22 in Table 3.13-1).	once/6 months
(ii) Those not testable during plant operation (items 11 and 22 in Table 3.13-1)	once/18 months
b. Heat Detectors	
(i) Those associated with the Diesel Generator Building (item 7 in Table 3.13-1)	once/6 months
(ii) Those associated with the Electrical Tunnel (item 4 in Table 3.13-1).	once/12 months
(iii) Those associated with the Containment Fan Cooler Units (item 10 in Table 3.13-1).	once/18 months

E. FIRE HOSE STATION AND HYDRANT TESTING

1. Fire hose stations and hydrants described in Specification 3.13.E.1 shall be demonstrated operable by the following surveillance testing requirements:

Item	Frequency
a. <u>Visual Inspection Test</u> Visual inspection of the hose stations and hose houses to assure all required equipment is at the station or hose house.	once/month
b. <u>Hydrant Inspection</u>	
1. Visually inspect each hydrant barrel to verify it is drained.	once/year (in the fall)
2. Flow test each hydrant to demonstrate hydrant and hydrant valve operability.	once/year (in the spring)
c. <u>Hose Removal Check</u> Removal of the hose for inspection and re-placement of all degraded gaskets in couplings.	once/18 months for interior fire hose; once/year for outside fire hose.
d. <u>Hose Flow Test</u> Partial opening of each hose station valve to verify valve operability and no flow blockage.	once/3 years
e. <u>Hose Hydrostatic Test</u> Conduct a hose hydro-static test at a pressure at least 50 psig greater than the maximum pressure available at any hose station.	once/3 years for interior fire hose; once/year for outside fire hose.

F. CABLE SPREADING ROOM HALON SYSTEM

1. The Cable Spreading Room Halon System required operable by Specification 3.13.F.1 shall be demonstrated operable by the following surveillance requirements:

Item	Frequency
a. Halon Storage Tanks Verification of charge weight and pressure.	once/6 months
b. System Functional Test Verification that the system, including ventilation dampers and fans, actuates properly upon receipt of a manual simulated test signal.	once/18 months
c. Air Flow Test Performance of an air flow test through headers and nozzles to verify no blockage.	once/18 months

Basis

These specifications establish the surveillance program for fire protection and detection systems provided to protect equipment utilized for safe shutdown of the unit. This surveillance program is intended to verify the operability of these systems and will identify for corrective action any conditions which could prevent any portion of those systems from performing its intended function.

The fire protection and detection systems are described in Revision 1 to "Review of the Indian Point Station Fire Protection Program" submitted to the NRC by letter dated April 15, 1977 and also in the Fire Protection Safety Evaluation Report issued by the NRC Regulatory Staff in conjunction with Amendment No. 46 to DPR-26 on January 31, 1979.

4.15 RADIOACTIVE MATERIALS SURVEILLANCE

Applicability

Applies to the surveillance of sealed special nuclear, source and byproduct material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specifications

A. Tests for leakage and/or contamination shall be performed as follows:

1. Each sealed source, except startup sources and fission detectors, containing radioactive material, other than Hydrogen-3, with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at six-month intervals.

2. The periodic leak test required does not apply to sources that are stored and not being used. These sources shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Primary startup sources and fission detectors shall be leak tested prior to being subjected to core flux and following repair or maintenance to the source.

B. Sealed sources are exempt from Specification 4.15.A when the source contains 100 microcuries or less of beta- and/or gamma-emitting material or 5 microcuries or less of alpha-emitting material.

C. The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material on the test sample. If the test reveals the presence of 0.005 microcuries or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with Commission regulations.

D. A special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.c within 30 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

Basis

The objective of this specification is to assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed the allowable limits specified in the Code of Federal Regulations.

4.16 REACTOR COOLANT SYSTEM AND CONTAINMENT FREE VOLUME LEAKAGE DETECTION AND REMOVAL SYSTEMS SURVEILLANCE

Applicability

Applies to the surveillance and monitoring of leakage detection and removal systems provided for determining and removing reactor coolant leakage and leakage into the containment free volume. Applies to the testing of certain LPI/RHR check valves[1,2].

Objective

To verify compliance with operational leakage limits of Specification 3.1.F. To specify a test to check for RCS leakage through certain check valves.

Specifications

A. For the purposes of demonstrating compliance with the operational leakage limits of Specification 3.1.F., the following shall be performed:

1. At least once a shift, monitor the leakage detection systems required by Specification 3.1.F.1.a(6).

2. At least once a shift, monitor the containment sump inventory and discharge.

3. At least once a shift, monitor the recirculation sump inventory and the reactor cavity inventory.

4. At least once daily, perform a reactor coolant system water inventory balance.

5. For the RCS/RHR pressure isolation valves, periodic leakage testing* shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for at least 72 consecutive hours if testing has not been

accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

B. A test shall be performed, whenever the RCS pressure decreases to 700 psig (i.e. within 100 psig of the RHR design pressure) or whenever the RHR is secured to go to hot shutdown, to check for leakage through SIS low head injection line check valves 897A-D and RHR check valves 838A-D.

C. The containment sump pumps required to be operable by Specification 3.1.F.1.a(1) shall be demonstrated to be operable by performance of the following surveillance program:

1. At monthly intervals, each sump pump shall be started and a discharge flow of at least 25 gpm verified.

2. At refueling intervals, each sump pump shall be operated under visual observation to verify that the pumps start and stop at the appropriate setpoints and that the discharge flow is at least 25 gpm per pump.

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* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. Minimum test differential pressure shall not be less than 150 psid.

Basis

Specifications 4.16.A and 4.16.C establish the surveillance program for monitoring reactor coolant system leakage and leakage into the containment free volume during plant operation and ensure compliance with Specification 3.1.F. These specifications also establish surveillance requirements for the containment sump pumps. Surveillance requirements for the various leakage detection instrumentation systems are contained in Table 4.1-1 of these specifications.

Specification 4.16.B was added to the Technical Specifications in response to NRC's July 5, 1985 rescission of our February 11, 1980 Confirmatory Order Item A.5. Item A.5 was developed to address the intersystem loss-of-coolant accident (Event V) identified in WASH-1400[1,2]. The RHR system design pressure is 600 psig.

References

(1) NRC letters to Con Edison dated July 5, 1985, and February 11, 1980

(2) Con Edison letter to NRC dated March 14, 1980

4.17 HURRICANE ALERT

Applicability

Applies to the monitoring requirements of a hurricane when Hurricane Warnings are issued for any coastal area south of Indian Point or as far east as New Haven, Connecticut.

Objective

To begin tracking a hurricane's movement for the purpose of taking the actions of Specification 3.14.

Specification

Upon receipt of Hurricane Warnings for the mid-Atlantic coast of the United States, reports issued by the National Weather Service and the National Hurricane Center shall be monitored at least every three (3) hours.

4.18 OVERPRESSURE PROTECTION SYSTEM

Applicability

This specification applies to the surveillance requirements for the OPS provided for prevention of RCS overpressurization.

Objective

To verify the operability of OPS.

Specifications

A. When the OPS PORVs are being used for overpressure protection as required by Specification 3.1.A.4, their associated series MOVs shall be verified to be open at least twice weekly with a maximum time between checks of 5 days.

B. When RCS venting is being used for overpressure protection as permitted by Specification 3.1.A.4, the vent(s) shall be verified to be open at least daily. When the venting pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then only these valves need be verified to be open at monthly intervals.

C. When pressurizer pressure and level control is being used for overpressure protection, as permitted by Specification 3.1.A.4, then these parameters shall be verified to be within their limits at least once per shift.

D. When safety injection pumps and/or charging pumps are required to be de-energized per Specification 3.1.A.4, the pumps shall be demonstrated to be inoperable at monthly intervals by verifying lockout of the pump circuit breakers at the 480 volt switchgear, or once per shift if other means of de-energizing the pumps are used.

E. The PORV backup nitrogen system shall be demonstrated to be operable at refueling surveillance intervals.

Basis

These specifications establish the surveillance program for the Overpressure Protection System (OPS). This surveillance program is intended to verify the operability of the system and will identify for corrective action any conditions which could prevent any portion of the system from performing its intended function.

The PORVs and MOVs associated with the OPS are not included in this specification since the valve cycling and operability tests for these valves are performed in accordance with applicable testing requirements of the ASME Code Section XI and 10 CFR 50.55a.

4.19 METEOROLOGICAL MONITORING SYSTEM

Applicability

This specification applies to the surveillance requirements for the meteorological monitoring system.

Objective

To verify operability of the meteorological monitoring system such that adequate measurement and documentation of meteorological conditions at the site can be effected.

Specifications

A. Each meteorological monitoring instrumentation channel shall be demonstrated operable by performance of the surveillance testing required by Table 4.19-1.

B. Meteorological data shall be summarized and reported as required for inclusion in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6.

Basis

This specification assures the operability of the meteorological monitoring instrumentation and the collection of meteorological data at the plant site. This data is used for estimating potential radiation doses to the public resulting from routine or accidental releases of radioactive materials to the atmosphere. A meteorological data collection program, as described in this specification, is necessary to meet the requirements of 10 CFR 50.36a (a) (2), Appendix E to 10 CFR 50 and 10 CFR 51.

Table 4.19-1

Meteorological Monitoring Instrumentation

Surveillance Requirements

Instrument		Channel Check	Channel Calibration
1. WIND SPEED:			
a. Nominal Elev.	10m*	D	SA
b. Nominal Elev.	60m	D	SA
c. Nominal Elev.	122m	D	SA
2. WIND DIRECTION:			
a. Nominal Elev.	10m	D	SA
b. Nominal Elev.	60m	D	SA
c. Nominal Elev.	122m	D	SA
3. AIR TEMPERATURE DIFFERENTIAL (DELTA T):			
a. Nominal Elev.	60 - 10m	D	SA
b. Nominal Elev.	122 - 10m	D	SA

* 10m as measured by the primary and back-up meteorological tower.

4.20 REACTOR COOLANT SYSTEM VENTS

Applicability

Applies to the periodic testing requirements for the reactor coolant system vents at refueling intervals.

Objective

To verify the operability of the reactor coolant system vents and their ability to exhaust noncondensible gases from the primary system when required.

Specification

A. Each reactor coolant system vent shall be demonstrated operable at refueling intervals by verifying flow through the reactor coolant system vents during cold shutdown.

Basis

The requirement in Specification 4.20.A establishes the surveillance test to be performed at refueling intervals to verify the operability of the reactor coolant system vents. This qualitative flow test will verify that the vents identified in Specification 3.16.A will be available to exhaust gases from the primary coolant system by demonstrating that no blockage exists in the vent system paths.

The periodic testing required by the ASME Code Section XI for each valve in the vents is conducted as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program and is therefore not included in these specifications.

5.0 DESIGN FEATURES

5.1 SITE

Applicability

Applies to the location and extent of the reactor site.

Objective

To define those aspects of the site which affect the overall safety of the installation.

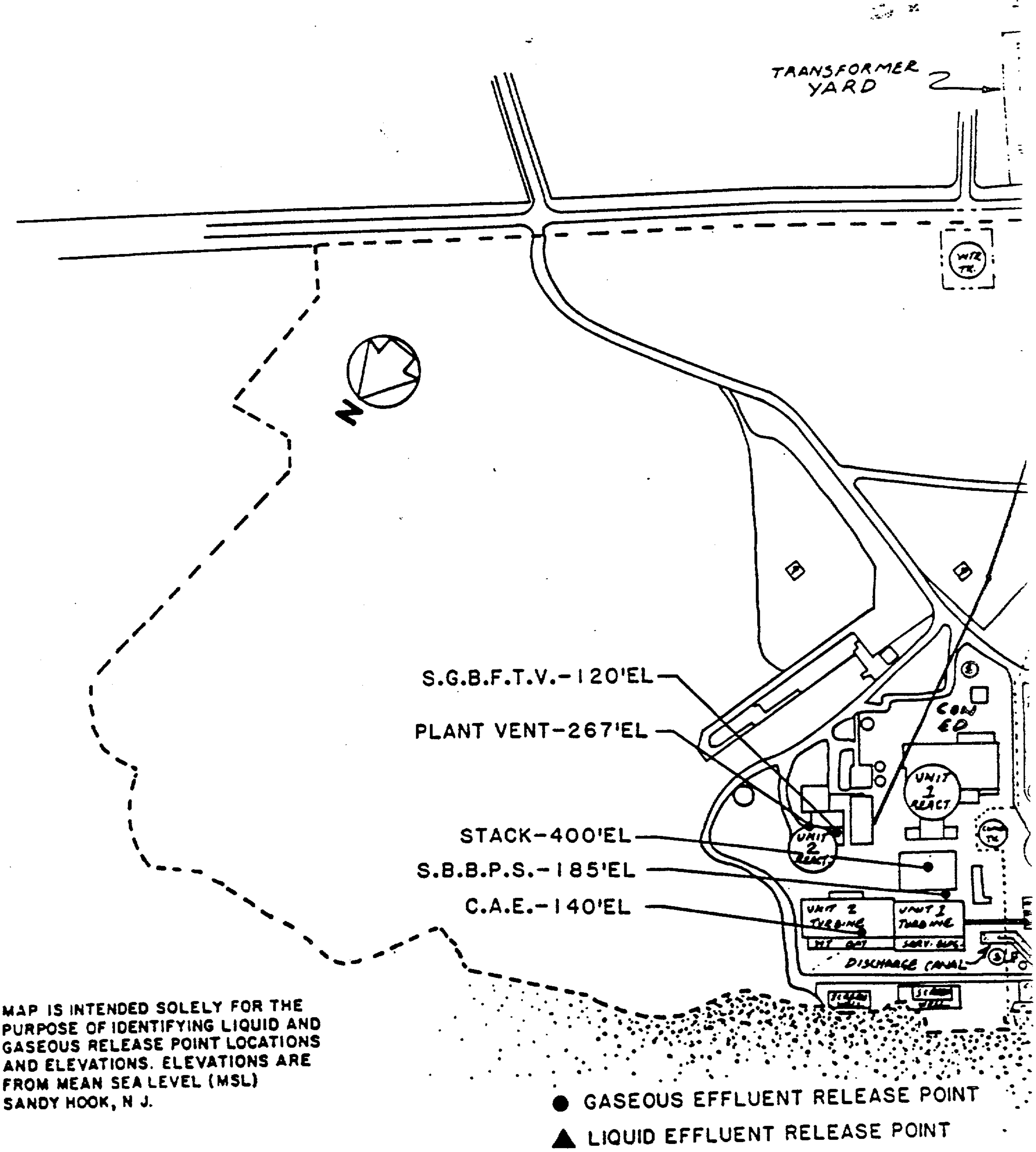
Specifications

A. EXCLUSION AREA AND LOW POPULATION ZONE

The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 520 meters and 1100 meters, respectively. For the purpose of satisfying 10 CFR Part 20, the "Restricted Area" is the same as the "Exclusion Area" defined in Figure 2.2-2 of Section 2.2 of the UFSAR.

B. MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

Information which will allow identification of structures and release points for radioactive gaseous and liquid effluents is shown in Figure 5.1-1. For purpose of effluent release calculations, the unrestricted area shall be as shown in this figure.

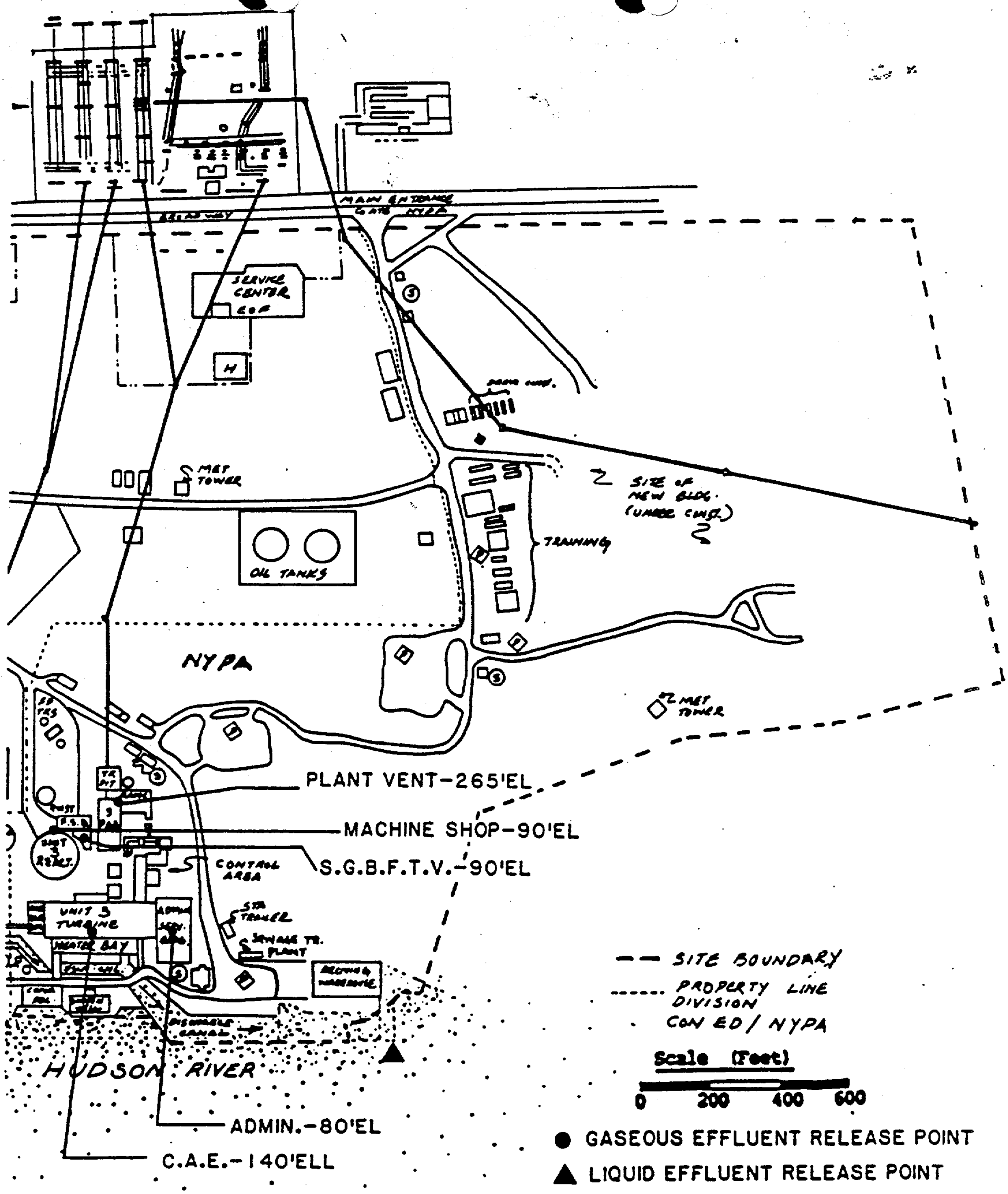


* MAP IS INTENDED SOLELY FOR THE PURPOSE OF IDENTIFYING LIQUID AND GASEOUS RELEASE POINT LOCATIONS AND ELEVATIONS. ELEVATIONS ARE FROM MEAN SEA LEVEL (MSL) SANDY HOOK, N J.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS *

FIGURE 5.1-1

A



* MAP IS INTENDED SOLELY FOR THE PURPOSE OF IDENTIFYING LIQUID AND GASEOUS RELEASE POINT LOCATIONS AND ELEVATIONS. ELEVATIONS ARE FROM MEAN SEA LEVEL (MSL) SANDY HOOK, N.J.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS *

FIGURE 5.1-1

B

5.2 CONTAINMENT

Applicability

Applies to those design features of the Containment System relating to operational and public safety.

Objective

To define the significant design features of the reactor containment structure.

Specifications

A. REACTOR CONTAINMENT

1. The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding for both normal and accident situations.

2. The containment structure is designed for an internal pressure of 47 psig, plus the loads resulting from an earthquake producing 0.10g applied horizontally and 0.05g applied vertically at the same time[(1)]. The containment is also structurally designed to withstand an external pressure 2.5 psig higher than the internal pressure.

B. PENETRATIONS

1. All penetrations through the containment reinforced concrete pressure barrier for pipe, electrical conductors, ducts and access hatches are of the double barrier type[(2)].

2. The automatic Phase A containment isolation (trip) valves are actuated to the closed position either manually or by an automatically-derived safety injection signal. The automatic Phase B containment isolation valves are

tripped closed by automatic or manual containment spray actuation. The actuation system is designed such that no single component failure will prevent containment isolation if required.

C. CONTAINMENT SYSTEMS

1. The containment vessel has an internal spray system which is capable of providing a distributed borated water spray of at least 2200 gpm. During the initial period of spray operation, sodium hydroxide would be added to the spray water to increase the removal of iodine from the containment atmosphere[3].

2. The containment vessel has an internal air recirculation system which includes five fan-cooler units (centrifugal fans and water cooled heat exchangers), with a total heat removal capability of at least 308.5 MBtu/hr under conditions following a loss-of-coolant accident and at service water temperature of 95°F.[4] All of the fan cooler units are equipped with activated charcoal filters to remove volatile iodine following an accident.

References

(1) UFSAR Section 5.1.2.2
(2) UFSAR Section 5.1.4
(3) UFSAR Section 6.3
(4) UFSAR Section 6.4

5.3 REACTOR

Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

Objective

To define those design features which are essential in providing for safe system operations.

A. REACTOR CORE

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods(1).

2. Deleted

3. The enrichment of reload fuel will be no more than 4.3 weight percent U-235 and will be stored in accordance with Technical Specification 5.4.

4. Deleted

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with stainless steel(2).

B. REACTOR COOLANT SYSTEM

1. The design of the reactor coolant system complies with the code requirements(3). Design values for system temperature and pressure are 650°F and 2485 psig, respectively.

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, and with 0% Steam Generator tube plugging is 11,350 cubic feet.

References

(1) UFSAR Section 3.2
(2) UFSAR Section 3.2
(3) UFSAR Table 4.1-9

5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specifications

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stainless steel liner to ensure against loss of water.

2.A. The new fuel storage rack is designed so that it is impossible to insert assemblies in other than an array of vertical fuel assemblies with the sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with the fuel loading in the asemblies limited to 54.33 grams of U-235 per axial centimeter of fuel assembly.

2.B. The spent fuel storage racks are designed and their loading maintained within the limits of Technical Specification 3.8.D.1, such that $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with the fuel loading in the assemblies limited to 56.6 grams U-235 per axial centimeter of fuel assembly.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Vice President-Nuclear Power shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The General Manager-Nuclear Power Generation shall be responsible for facility operations and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 <u>Facility Management and Technical Support</u>

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 Updated FSAR.

b. The General Manager-Nuclear Power Generation 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.

c. The Vice President-Nuclear Power 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 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.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

e. All core alterations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling. This individual shall have no other concurrent responsibilities during this operation.

f. A Fire Brigade of at least five members shall be maintained on-site at all times.* This excludes four members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency. During periods of cold shutdown, the Fire Brigade will exclude two members of the minimum shift crew.

g. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

h. The Operations Manager shall hold a senior reactor operator license.

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* Fire Brigade composition may be one member less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

Table 6.2-1

Minimum Shift Crew Composition**

License Category	During Operations Involving Core Alterations	During Cold Shutdown or Refueling Periods	At All Other Times
Senior Operator License	2*	1	1
Operator License	1	1	2
Non-Licensed	(As Required)	1	2
Watch Engineer	1	(None Required)	1

* Includes individual with SRO license supervising fuel movement as per Specification 6.2.2(e).

** Shift crew composition may be one less than the minimum requirements 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.

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, except for the Radiation Protection Manager who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The General Manager-Nuclear Power Generation shall meet or exceed the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

6.3.3 The Watch Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Director and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Training Director and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976 with the exception of the training program schedule.

6.5 REVIEW AND AUDIT

6.5.1 Station Nuclear Safety Committee (SNSC)

Function

6.5.1.1 The Station Nuclear Safety Committee shall function to advise the Vice President-Nuclear Power on all matters related to nuclear safety.

Composition

6.5.1.2 The Station Nuclear Safety Committee shall, as a minimum, be composed as follows:

Chairman:	General Manager-Technical Services
Member:	Chief Plant Engineer
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Instrument and Control Engineer
Member:	Radiation Protection Manager
Member:	Reactor Engineer

6.5.1.2.1 In addition, other technically competent individuals may be appointed by the SNSC Chairman to serve as SNSC members.

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the SNSC Chairman to serve on a temporary basis, and must have qualifications similar to the member being replaced.

Meeting Frequency

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

Quorum

6.5.1.5 A quorum of the SNSC shall consist of the Chairman or his designated alternate and four members. No more than two alternates shall be included in the quorum.

Responsibilities

6.5.1.6 The Station Nuclear Safety Committee 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 General Manager-Technical Services 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 and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence to the Vice President-Nuclear Power and to the Chairman of the Nuclear Facilities Safety Committee,

f. review of facility operations to detect potential nuclear safety hazards,

g. performance of special reviews and investigations and the issuance of reports thereon as required by the Chairman of the Nuclear Facilities Safety Committee,

h. review of the Plant Security Plan and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee,

i. review of the Emergency Plan and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee,

j. review of any unplanned, radioactive release, including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Power and to the Nuclear Facility Safety Committee, and

k. review of changes to the Process Control Program and the Offsite Dose Calculation Manual.

<u>Authority</u>

6.5.1.7 The Station Nuclear Safety Committee shall:

a. recommend to the Vice President-Nuclear Power, in writing, approval or disapproval of items considered under Specifications 6.5.1.6(a) through (d) above,

b. render determinations, in writing, with regard to whether or not each item considered under Specifications 6.5.1.6(a) through (e) above constitutes an unreviewed safety question, and

c. provide immediate written notification to the Chairman, Nuclear Facilities Safety Committee of disagreement between the recommendations of the SNSC and the actions contemplated onsite. However, the course of action determined by the Vice President-Nuclear Power pursuant to Specification 6.1.1 above or the General Manager-Nuclear Power Generation pursuant to Specification 6.1.2 above shall be followed.

<u>Records</u>

6.5.1.8 The Station Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to, as a minimum, the Vice President-Nuclear Power and the Chairman, Nuclear Facilities Safety Committee.

6.5.2 Nuclear Facilities Safety Committee (NFSC)

Function

6.5.2.1 The Nuclear Facilities Safety Committee shall function to provide independent review and audit of designated activities in the areas of:

a. reactor operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy and non-destructive testing
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. administrative controls and quality assurance practices
i. radiological environmental effects
j. other appropriate fields associated with the unique characteristics of the nuclear power plant

Composition

6.5.2.2 The Committee shall have a permanent voting membership of at least 5 persons of which a majority are independent of the Nuclear Power organization and shall include technically competent persons from departments of Consolidated Edison having a direct interest in nuclear plant design, construction, operation or in nuclear safety. In addition, persons from departments not having a direct interest in nuclear plant design, construction, operation or nuclear safety may serve as members of the Committee if experienced in the field of nuclear energy. The Chairman and Vice Chairman will be senior officials of the Company experienced in the field of nuclear energy.

The Chairman of the Nuclear Facilities Safety Committee, hereafter referred to as the Chairman, shall be appointed by the Chairman of the Board or the President of the Company.

The Vice Chairman shall be appointed by the Chairman of the Board or the President of the Company. In the absence of the Chairman, he will serve as Chairman.

The Secretary shall be appointed by the Chairman of the Committee.

Committee members from departments having a direct interest in nuclear plant design, construction and operation or in nuclear safety shall be designated by the Vice President of the Company, who is responsible for the functioning of the department subject to the approval of the Chairman. Committee members from other departments may be appointed by the Chairman with the concurrence of the Vice President of that department.

Alternates

6.5.2.3 Each permanent voting member, subject to the Chairman's approval, may appoint an alternate to serve in his absence. Committee records shall be maintained showing each such current designation.

No more than two alternates shall participate in activities at any one time.

Alternate members shall have voting rights.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the NFSC Chairman.

Meeting Frequency

6.5.2.5 The NFSC shall meet at least once per calendar quarter or at more frequent intervals at the call of the Chairman or, in his absence, the Vice Chairman.

Quorum

6.5.2.6 A majority of the permanent voting committee members, or duly appointed alternates, which shall include the Chairman or the Vice Chairman and of which a minority are from the Nuclear Power Organization shall constitute a quorum for meetings of the Committee. In the event both the Chairman and the Vice Chairman are absent, one of the permanent voting members will serve as Acting Chairman.

Review

6.5.2.7 The following subjects shall be reported to and reviewed by the Committee insofar as they relate to matters of nuclear safety:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

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

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

d. Proposed changes in Technical Specifications or licenses.

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

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. Reportable Events, as specified by 10 CFR 50.73.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

i. Reports and meeting minutes of the Station Nuclear Safety Committee.

j. Environmental surveillance program pertaining to radiological matters.

Audits

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NFSC. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Radiological Technical Specifications (Appendix A) and applicable license conditions at least once per 12 months.

b. The conformance to all provisions contained within the Environmental Technical Specifications (Appendix B) pertaining to radiological matters and applicable license conditions at least once per 12 months.

c. The performance, training and qualifications of the entire facility staff at least once per 12 months.

d. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

e. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per 24 months.

f. The Facility Emergency Plan and implementing procedures at least once per 12 months.

g. The Facility Security Plan and implementing procedures at least once per 12 months.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

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

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at least once per 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The Offsite Dose Calculations Manual and implementing procedures at least once per 24 months.

m. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

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

o. Any other area of facility operation considered appropriate by the NFSC or the President of the Company.

Authority

6.5.2.9 The NFSC shall report to and advise the President of the Company on those areas of responsibility in Specifications 6.5.2.7 and 6.5.2.8.

Records

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

a. Minutes of each NFSC meeting shall be prepared, approved and forwarded to the President and to Senior Company Officers concerned with nuclear facilities within 14 days following each meeting.

b. Reports of reviews encompassed by Specifications 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the President and to Senior Company Officers concerned with nuclear facilities within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the Senior Company Officers concerned with nuclear facilities and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73a(2).

6.6.1 The following actions shall be taken in the event of a Reportable Event:

a. A report shall be submitted to the Commission pursuant to the requirements of 10 CFR 50.73.

b. Each Licensee Event Report submitted to the Commission shall be submitted to the NFSC Chairman and the Vice President-Nuclear Power and be reviewed by the SNSC.

6.7 SAFETY LIMIT VIOLATION

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

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The Safety Limit Violation Report shall be reported to the Commission, the Vice President-Nuclear Power and to the NFSC Chairman immediately.

c. The Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSC. 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 NFSC Chairman and the Vice President-Nuclear Power within 10 days of the violation.

6.8 PROCEDURES AND PROGRAMS

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

a. The requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USAEC Regulatory Guide 1.33 (issued November 1972) except as provided in 6.8.2 and 6.8.3 below.

b. Process Control Program implementation.

c. Offsite Dose Calculation Manual implementation.

d. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Revision 1, April 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and any changes to them shall be reviewed and approved for implementation in accordance with a written administrative control procedure approved by the appropriate General Manager, with the concurrence of the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. The administrative control procedure required by this specification shall, as a minimum, require that:

a. Each proposed procedure/procedure change involving safety-related components and/or operation of same receives a pre-implementation review by the SNSC except in case of an emergency.

b. Each proposed procedure/procedure change which renders or may render the Updated Final Safety Analysis Report or subsequent safety analysis reports inaccurate and those which involve or may involve potential unreviewed safety questions are approved by the SNSC prior to implementation.

c. The approval of the Nuclear Facilities Safety Committee shall be sought if, following its review, the Station Nuclear Safety Committee finds that the proposed procedure/procedure change either involves an unreviewed safety question or if it is in doubt as to whether or not an unreviewed safety question is involved.

6.8.3 A mechanism shall exist for making temporary changes and they shall only be made by approved management personnel in accordance with the requirements of ANSI 18.7-1972. The change shall be documented, and reviewed by the SNSC and approved by a General Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

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

(i) training of personnel,

(ii) procedures for sampling and analysis, and

(iii) provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

Routine Reports and Reportable Occurrences

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

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) amendments 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. The report shall address each of the appropriate tests identified in the UFSAR 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 specific details required in license conditions based on other commitments shall be included in this report.

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

ANNUAL RADIATION EXPOSURE REPORT[1]

6.9.1.3 Routine reports of occupational radiation exposure data during the previous calendar year shall be submitted no later than March 1 of each year.

6.9.1.4 The annual radiation exposure reports shall provide 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[2], e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty

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

2 This tabulation supplements the requirements of 10 CFR Part 20.407.

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3]

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

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The report shall also include the results of land use censuses required by Specification 4.11.B.

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

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

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps[4] covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 4.11.C; discussion and all deviations from the sampling schedule of Table 4.11-1; and discussion of all analyses in which the LLD required by Table 4.11-3 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT REPORT[5]

6.9.1.6 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

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4 One map shall cover stations near the site boundary; a second shall include more distant stations.

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

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability[6].

This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents releases from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Radioactive Effluent Release Report shall include the following information for each class of solid waste (in compliance with 10 CFR Part 61) shipped offsite during the report period:

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6 In lieu of submission with the first half-year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data onsite in a file that shall be provided to the NRC upon request.

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate),

c. principal radionuclides (specify whether determined by measurement or estimate),

d. source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to Unrestricted Areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Report shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 4.11.B.

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or pressurizer safety valves shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, no later than the 15th of each month following the calendar month covered by the report.

Special Reports

6.9.2 Special reports shall be submitted to the NRC Regional Administrator of the Region I Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate test since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

b. Inoperable fire protection and detection equipment (Specification 3.13).

c. Sealed source leakage in excess of limits (Specification 4.15).

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C.).

e. Radioactive effluents (Specification 3.9).

f. Radiological environmental monitoring (Specification 4.11).

g. Meteorological monitoring instrumentation (Specification 3.15).

h. Inoperable radiation and hydrogen monitoring instrumentation (Specification 3.5) outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

i. Operation of overpressure protection system (Specification 3.1.A.4).

6.10 RECORD RETENTION

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

a. Records and logs of facility operation covering time intervals 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. Reportable Event Reports.

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 Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material on record.

6.10.2 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 Updated Final Safety Analysis Report.

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

c. Records of facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

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

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

i. Records of Quality Assurance activities required by the QA Manual except as noted in 6.10.1.

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

k. Records of meetings of the SNSC and the NFSC.

l. Records for Environmental Qualification which are covered under the provisions of Specification 6.13.

m. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of the service lives of all snubbers listed in Table 3.12-1, including the date at which the service life commences and associated installation and maintenance records.*

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 As an acceptable alternative to the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2):

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of Specification 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Watch Supervisor on duty.

* The documentation referred to herein is required for all snubbers beginning with those replaced following the issuance of Amendment 112.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-26 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines of NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Licensee initiated changes to the PCP:

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

    a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information,

    b. a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes, and

c. docume tion of the fact that the change s been reviewed and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.15.2 Licensee initiated changes to the ODCM:

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

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

   b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations, and

   c. documentation of the fact the change has been revised and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.16 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE SYSTEMS

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change was made. The discussion of each change shall contain:

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

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

c. a detailed description of the equipment, components and processes involved and the interfaces with other plant systems,

d. an evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto,

e. an evaluation of the change, which shows the expected maximum exposures to individuals in the Unrestricted Area and to the general population that differ from those previously estimated in the license application and amendments thereto,

f. a comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes are to be made;

g. an estimate of the exposure to plant operating personnel as a result of the change, and

h. documentation of the fact that the change was reviewed and found acceptable by the SNSC.