ATTACHMENT A

Technical Specification

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TECHNICAL SPECIFICATION

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TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

Section	Title	Page
1	Definitions	1-1
2	Safety Limits and Limiting Safety System Settings	2.1-1
2.1	Safety Limit, Reactor Core	2.1-1
2.2	Safety Limit, Reactor Coolant System Pressure	2.2-1
2.3	Limiting Safety System Settings, Protective Instrumentation	2.3-1
3/3.0	Limiting Conditions for Operation	3.1.A-1
3.1	Reactor Coolant System	3.1.A-1
	Operational Components	3.1.A-1
	Heatup and Cooldown	3.1.B-1
	Minimum Conditions for Criticality	3.1.C-1
	Maximum Reactor Coolant Activity	3.1.D-1
	Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	3.1.E-1
	Reactor Coolant System Leakage and Leakage into the Containment Free Volume	3.1.F-1
	Reactor Coolant System Pressure Temperature, and Flow Rate	3.1.G-1
3.2	Chemical and Volume Control System	3.2-1
3.3	Engineered Safety Features	3.3-1
	Safety Injection and Residual Heat Removal Systems	3.3-1
	Containment Cooling and Iodine Removal Systems	3.3-3
	Isolation Valve Seal Water System	3.3-4
	Weld Channel and Penetration Pressurization System	3.3-4
	Component Cooling System	3.3-5
	Service Water System	3.3-6
	Hydrogen Recombiner System and Post Accident Containment Venting System	3.3-6
	Control Room Air Filtration System	3.3-7
	Cable Tunnel Ventilation Fans	3.3-7
3.4	Steam and Power Conversion System	3.4-1
3.5	Instrumentation Systems	3.5-1
3.6	Containment System	3.6-1
	Containment Integrity	3.6-1
	Internal Pressure	3.6-2
	Containment Temperature	3.6-2
3.7	Auxiliary Electrical Systems	3.7-1
3.8	Refueling, Fuel Storage and Operations with the Reactor Vessel Head Bolts Less Than Fully Tensioned	3.8-1
3.9	DELETED	

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.10	Control Rod and Power Distribution Limits	3.10-1
	Shutdown Reactivity	3.10-1
	Power Distribution Limits	3.10-1
	Quadrant Power Tilt Limits	3.10-3
	Rod Insertion Limits	3.10-4
	Rod Misalignment Limitations	3.10-5
	Inoperable Rod Position Indicator Channels	3.10-5
	Inoperable Rod Limitations	3.10-6
	Rod Drop Time	3.10-6
	Rod Position Monitor	3.10-6
	Quadrant Power Tilt Monitor	3.10-6
	Notification	3.10-7
3.11	Movable In-Core Instrumentation	3.11-1
3.12	Shock Suppressors (Snubbers)	3.12-1
3.13	Fire Protection and Detection Systems	3.13-1
3.14	Hurricane Alert	3.14-1
4	Surveillance Requirements	4.1-1
4.1	Operational Safety Review	4.1-1
4.2	Primary System Surveillance	4.2-1
4.3	Reactor Coolant System Integrity Testing	4.3-1
4.4	Containment Tests	4.4-1
	Integrated Leakage Rate	4.4-1
	Sensitive Leakage Rate	4.4-2
	Air Lock Tests	4.4-3
	Containment Isolation Valves	4.4-3
	Containment Modifications	4.4-4
	Report of Test Results	4.4-5
	Visual Inspection	4.4-5
	Residual Heat Removal System	4.4-5
4.5	Engineered Safety Features	4.5-1
	Safety Injection System	4.5-1
	Containment Spray System	4.5-1
	Hydrogen Recombiner System	4.5-2
	Containment Air Filtration System	4.5-2
	Control Room Air Filtration System	4.5-3
	Fuel Storage Building Air Filtration System	4.5-5
	Post Accident Containment Venting System	4.5-6
4.6	Emergency Power System Periodic Tests	4.6-1
	Diesel Generators	4.6-1
	Diesel Fuel Tanks	4.6-2
	Station Batteries	4.6-2
	Gas Turbine Generators	4.6-2
	Gas Turbine Fuel Supply	4.6-3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
4.7	Main Steam Stop Valves	4.7-1
4.8	Auxiliary Feedwater System	4.8-1
4.9	Reactivity Anomalies	4.9-1
4.10	DELETED	
4.11	DELETED	
4.12	Shock Suppressors (Snubbers)	4.12-1
4.13	Steam Generator Tube Inservice Surveillance	4.13-1
	Inspection Requirements	4.13-1
	Acceptance Criteria and Corrective Action	4.13-7
	Reports and Review and Approval of Results	4.13-8
4.14	Fire Protection and Detection Systems	4.14-1
4.15	Radioactive Materials Surveillance	4.15-1
4.16	Reactor Coolant System and Containment Free Volume Leakage Detection and Removal Systems Surveillance	4.16-1
4.17	Hurricane Alert	4.17-1
4.18	Overpressure Protection System	4.18-1
5	Design Features	5.1-1
5.1	Site	5.1-1
5.2	Containment	5.2-1
	Reactor Containment	5.2-1
	Penetrations	5.2-1
	Containment Systems	5.2-2
5.3	Reactor	5.3-1
	Reactor Core	5.3-1
	Reactor Coolant System	5.3-1
5.4	Fuel Storage	5.4-1
6.0	Administrative Controls	6-1
6.1	Responsibility	6-1
6.2	Organization	6-1
	Facility Management and Technical Support	6-1
	Facility Staff	6-1
6.3	Facility Staff Qualifications	6-1
6.4	Training	6-5
6.5	Review and Audit	6-5
	Station Nuclear Safety Committee	6-5
	1) Function	6-5
	2) Composition	6-5
	3) Alternates	6-5
	4) Meeting Frequency	6-5
	5) Quorum	6-5
	6) Responsibilities	6-6
	7) Authority	6-6
	8) Records	6-7

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
	Nuclear Facilities Safety Committee	6-7
	1) Function	6-7
	2) Composition	6-8
	3) Alternates	6-8
	4) Consultants	6-9
	5) Meeting Frequency	6-9
	6) Quorum	6-9
	7) Review	6-9
	8) Audits	6-10
	9) Authority	6-11
	10) Records	6-11
6.6	Reportable Occurrence Action	6-12
6.7	Safety Limit Violation	6-12
6.8	Procedures	6-12
6.9	Reporting Requirements	6-13
	Routine and Reportable Occurrence Reports	6-13
	Special Reports	6-19
6.10	Record Retention	6-20
6.11	Radiation Protection Program	6-21
6.12	High Radiation Area	6-21

LIST OF TABLES

Frequency Notation	1-1
Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) Pump(s) Operability/Operating Requirements for Decay Heat Removal and Core Mixing	3.1.A-1
OPS Operability Requirements	3.1.A-2
IP2 Reactor Vessel Core Region Material	3.1.B-1
Engineered Safety Features Initiation Instrument Setting Limits	3.5-1
Reactor Trip Instrumentation Limiting Operating Conditions	3.5-2
Instrumentation Operating Condition for Engineered Safety Features	3.5-3
Instrumentation Operating Conditions for Isolation Functions	3.5-4
Table of Indications Available to the Control Room Operator	3.5-5
Non-Automatic Containment Isolation Valves Open Continuously or Intermittently for Plant Operation	3.6-1
Radioactive Liquid Effluent Monitoring Instrumentation	3.9-1
Radioactive Gaseous Effluent Monitoring Instrumentation	3.9-2
Analyses Requiring Evaluation in the Event of an Inoperable Rod	3.10-1
Safety Related Shock Suppressors (Snubbers)	3.12-1
Fire Detection Instruments	3.13-1
Fire Hose Stations and Fire Hydrants	3.13-2
Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	4.1-1
Frequencies for Sampling Tests	4.1-2
Frequencies for Equipment Tests	4.1-3

LIST OF TABLES (Cont'd)

Inservice Inspection Requirements for Indian Point No. 2	4.2-1
Containment Isolation Valves	4.4-1
Steam Generator Tube Inspection	4.13-1
Minimum Shift Crew Composition	6.2-1

LIST OF FIGURES

Safety Limits Four Loop Operation 100% Flow	2.1-1
PORV Opening Pressure for Operation Less Than or Equal to 310^{o}F	3.1.A-1
Maximum Presurizer Level with PORV's Inoperable and One Charging Pump Energized	3.1.A-2
Maximum Reactor Coolant System Pressure for Operation With PORV's Inoperable and One Safety Injection Pump and/or Three Charging Pumps Energized	3.1.A-3
Reactor Coolant System Heatup Limitations	3.1.B-1
Reactor Coolant System Cooldown Limitations	3.1.B-2
Required Hot Shutdown Margin vs. Reactor Coolant Boron Concentration	3.10-1
Hot Channel Factor Normalized Operating Envelope	3.10-2
Insertion Limits, 100 Step Overlap Four Loop Operation	3.10-3
Insertion Limits, 100 Step Overlap Three Loop Operation	3.10-4
Target Band on Indicated Flux Difference as a Function of Operating Power Level	3.10-5
Permissible Operating Band on Indicated Flux Difference as a Function of Burnup	3.10-6
Reactor Coolant System Heatup Limitation	4.3-1
Facility Management and Technical Support Organization	6.2-1
Facility Organization	6.2-2

TECHNICAL SPECIFICATIONS

1 DEFINTIONS

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

1.1 a. Rated Power

A steady state reactor thermal power of 2758 MWT.

b. Thermal Power

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

1.2 Reactor Operating Conditions

1.2.1 Cold Shutdown Condition

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

1.2.2 Hot Shutdown Condition

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 $> 200^{o}F$ and $\leq 550^{o}F$.

1.2.3 Reactor Critical

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

1.2.4 Power Operation Condition

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 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.

1.7 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 3.6.1, are closed and blind flanges are installed where required.

b. The equipment door is properly closed and sealed by the Weld Channel and Penetration and Pressurization System.

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

d. 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.

e. Containment leakage has been verified in accordance with the surveillance requirements of Specification 4.4, and the requirements of Specification 3.3.D are being satisfied.

1.8 Quadrant Power Tilt Ratio

The quadrant power tilt ratio shall be the ratio of maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

1.9 Surveillance Intervals

Unless otherwise noted in an individual surveillance requirement, surveillance intervals shall be as specified in Table 1-1 with extensions as provided in 1.10 below. The extensions provided in 1.10 below also apply to surveillance intervals not listed in Table 1-1 unless the extensions are specifically not allowed.

1.10 Surveillance Interval Maximums

Each surveillance requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

1.11 Pressure Boundary Leakage

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

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

Dose Equivalent I-131 shall be 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 dose conversion factors shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

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-Annualy 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 or P	Prior to each reactor startup	--
N.A.	Not Applicable	--

Amendment No.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure and temperature.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

The Region 1 fuel residence time shall be limited to 21,000 effective full power hours (EFPH) under design operation conditions. The licensee may propose to operate individual assemblies from Region 1 in excess of 21,000 EFPH by providing an analysis which includes the effect of clad flattening or a change in operation conditions. Any such analysis, if proposed, shall be approved by the Regulatory Staff prior to operation in excess of 21,000 EFPH.

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 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the 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 minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves in Figure 2.1-1 represents the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.30. The area where clad integrity is assured is below these lines. The curves are based on the following nuclear hot channel factors[2]:

$$F^N_q = 3.12$$

$$F^N_{\Delta H} = 1.75$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion.[3] The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figures 3.10-3 and 3.10-4 insure that the DNBR is always greater at partial power than at full power.

Rod withdrawal block and load runback occurs if reactor trip setpoints are approached within a fixed limit.

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 1.30.[4]

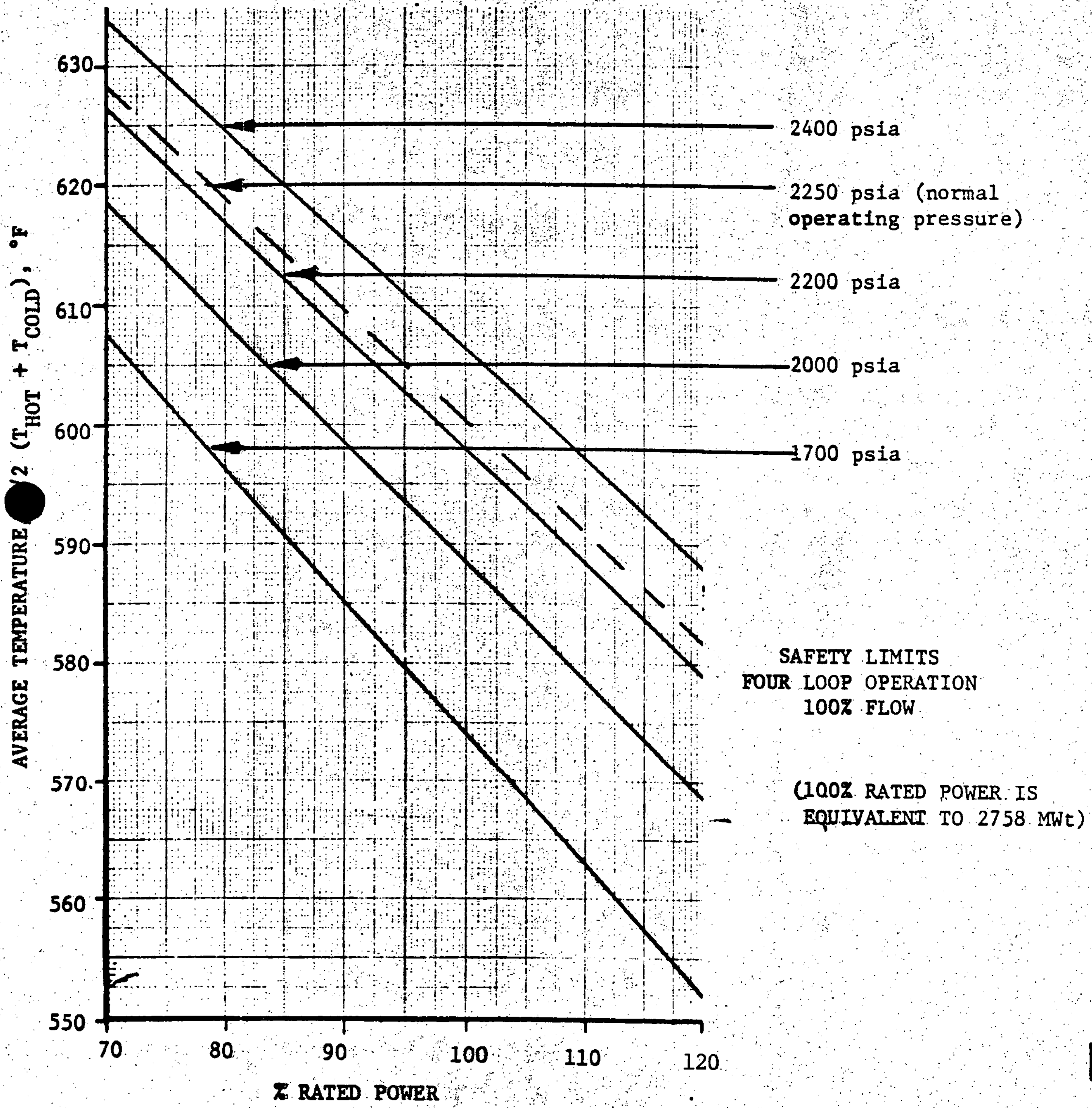
<u>References</u>

1. FSAR Section 3.2.2

2. FSAR Section 3.2.1

3. FSAR Technical Specification 3.10

4. FSAR Section 14.1.1



Amendment No.

FIGURE 2.1-1

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Figure 2.1-2

3 LIMITING CONDITIONS FOR OPERATION

3.0 Compliance with the limiting conditions for operation (LCO) contained in the succeeding Specifications is required when the unit is in the condition designated in the particular specification. When a LCO is not met, except as provided in the individual specification, action shall be initiated within one (1) hour to place the unit in a condition in which the specification is satisfied.

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 Pumps

   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 310°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 Generators

   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 the applicable version of Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift settings shall be set at 2485 psig with $\pm 1\%$ allowance for error.

4. Overpressure Protection System (OPS)

a. Except as permitted by Table 3.1.A-2, the OPS shall be armed and operable when the RCS temperature is $\leq$ 310°F. When OPS is required to be operable, the PORVs shall have settings within the limits shown in Figure 3.1.A-1.

b. The requirements of 3.1.A.4.a may be modified to permit one PORV and/or its associated motor operated valve to be inoperable for a maximum of seven (7) consecutive days. If the PORV and/or its series motor operated valve is not restored to operable status within this seven (7) day period, or if both PORVs or their associated block valves are inoperable, action shall be initiated immediately to place the reactor in a condition where OPS operability is not required.

c. In the event either a PORV(s) or a RCS vent(s) is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Nuclear Regulatory Commission within 30 days pursuant to Specification 6.9.2.f. The report shall describe the circumstances initiating the transient, the effect of the PORV(s) or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

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.

6. <u>Pressurizer Heaters</u>

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 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 remove core decay heat, 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 1.30; 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 a peak RCS pressure exceeding 10CFR50, 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 higher 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 310° F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the PORV's 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 PORV's 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 PORV's. When pressurizer level is less than 30% of span additional restrictions on pressurizer pressure make reliance on the PORV's 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 PORV's 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 282°F and 310°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 instrumentaion 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 instrumention, 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 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.

Reference

1) FSAR Section 14.1.6
2) FSAR Section 9.3.1
3) FSAR Section 14.1.10

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 Tavg $>$ 350°F (Excluding loss of off-site power)	One RCP	Two RCPs	The requirement to have at least one reactor coolant 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 RCP operable, restore a second RCP to operable status within 72 hours or bring the RCS temperature to $\leq$ 350°F. Except for testing, with no RCPs operable, immediately initiate action to bring RCS temperature to $\leq$ 350°F.

Amendment No.

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 $T_{avg} \leq 350^{\circ}F$	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 $10^{\circ}F$ below saturation temperature. 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.

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^{o}F below saturation temperature. With only one pump operable, stay in cold shutdown until a second pump is restored to operable status.

Table 3.1.A-1 (4 of 4)

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)			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: (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 310°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 282°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 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 310°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.

OPS Operability Requirements

Safety Injection and Charging Pumps

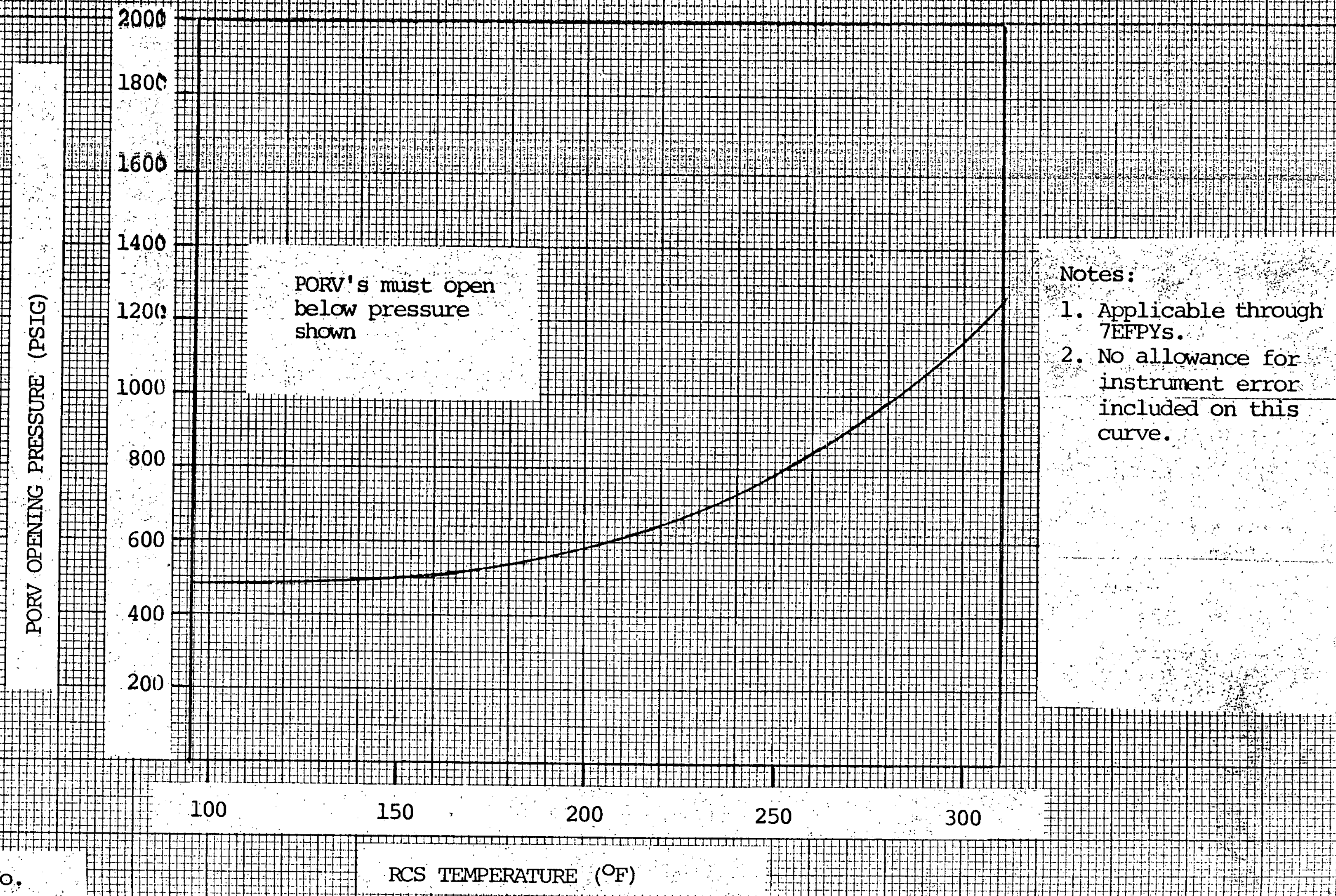
With OPS operable at or below 310°F, no more than one (1) safety injection (SI) and three (3) charging pumps may be energized.

OPS is not required to be operable at or below 310°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

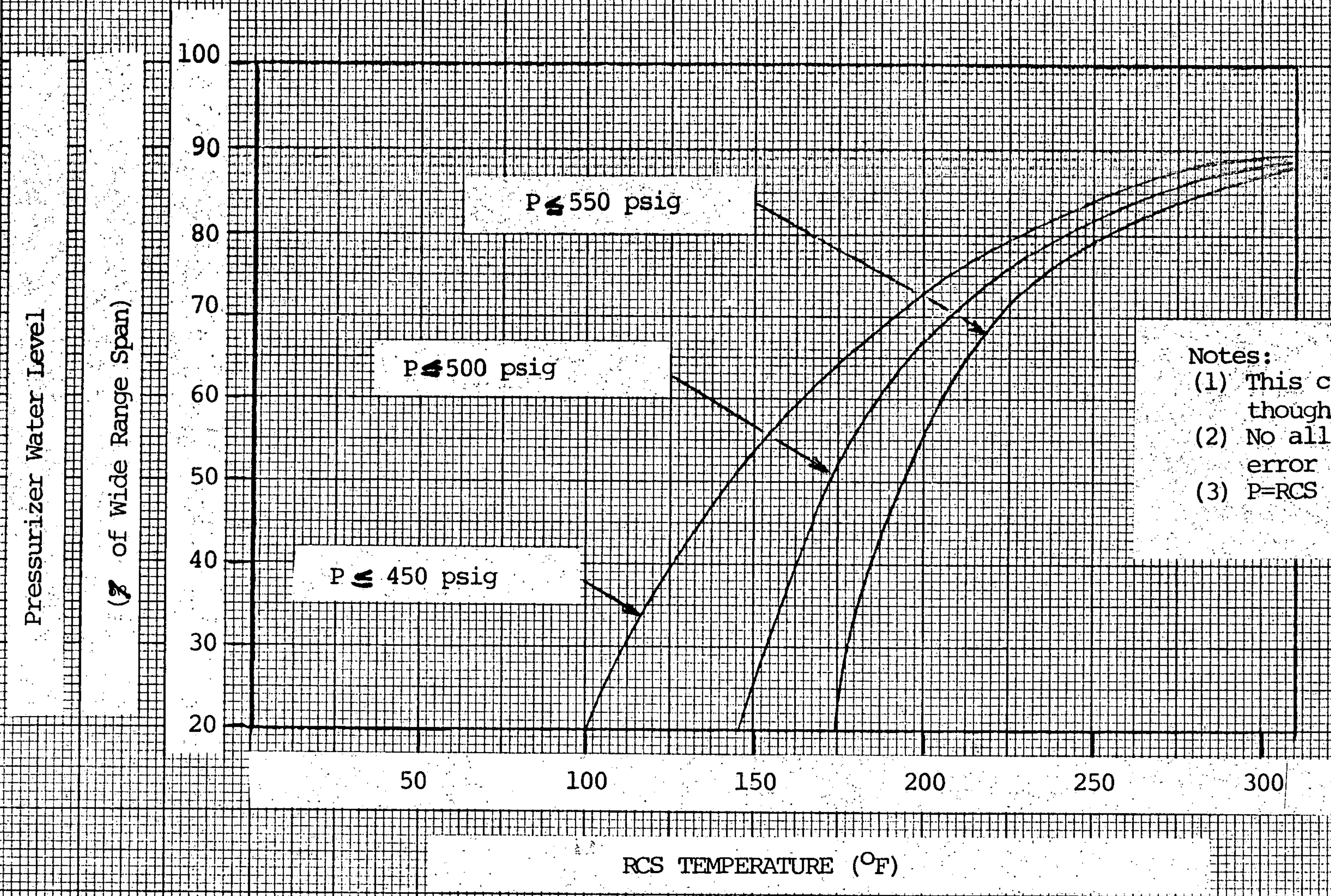
PORV OPENING PRESSURE FOR OPERATION LESS THAN OR EQUAL TO 310°F



Amendment No.

Figure 3.1.A-2

MAXIMUM PRESSURIZER LEVEL WITH PORV's INOPERABLE & ONE CHARGING PUMP ENERGIZED

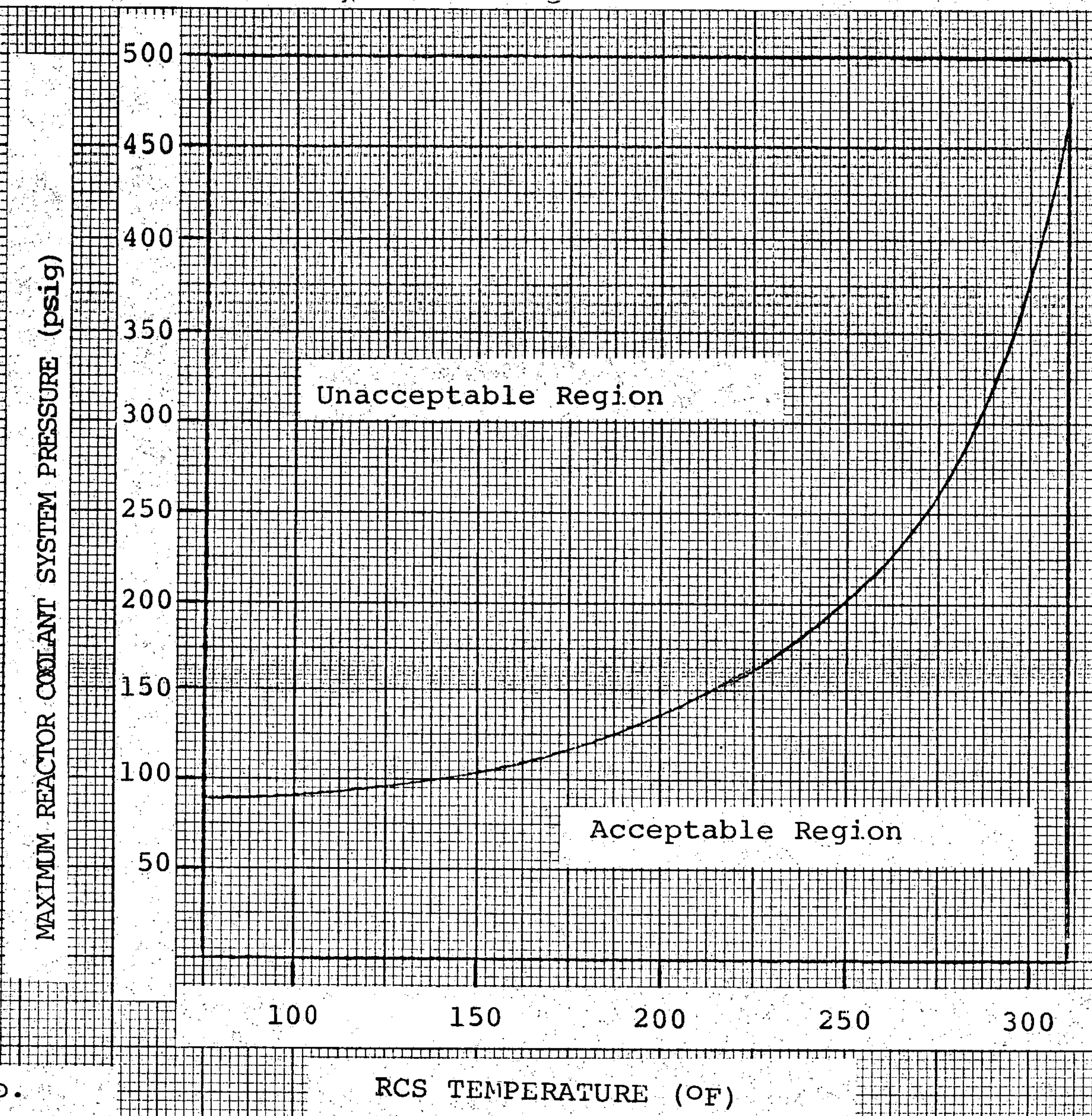


Notes:
(1) This curve is applicable though 7 EFPYs.
(2) No allowance for instrument error included in this curve.
(3) P=RCS pressure.

Amendment No.

Figure 3.1.A.-3

MAXIMUM REACTOR COOLANT SYSTEM PRESSURE FOR OPERATION WITH PORV's INOPERABLE AND ONE SAFETY INJECTION PUMP AND/OR THREE CHARGING PUMPS ENERGIZED



NOTES:

1. Use of this curve requires pressurizer wide range level ≤ 30% of span.

2. No allowance for instrument error included in this curve.

3. Applicable through 7 EFPYs.

Amendment No.

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 7 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 heat up 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.1.3 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 post-irradiation tensile and charpy V notch (wedge open loading) testing of specimens. The specimens will be removed and examined at the following intervals:

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* Pending NRC approval of 100°F/hr heat up curves, reactor coolant system heat up rate shall not exceed 60°F/hr.

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

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°F.

5. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater 320°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 FSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°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 functon.(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 FSAR. 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 by Figure 2-4 of WCAP 7924A[4]. Exposure of the Indian Point Unit No. 2 vessel will be less than that indicated by 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 & 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. This capsule has been tested by SWRI and the results have been evaluated and reported.[10] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed for up to seven (7) effective full power years (EFPYs) of reactor operation.

The maximum shift in RT_{NDT} after 7 EFPYs of operation is projected to be 130°F at the 1/4T and 65°F at the 3/4T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for the IP2 reactor vessel was 60°F based on Plates B2002-1 and B2002-3 as shown in Table 3.1.B-1. The heatup and cooldown curves for 7 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.

<u>Heatup and Cooldown Curves</u>

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 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:

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

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.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4T location and compressive stresses at the 3/4 T position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calcuated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

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 FSAR, Section 4.1.5.

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

(3) Indian Point Unit No. 3 FSAR, 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.

(8) Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.

(9) Supplement to Final Report - SWRI Project No. 02-4531- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.

(10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.

TABLE 3.1.B-1

Indian Point Unit No. 2
Reactor Vessel Core Region Material

Plate	Copper Content(1)	Lowest Temperature 50 ft. lb. Charpy (Longitudinal) (2)	Lowest Temperature 50 ft. lb. Charpy (Transverse)(3)	Assumed RT NDT(4)
B 2002-1	0.25	60°F	120°F	60°F
B 2002-2	0.14	62°F	112°F	52°F
B 2002-3	0.14	75°F	120°F	60°F
HAZ	-	-45°F	5°F	-55°F
Weld Material	-	-10°F	15°F	-45°F

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

(2) Reference: WCAP-7323, "Con Edison Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", Dated May 1969.

(3) Estimated from Longitudinal Data for 77 ft. lb/54 Mil Lateral Expansion (In All Cases, Expansion Data Exceed Requirements).

(4) Reference: ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition Appendix G, $RT_{NDT} = T_{cv} - 60^{o}F$

T_{cv} = Transfer Charpy Temperature at 50 ft. lb energy

Amendment No.

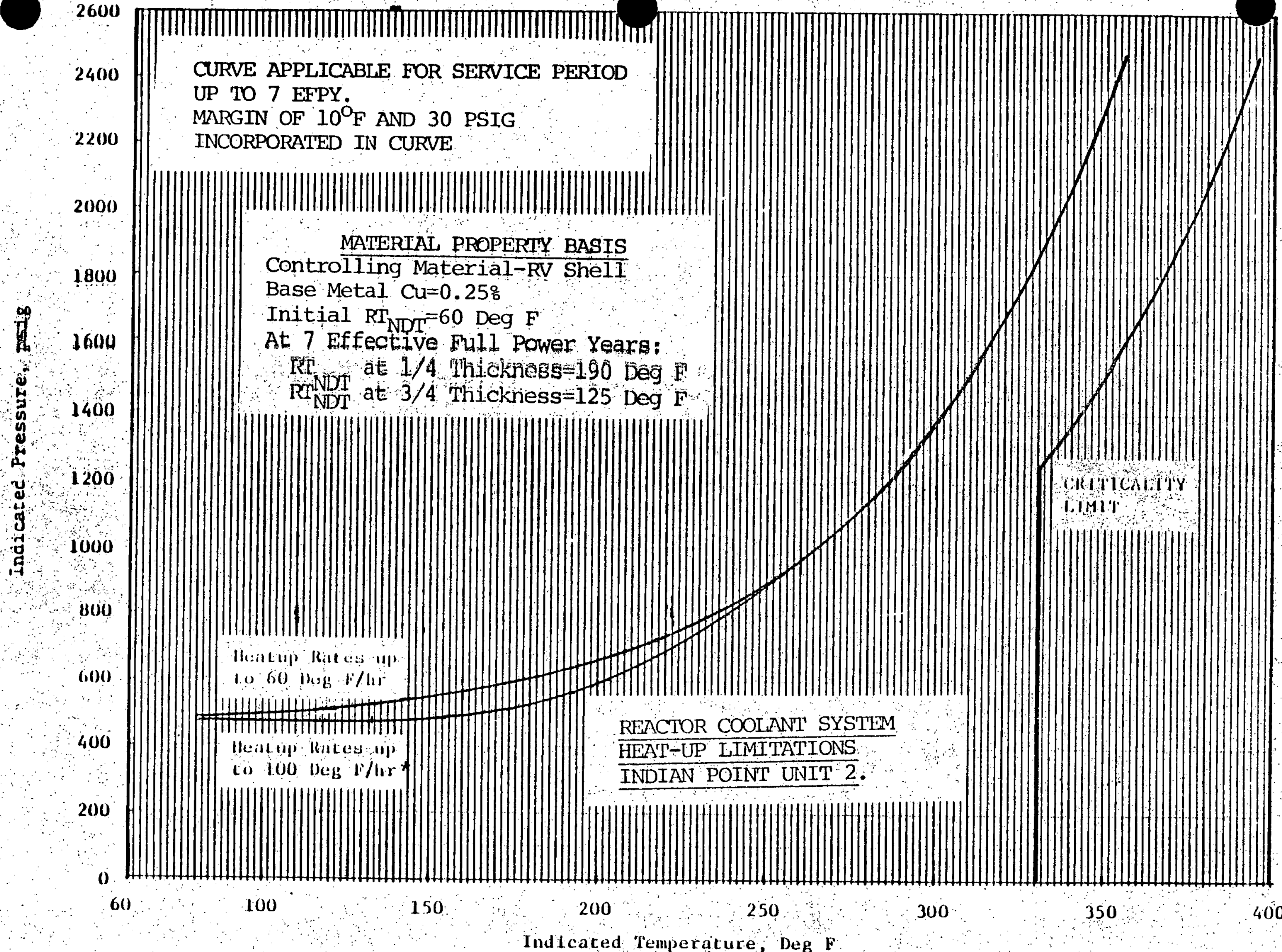

Figure 3.1.B -1

*Pending NRC approval of 100°F/hr heat up curve, reactor coolant system heat up rate shall not exceed 60°F/hr.

Amendment No.

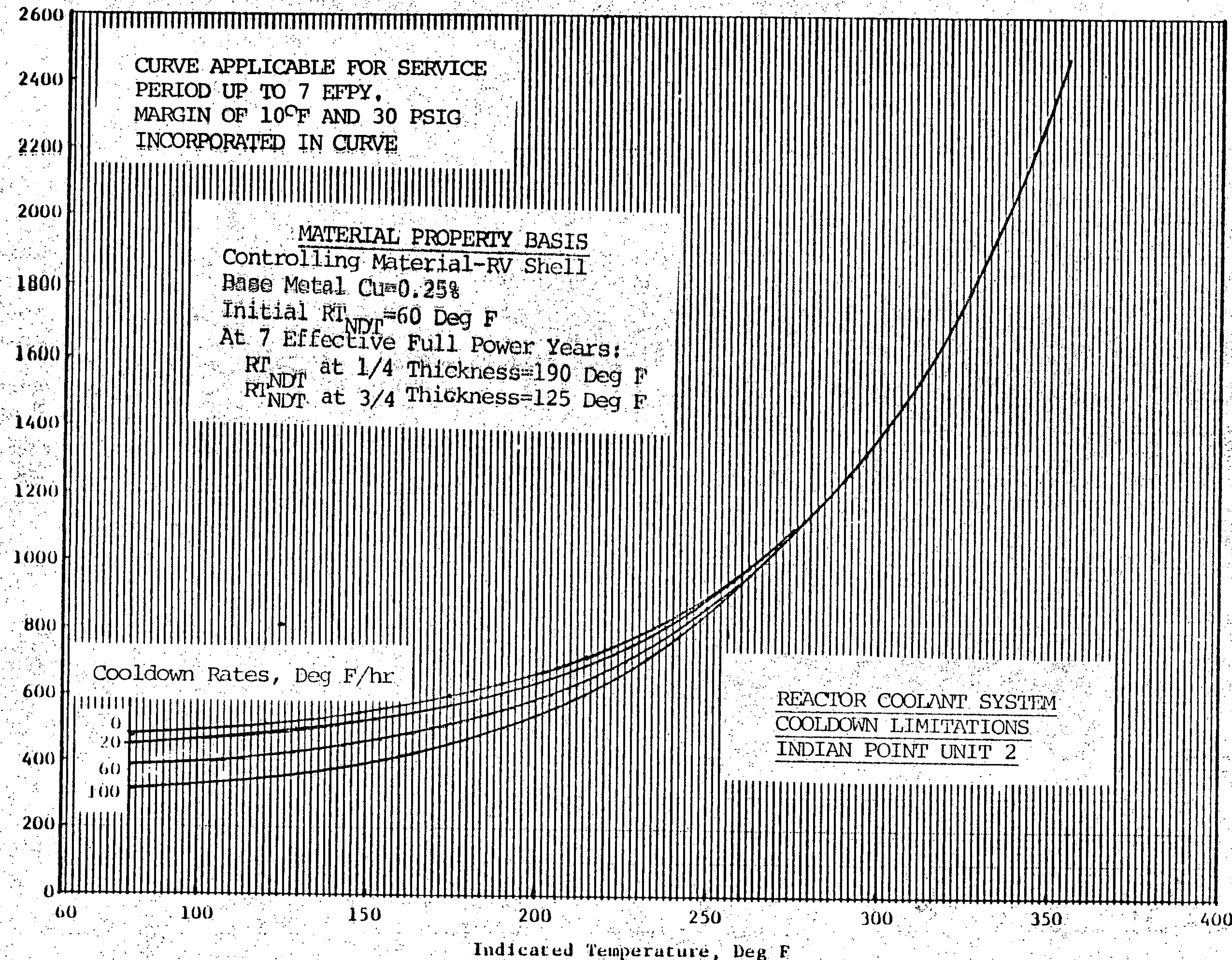

Figure 3.1.B -2

C. MINIMUM CONDITIONS FOR CRITICALITY

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)(2) 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 will be negative. At all times, the moderator coefficient is negative in the power operating range.(1)(2) 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 10CFR50 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. FSAR Table 3.2.1-1

2. FSAR Figure 3.2.1-9

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.

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.[1]

The limiting off-site 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.

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

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

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

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

A = primary coolant activity (Ci/m^3)

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

X/Q = 7.5×10^{-4} sec/m^3, the 0-2 hr. dispersion coefficent at the site boundary(2)

3.7×10^{10} dis/sec - Ci

1.33×10^{-11} rem/Mev/m^3

The resulting dose is 0.5 rem at the site boundary when A is equal to $60/\bar{\bar{E}}$, 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 atmophere 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 $\bar{\bar{E}}$ 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 $\bar{\bar{E}}$ by appropriate weighting of each nuclides beta and gamma energy with its concentration as determined in (1) above.

References

(1) FSAR Table 9.2-5

(2) FSAR Section 11.1.3

(3) FSAR Table 14.2.4

(4) FSAR Table 2.7.3

E. MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Specification

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 can not 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 are 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.[1]

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[2], and further 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 shutdown 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) FSAR Section 4.2

(2) FSAR Section 9.2

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

Specification

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) The reactor cavity ~~continuous~~ level monitoring system and an independent reactor cavity level alarm.
(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) Either the reactor cavity continuous level monitoring system or the required independent reactor cavity level alarm 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 both radioactivity monitoring systems specified in specification 3.1.F.1.a.(6) are inoperable, operation may continue for a period not to exceed thirty (30) days provided grab samples of the containment atmosphere are 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. <u>OPERATIONAL LEAKAGE LIMITS</u>

a. <u>Primary to Secondary Leakge:</u>

(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 with 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 two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown, Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Whenever the reactor is shutdown 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. <u>RCS/RHR Pressure Isolation Valves Leakage:</u>

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

(a) Leakage rates 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. <u>Leakage Into The Containment Free Volume:</u>

(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; and 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 0.025 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in reactor coolant system leakage of 1 gpm is detectable within 1 minute after it occurs.

b. The containment radiogas monitor is less sensitive than the air particulate monitor. The sensitivity range of the instrument is 10^{-3} $\mu c/cc$ to 10^{-6} $\mu c/cc$.

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 switch assemblies 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 transmitter 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 the containment sump inventory and discharge via level and 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 containment floor (EL. 46') sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level switch assemblies 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 transmitter provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a level transmitter which provides a continuous level readout in the control room and two float 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 such 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) prior to the water level being 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

FSAR Sections 6.7, 11.2.3 and 14.2.4

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

Specification

1. During four loop steady state operations at power levels greater than 98%, the following conditions shall be met:

   (a) Reactor Coolant System $T_{avg} \leq 573.5^{o}F$

   (b) Pressurizer pressure $\geq$ 2200 psia; except during a thermal power change in excess of 5% of rated thermal power, or a thermal power step change in excess of 10% of rated thermal power.

   (c) Reactor Coolant System total flow rate $\geq$ 340,800 gpm

2. Should the reactor coolant system temperature or pressure fall outside of the limits specified in 3.1.G.1 above, the parameter(s) shall be restored to its applicable range within two hours or immediately proceed to hot shutdown using normal operating procedures.

Basis

The ranges on reactor coolant system temperature, pressure and loop coolant flow[1] during steady-state four-loop power operation are specified to assure that the values assumed in the accident analysis are not exceeded during normal operation.

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

Compliance with the specified range on RCS total flow is demonstrated by verifying the parameter is within its range at refueling intervals.

Reference:

(1) "Analysis and Evaluation of Non-LOCA Transients for Operation with 95% RCS Thermal Design Flow and with 25% Uniform Steam Generator Tube Plugging," dated April, 1980.

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.

Specification

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 4400 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 310°F, the requirements of Table 3.1.A-2 regarding the number 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 3 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 either the refueling water storage tank or the boron injection 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 4000 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 4400 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°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

FSAR - 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.

Specification

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. The boron injection tank contains not less than 1000 gallons of a 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. Two channels of heat tracing shall be available for the flow path. Valves 1821 and 1831 shall be open and valves 1822A and 1822B shall be closed, except during short periods of time when they can be cycled to demonstrate their operability.

   c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 716 ft^3 and a maximum of 731 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 92,800 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 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. One channel of heat tracing may be out of service for 48 hours.

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 310°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. Containment Cooling and Iodine Removal Systems

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 30% 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. Fan cooler unit 23, 24, or 25 may be inoperable during normal reactor operation for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.

OR

Fan cooler unit 21 or 22 may be inoperable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are demonstrated daily to be operable.

b. One containment spray pump may be out of service during normal reactor operation, for a period not to exceed 24 hours, provided the five fan cooler units are operable and the remaining containment spray pump is demonstrated to be 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 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

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°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. The reactor shall not be made critical unless the following condition is met:

   Three service water pumps on the designated essential header together with their associated piping and valves are operable.

2. If during power operation one of the three service water pumps on the designated essential header or any of their associated piping or valves is found inoperable, the operator shall immediately proceed to place in service an essential service water system which meets the requirements of 3.3.F.1. If an essential service water system cannot be restored within eight hours, the reactor shall be placed in cold shutdown condition.

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 of 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) The containment atmosphere sampling system including the sampling pump, piping and valves is operable.

   d) 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.

   c) One containment atmosphere sampling line may be inoperable for a period not to exceed seven days, provided the other sampling lines are operable.

   d) The containment atmosphere sampling pump may be inoperable for a period not to exceed seven days, provided a spare pump is available at the site for service if required.

H. <u>Control Room Air Filtration System</u>

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 day 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.

I. <u>Cable Tunnel Ventilation Fans</u>

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. For a single component to be inoperable 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 (a) the inoperable component is not repaired within the specified allowable time period, or (b) 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 possiblity 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) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances 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.

The line from the Boron Injection Tank to the high head pump suction piping is provided with four motorized valves; two valves in series with each other and two valves in parallel with each other. Valves 1821 and 1831 are in series and are redundant to each other to assure tank isolation after boron injection, i.e., at least one valve must close. Valves 1822 A and B are in parallel and are redundant to each other, to assure an open path for boron injection following a safety injection signal.

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 occuring 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.[10, 11] 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 6.2-6 of the FSAR.

The requirement regarding the maximum number of SI pumps that can be energized when RCS temperature is less than or equal to 310°F is discussed under specification 3.1.A.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) 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 85°F).[4] 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 off-site 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 off-site 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 off-site 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 off-site 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 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]

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, 10, 11]

Two independent diverse systems are provided for removal of combustible hydrogen from the containment building atmosphere: (a) the hydrogen recombiners, and (b) 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 of the systems 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 back up.

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% 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 cold shutdown condition.

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 95°F. The second fan will start if 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 below a tolerable value of 140°F. However, in order to provide for ample tunnel ventilation capacity, the two ventilation fans are requried 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 unavilable water (4687 gals) in the tank bottom, inaccuracies (6200 gals) in the alarm setpoints, and minimum quantities required during injection (246,000 gals)(12) and recirculation phases (80,000 gals).(12) The minimum RWST (i.e., 345,000 gals) provides approximately 8,100 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 off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.(13) The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 47 psig.

References

(1) FSAR Section 9
(2) FSAR Section 6.2
(3) FSAR Section 6.2
(4) FSAR Section 6.3
(5) FSAR Section 14.3.5
(6) FSAR Section 1.2
(7) FSAR Section 8.2
(8) FSAR Section 9.6.1
(9) FSAR Section 14.3
(10) Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and Appendix K of 10CFR50", dated December 1978, and "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and 10 CFR Part 50, Appendix K", dated April, 1980.

(11) Letter from William J. Cahill, Jr. of Consolidated Edison Company of New York, to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

(12) Indian Point Unit No. 3 FSAR 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.

(13) FSAR Sections 6.5 and 6.6.

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.

Specification

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 safety valves shall be operable (except for testing).

2. Three auxiliary feedwater pumps shall be operable.

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

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

5. The specific iodine activity of the secondary coolant system shall be less than or equal to 0.1 microcuries per gram of dose equivalent I-131.

B. Except as modified by 3.4.C below, if any of the conditions of specification 3.4.A cannot be met, then the following action shall be taken:

1. With one or more main steam line code safety valves inoperable, either restore the inoperable valve(s) to operable status within 4 hours or reduce the high flux, power range setpoint as follows:

Maximum Number of Inoperable Safety Valves on Any Steam Generator	Allowable High Flux, Power Range Setpoint (Percent of rated thermal power)
1	87
2	64
3	42

Otherwise immediately initiate action to bring the plant to hot shutdown and cool the reactor coolant system below 350°F within the next 36 hours.

2. 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 within the next 12 hours and subsequently cool the RCS to below 350°F using normal operating procedures.

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

   c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump train to operable status.

3. With the condensate storage tank inoperable, either:

   a. Restore the condensate storage tank to operable status within six (6) hours; or

   b. Assure the availability of the supply from the city water system within six (6) hours and restore the condensate storage tank to operable status within the next seven (7) days; or

   c. Bring the reactor to hot shutdown within the next six hours and subsequently cool the RCS below 350°F using normal operating procedures.

4. With one main steam isolation valve inoperable, either:

   a. restore the inoperable valve to operable status within six (6) hours; or

   b. Bring the reactor to hot shutdown within the next six hours and subsequently cool the RCS below 350°F using normal operating procedures.

5. With the specific activity of the secondary coolant system greater than 0.1 microcuries/gram of dose equivalent I-131, either restore the value to acceptable limits within six (6) hours or place the reactor in hot shutdown in the next twelve (12) hours and subsequently cool the plant to cold shutdown using normal operating procedures.

C. If when the RCS is 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

A 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 twenty main steam safety valves have a total combined rated capability of 15,108,000 lbs/hr. The total full power steam flow is 13,283,000 lbs/hr, therefore twenty (20) main steam safety valves will be able to relieve the total steam flow if necessary.

Startup and/or power operation is allowable with safety valves inoperable provided the maximum thermal power is limited by a reduction in the high flux, power range trip setpoints. The reactor trip setpoint reductions are derived on the following basis:

$$\text{Setpoint} = \frac{(X)-(Y)(V)}{X} \times (109)$$

Where: V = number of inoperable valves per steam line

X = Total relieving capacity of all safety valves per steam line (lb/hr)

Y = Maximum relieving capacity of any one safety valve (lbs/hr)

In the unlikely event of complete loss of offsite 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 feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. Each auxiliary feedwater pump is capable of providing sufficient feedwater (400 gpm) 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 shutdown. When the condensate storage tank supply is exhausted, city water will be used.

The operability of the main steam isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. The operability of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analysis.

Reference

FSAR - Section 10.4 and 14.1.9

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.

Specification:

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.

3.5.7 When the reactor coolant system is above 350°F, the following remote instrumentation shall be operable:

a. Two level indications for two different steam generators located either at the auxiliary feedwater pump room or at the main feedwater control valves.

b. Pressure indication located at the auxiliary feedwater pump room for the same two steam generators having level indication specified in 3.5.7.a, above.

c. Pressurizer level and pressure indication at the auxiliary feedwater pump room or near the charging pumps.

3.5.8 With the number of operable instrumentation channels less than the minimum number of channels required by Specification 3.5.6 or 3.5.7, either:

a. Restore the inoperable system(s) to operable status within seven (7) days; or

b. Bring the reactor to hot shutdown within the following 12 hours and subsequently cool the RCS below 350°F using normal operating procedures.

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 steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

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

Protection is also provided for a steam line 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 steam line 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 steam line 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.

Steam Line Isolation

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

coolant[2][4] or steam line 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 Steam Line Isolation protects against large loss of coolant[2] or steam line 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 steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis[3].

5. The high steam line flow limit is set at approximately 40% of the full steam flow at the no load to 20% load. Between 20% and 100% (full) load, the trip set point 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 steam line break accident. Coincident low Tavg setting limit for SIS and steam line isolation initiation is set below its hot shutdown value. The coincident steam line 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 steam line break.[3]

<u>Instrument Operating Conditions</u>

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 instrumentaton 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: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; and (b) defeating the ΔT

protection CHANNEL SET that is being fed fom the NIS channel and (c) defeating the power mismatch section of Tavg 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.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The operability of the remote shutdown instrumentation ensures that sufficient capability is available to maintain the plant in a hot shutdown condition from facilities outside of the control room.

Reference

(1) FSAR-Section 7.5

(2) FSAR-Section 14.3

(3) FSAR-Section 14.2.5

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

Table 3.5-1 (1 of 1)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMITS
1.	High Containment Pressure (Hi level)	Safety Injection	$\leq$ 2.0 psig
2.	High Containment Pressure (Hi-Hi level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 30 psig
3.	Pressurizer Low Pressure	Safety Injection	$\geq$ 1700 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	$\leq$ 150 psi
5.	High Steam Flow in 2/4 Steam Lines Coincident with Low Tavg or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	$\leq$ 40% of full steam flow at zero load $\leq$ 40% of full steam flow at 20% load $\leq$ 110% of full steam at full load $\geq$ 540°F Tavg $\geq$ 600 psig steam line pressure
6.	Steam Generator Water Level (low-low)	Auxiliary Feedwater	$\geq$ 5% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	$\geq$ 40% nominal voltage
8a.	480v Emergency Bus Undervoltage (Loss of Voltage)	---	220V + 100V, -20V 3 sec $\pm$ 1 sec
8b.	480v Emergency Bus Undervoltage (Degraded Voltage)	---	403V $\pm$ 5V 180 sec $\pm$ 30 sec

Amendment No.

Table 3.5-2 (1 of 3)
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
1.	Manual	2	1	1	0	Maintain hot shutdown
2.	Nuclear Flow 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 ΔT	4	2	3	2	Maintain hot shutdown
6.	Overpower ΔT	4	2	3	2	Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown
8.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
9.	Pressurizer-Hi Water Level	3	2	2	1	Maintain hot shutdown

Amendment No.

Table 3.5-2 (2 of 3)
REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

		1	2	3	4	5
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	MIN. DEGREE OF REDUNDANCY	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
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	Maintain hot shutdown
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%
15.	Turbine trip (overspeed protection)	3	2	2	1	Maintain hot shutdown

TABLE 3.5-2 (3 of 3)
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
16.	Control Rod Protection ****	3	2	2	1	During RCS cool-down, manually open reactor trip breakers prior to T_{cold} decreasing below 350°F. Maintain reactor trip breakers open during RCS cool-down when T_{cold} is less than 350°F.

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

** If one of two intermediate range channels 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.

F.P.= Rated Power

Amendment No.

Table 3.5-3 (1 of 3)
INSTRUMENTATION OPERATING CONDITION 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
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 Tavg or Low	4 Tavg Signals	2	3	2	
	Steam Line Pressure	4 Pressure Signals	2	3	2	
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

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

Amendment No.

TABLE 3.5-3 (2 of 3)
INSTRUMENTATION OPERATING CONDITION 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
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 $T < 350^{o}F$
	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	$T < 350^{o}F$

Amendment No.

INSTRUMENTATION OPERATING CONDITION 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
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 Specifi-cation 3.1.A.4

TABLE 3.5-4 (1 of 1)
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 Tavg 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
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 OF INDICATIONS AVAILABLE TO THE CONTROL ROOM OPERATOR

Instrument	Number of Indications Available*	Minimum No. of Indications Required to be Operable
1. Containment Pressure	6	1
2. Refueling Water Storage Tank Level	2	1
3. Steam Generator Water Level (Narrow Range)	3 per steam generator	1/generator
4. Steam Line Pressure	3 per steam line	1/line
5. Pressurizer Level (narrow range)	3	2
6. Pressurizer Pressure (narrow range)	4	1
7. Reactor Coolant Pressure (wide range)	2	1
8. Cold Leg Temperature (Wide Range)	4	1
9. Containment Sump Level	3	1
10. Recirculation Sump Level	3	1
11. RCS Subcooling Margin Monitor	1(1)	1(1)
12. PORV Position Indicator	1 per valve(2)	1 per valve(2)
13. PORV Block Valve Position Indicator	1 per valve(3)	1 per valve(3)

Amendment No.

Instrument	Number of Indications Available*	Minimum No. of Indications Required to be Operable
15. Safety Valve Position Indicator (Acoustic Monitor)	1 per valve	1 per valve
16. Auxiliary Feedwater Flow Rate	1 per steam generator	1 per steam generator
17. RHR Flow	4	3

*Either an indicator or alarm is acceptable

Footnotes:

(1) 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.

(2) Except at times when the associated block valve is closed and de-energized.

(3) Except at times when the block valve is closed and de-energized.

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.

Specification

A. Containment Integrity

1. The containment integrity (as defined in 1.7) shall not be violated unless the reactor is in the cold shutdown condition. However, those non-automatic 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.

2. 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.

3. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin $\geq 10\% \frac{\Delta k}{k}$.

4. Except as specified in 3.6.A.5 below, if containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within four hours or the reactor shall be brought to a cold shutdown condition within the next 36 hours, utilizing normal operating procedures.

5. With one or more isolation valve(s) inoperable, maintain at least one isolation valve operable in each affected penetration and either:

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

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

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve 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.

B. Internal Pressure

1. Whenever the reactor is above cold shutdown, the containment pressure shall be maintained between -2 psig and 2 psig.

2. If the above requirement is not met, the containment pressure shall be restored to within the specified limits within one hour or the reactor shall be brought to the cold shutdown condition within the following 36 hours.

C. Containment Temperature

1. The reactor shall not be above cold shutdown unless the containment ambient temperature is greater than 50°F. If this condition is not met, the containment ambient temperature shall be restored to greater than 50°F within eight (8) hours or the reactor shall be brought to cold shutdown within the following 36 hours.

2. The reactor shall not be critical when the containment ambient temperature is less than 90°F or greater than 120°F. If the reactor is critical and the containment ambient temperature falls outside of the above limits, either restore containment ambient temperature to within the allowable limits within eight (8) hours or place the reactor in hot shutdown utilizing normal operating procedures.

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.

The shutdown margins are selected based on the type of activities that are being carried out. The 10% k/k shutdown margin when the head is off precludes criticality under any circumstances, even though fuel is being moved. When the reactor head is not to be removed, the specified cold shutdown margin of 1% k/k precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment calculated peak accident pressure 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.[2] The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

A 50°F minimum containment ambient temperature assures that the minimum service metal temperature of the containment liner is well above the NDT +30°F criterion for liner material. Maintaining the containment temperature between 90°F and 120°F when the reactor is critical not only assures that the minimum liner temperature criteria is met, but also that the internal containment temperatures are in the range assumed in the accident analyses.

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. During periods of normal 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 system. These valves to be open intermittently are under administrative control and are open only 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) FSAR - Section 14.3.5
(2) FSAR - Section 5.5
(3) FSAR - Section 5.1.1.1

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

Specification

A. The following conditions shall be satisfied when fuel is in the reactor vessel and the reactor vessel head is 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.

3. At least one residual heat removal (RHR) pump and heat exchanger shall be 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 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 Specification 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 and RCS temperature and the source range detectors shall be monitored hourly.

B. The following conditions shall be satisfied during the period of time when fuel is in the reactor vessel, the reactor vessel head is being moved, during movement of the upper internals, while loading and unloading fuel from the reactor, or when moving heavy loads greater than 2300 pounds (except for installed crane systems) over the reactor with the reactor vessel head removed:

1. The reactor Tavg shall be less than or equal to 140°F.

2. The minimum boron concentration shall be sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$. 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 131 hours. In the event that more than one region of fuel (72 assemblies or less) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before a continuous interval of 400 hours has elapsed after shutdown.

5. A dead-load test shall be successfully performed on the fuel storage building 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 being made 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 being made in that area.

8. The equipment door and at least one door in each 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 the 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 reactor pressure vessel flange shall be at least 23 feet (El. 92'0") whenever movement of spent fuel is being made.

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. At least 23 feet of water shall be maintained over the top of the irradiated fuel seated in the storage racks. With less than 23 feet (El.92'-7") of water over the top of the irradiated fuel seated in the storage racks, return any suspended irradiated fuel to the storage rack, suspend any further movement of irradiated fuel and initiate action to restore the water level to its required minimum level.

<u>Basis</u>

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 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, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with borated water. The minimum boron concentration of this water will be sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$ in the cold shutdown condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration insure 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 131 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 waiting time of 400 hours required following plant shutdown before unloading the entire reactor core assures that maximum pool water temperature will be within design objectives.

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

(2) Fuel Densification-Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Table 3.3.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

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

Objectives:

To ensure:

1. Core subcriticality after reactor trip,

2. 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. 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 shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

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

(a) $F^N_{\Delta H} \leq 1.55 \left[1 + 0.2\,(1-P)\right]$

(b) For $\leq$ 6% steam generator tube plugging:

$F_Q\,(Z) \leq (2.31/P) \times K(Z)$ for $P > .5$

$F_Q\,(Z) \leq (4.62) \times K(Z)$ for $P \leq .5$

(c) For $>$ 6% but $\leq$ 12% steam generator tube plugging:

$F_Q\,(Z) \leq (2.25/P) \times K(Z)$ for $P > .5$

$F_Q\,(Z) \leq (4.50) \times K(Z)$ 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 3.10-2a (for $\leq$ 6% tube plugging) or Figure 3.10-2b (for $>$ 6% but $\leq$ 12% tube plugging); 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 zero 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 $\pm$ 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 Item 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 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 quandrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 <u>Quadrant Power Tilt Limits</u>

3.10.3.1 Whenever the indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, within two hours the tilt condition shall be eliminated 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) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be reset to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ratio exceeds 1.09 and 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 and realign the rod within two hours. If the rod is not realigned within two hours or if there is no simultaneous indication of a misaligned control rod, the reactor shall be brought to the hot shutdown condition within 4 hours. If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt.

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 <u>Rod Insertion Limits</u>

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 or Figure 3.10-4.

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 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 fails to meet the requirements of 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, and the shutdown margin requirement of Specification 3.10.1 is satisfied, power operation may then continue provided: (1) the power is reduced to 75% within one hour, (2) the high flux, power range trip setpoint is reduced to 85% within five hours, (3) the shutdown margin requirement of Specification 3.10.1 is determined at least once per 12 hours; (4) a power distribution map is obtained from the movable incore detectors within 72 hours and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits, and (5) an evaluation of each accident listed in Table 3.10-1 is performed within five days to confirm that previously analyzed results remain valid for the duration of operation under these conditions.

3.10.8 Rod Drop Time

At operating temperature and full flow , the drop time of each control rod shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage 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.

3.10.11 Notification

Any event requiring plant shutdown or trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR 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 not be less than 1.30 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°F. This is required to meet the initial conditions assumed for 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 minimum 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 Figures 3.10-2a and b 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 3.10-2a and b indicate a peak clad temperature of less than 2200°F for the double-ended cold leg guillotine break with $C_D=0.6$, the worst case break.(1) (2)

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 moveable 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 $F^N_{\Delta H} \leq 1.55/1.08$. 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 moveable 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 basis including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design basis remain inviolate and identify operational anomolies 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 a 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 (Δ I) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = Δ I/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 Figures 3.10-2a and b 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 $\pm$ 5 percent Δ I 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 occuring during these operations.

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 band, 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 resulting from operation within the target band. The instantaneous consequences 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 $\pm$ 1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the $\pm$ 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 1.30 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 buildup. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical 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 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. In the event that the 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 moveable detector system. For a tilt condition of 1.09, an additional 22 hours 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 the range required for low power physics testing. 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 tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition 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 one-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 core 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 $\lesssim 12$ 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 that potential accidents do not result in worse consequence than those analyzed in the safety analysis report. Restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are evaluated to confirm that the results remain valid during future operation.

The required drop time to dashpot entry is consistent with safety analysis.

<u>REFERENCE</u>

1. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10 CFR 50.46 and Appendix K of 10 CFR 50." See also Consolidated Edison Company's letter to NRC dated January 5, 1979 which submitted the results of this reanalysis based on the Westinghouse ECCS Evaluation Model approved by NRC letter to Westinghouse dated August 29, 1978.

2. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in accordance with the acceptance criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K," dated April 1980.

Table 3.10-1

Analyses Requiring Evaluation
in the Event of an
Inoperable Rod

1. Control Rod Insertion Characteristics
2. Control Rod Misalignment
3. Small Break LOCA
4. Large Break LOCA
5. Major Secondary System Pipe Rupture
6. Rupture of a Control Rod Drive
   Mechanism Housing (Control Rod Ejection)

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the in-core instrumentation system, for the recalibration of the excore axial off-set detection system.

Specification

A. During the in-core recalibration of the excore axial off-set detection systems, 75% of the movable detector guide thimbles shall be operable when performing full core flux maps.

B. Power shall be limited to 90% of rated power if re-calibration requirements for excore axial off-set detection system, identified in Table 4.1-1, are not met.

Basis

The Movable In-core Instrumentation System 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 ex-core detectors.

To calibrate the excore detector system, it is only necessary that the Movable In-core 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.[1] Thus the 10% reduction is considered to be very conservative.

Reference

(1) FSAR - Section 7.4

Table 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
	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	
	d. Trip of Main Feed-water 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.	

*Except when block valve operator is deenergized.

Table 4.1-1 (Continued)

Channel Description	Check	Calibrate	Test	Remarks
37. Overpressure Protection System (OPS)	N.A.	R	**	
38. Steam Line Pressure	M	R	R	
39. Instrumentation at auxiliary feed-water pump room				
o Steam Generator Level	M	R	R	
o Steam Generator Pressure	M	R	R	
o Pressurizer Pressure	M	R	R	
o Pressurizer Level	M	R	R	
40. Pressurizer Pressure/Level near charging pumps	M	R	R	
41. Steam Generator Level at main feedwater control valves	M	R	R	

** 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.

Amendment No.

ITEM 6.6 (CATEGORY K-1) - Intergrally-Welded Supports

There are no integrally-welded supports on the valves subject to this examination.

ITEM 6.7 (CATEGORY K-2) - Supports and Hangers

The supports and hangers of the valves subject to this examination shall be visually examined in accordance with Section XI of the code, as shown in Table 4.2-1.

G. Miscellaneous Inspections

ITEM 7.1 - 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 scheduled are shown in Table 4.2-1.

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 year Interval)	Remarks
6.4	G-1	Pressure-retaining bolting		Not applicable	
6.5	G-2	Pressure-retaining bolting	V	100%	Exception is taken for valves which are not accessible.
6.6	K-1	Integrally-welded supports		Not applicable	
6.7	K-2	Supports and hangers	V	100%	Exception is taken for supports and hangers which are not accessible.
		MISCELLANEOUS INSPECTIONS			
7.1		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.

4.3 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.

Specification

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 modifications 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 seven (7) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures 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 seven (7) effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 190^{o}F. The minimum inservice leak test temperature requirements for periods up to seven (7) 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

1. FSAR, Section 4

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.

Specification

A. SYSTEMS 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°F, a test safety injection signal will be applied to initiate operation of the system. The safety injection and residual heat removal 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 856A, C, D & 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 have 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. Containment atmosphere sampling system tests shall be performed at intervals no greater than six months. The test shall include drawing a sample from the fan cooler units and purging the sampling line.

4. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

5. 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 operable:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal absorbers 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 filter or charcoal absorber 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 $\pm$ 10% during filtration unit operation when tested in accordance with ANSI N510-1975. Verify that the flow rate through the charcoal absorbers 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 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 $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

   b) Verifying that with the system operating at ambient conditions and a flow rate 1840 CFM $\pm$ 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 $\pm$ 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 1978, 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 the 6 inches Water Gauge while operating the system at ambient conditions and at a flow rate of 1840 cfm $\pm$ 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 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 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%.

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 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 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 $\pm 10\%$.

   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 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 $\pm$ 10%.

G. Post Accident Containment Venting System

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 and Regulatory Guide 1.52, Revision 2, March 1978, at ambient conditions and at a flow rate 200 cfm $\pm$ 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 $\pm$ 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 $\pm$ 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 $\pm$ 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 inputs 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 ohmmeter 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 (856A, C, D & 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 contanment 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 10CFR100 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 10CFR Part 50, continues to be met.

The fuel storage building air filtration system is designed to filter the discharge of the fuel storage building atmosphre 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 unles 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 10CFR100 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 (4) 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, th charcoal will be replaced with new charcoal which satisfies the requirements for new charcoal outlined in Regulatory Guide 1.52, Revision 2.

References

(1) FSAR Section 6.2
(2) FSAR Section 6.4

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. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, the Nuclear Regulatory Commission shall be notified within 24 hours and an evaluation as to the cause of the discrepancy shall be made to the Nuclear Regulatory Commission within 10 days.

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.

4.18 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.

Specification

A. When the OPS PORV's are being used for overpressure protection as required by specification 3.1.A.4, their associated series MOV's 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.

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 PORV's and MOV's 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.

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. The average enrichment of the initial core is a nominal 2.8 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.3 weight per cent of U-235.[2]

3. The enrichment of reload fuel will be no more than 3.5 weight per cent U-235.

4. Burnable poison rods are incorporated in the initial core. There are 1412 poison rods in the form of 7,8,9,12,16 and 20-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borated pyrex glass cladd with stainless steel.[4]

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.[5]

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.[6] 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 he vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,350 cubic feet.

Reference

(1) FSAR Section 3.2.2 & Sec. 3 of Fuel Densification Indian Point Nuclear Generating Station Unit No. 2, Dated January, 1973

(2) FSAR Section 3.2.1 & Sec. 3 of Fuel Densification Indian Point Nuclear Generating Station Unit No. 2, Dated January 1973

(3) FSAR Section 3.2.1 & Figure 3.3 of Fuel Densification Indian Point Nuclear Generating Station Unit No. 2, Dated January, 1973

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-9

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region I, U.S. Nuclear Regulatory Commission 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 specificaton:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate tests 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. A report covering the X-Y xenon stability tests within three months upon completion of the tests.

c. To provide the Commission with added verifications of the safety and reliability of the pre-pressurized Zircaloy-clad nuclear fuel, a limited program of non-destructive fuel inspections will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup assemblies in each region during the first, second, and third refueling shutdowns. Any condition observed by this inspection which would lead to unacceptable fuel performance may be the object of an expanded surveillance effort. If another domestic plant which contains pre-pressurized fuel of a similar design reaches fuel exposures equal to or greater than at Indian Point Unit No. 2, and if a limited inspection program is or has been performed there, then the program may not have to be performed at Indian Point Unit No. 2. However, such action requires approval of the Nuclear Regulatory Commission. The results of these inspections will be reported to the Nuclear Regulatory Commission.

d. Inoperable fire protection and detection equipment (Specification 3.13).

e. Sealed source leakage in excess of limits (Specification 4.15).

f. Operation of the Overpressure Protection System (Specification 3.1.A.4.).

ATTACHMENT B

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
February, 1983

Safety Assessment

The proposed technical specification modifications, contained in Attachment A to this Application, are responsive to a number of NRC requests for changes in a variety of areas. The NRC Standard Technical Specifications (STS) for Westinghouse PWRs were used for guidance and were applied in a manner consistent with the specific Indian Point Unit No. 2 plant design and the safety or risk significance of the items. The proposed changes are described in further detail below.

By letter dated February 27, 1980 (Schwencer to Cahill), the NRC, in part, requested changes in the areas of overpressure protection and control of heavy loads over reactor fuel. Changes responsive to this request are contained in proposed sections 3.1.A, 3.3, 3.5, 3.8, 4.1 and 4.18. Consolidated Edison's response to NRC's September 7, 1982 letter requesting further information regarding overpressure protection is under preparation and will be provided shortly under separate cover.

NRC's generic April 10, 1980 letter requested further definition regarding use of terms operable/operability and their relationship to the application of limiting conditions for operation (LCOs). The proposed changes to specification 1.3, the addition of specification 3.0 and requirements contained in present specification 3.7 satisfy the intent of that request.

NRC's generic June 11, 1980 letter requested an amendment to the technical specifications to incorporate requirements for redundant decay heat removal capability during all modes of operation. The proposed changes to sections 3.1.A and 3.8 are responsive to this request.

By letters dated July 7, 1980 and June 1, 1982, the NRC requested upgrading certain of the Indian Point Unit No. 2 technical specifications in light of the requirements contained in the NRC's STS for Westinghouse PWRs. As discussed in Consolidated Edison's July 14, 1982 submittal, a number of items originally requested in the July 7, 1980 letter have already been incorporated into the technical specifications by subsequent independent amendments. Two of the other areas addressed, namely, chloride detection requirements and upgraded electrical system specifications (i.e., sections 3.7 and 4.6), are still under development and are scheduled for submittal by July, 1983. Also, the July 7, 1980 letter requested that Consolidated Edison verify the validity of Figure 3.10-1 of the technical specifications. Based on our review and discussions with Westinghouse, this figure is valid and is utilized in each reload safety analysis for reactor refuelings. All other items of the July 7, 1980 letter are addressed by the proposed changes to the various sections of the technical specifications contained in this application.

The June 1, 1982 NRC followup letter to the July 7, 1980 letter also addressed the STS requirements for external flooding conditions. The Indian Point Unit No. 2 FSAR analyzes a number of different hypothetical "worst case" floods. In particular, FSAR section 2.5 notes that even with the incredible simultaneous occurrence of the Standard Project Flood, a failure of the Ashokan Dam, and a storm surge in New York Harbor at the mouth of the Hudson River resulting from a Standard Project Hurricane, no safeguards equipment at

Indian Point Unit No. 2 will be affected by flooding. In addition, the Indian Point Probabilistic Safety Study (IPPSS), submitted on March 5, 1982, examined external flooding as a severe accident initiator. The IPPSS conclusion was that the external flooding contribution to core melt frequency is extremely small (i.e., less than 10^{-8}) and, as such, has essentially no impact on either core melt frequency or plant risk. Nevertheless, plant abnormal operating procedures do exist for initiating action for various conditions of Hudson River water level and the Emergency Plan contains emergency action levels for declaring either a Notification of Unusual Event, an Alert, or a Site Area Emergency based on river/flood level. In light of the above and the fact that the STS notes that external flooding specifications are optional depending on site/plant specific considerations, Consolidated Edison believes that such technical specifications are unwarranted and unnecessary and has not proposed such changes in this Application.

By letter dated July 28, 1980, the NRC requested that Consolidated Edison propose technical specfications for the Containment Purge System as a means of gaining further reduction in the consequences of a fuel handling accident inside containment (FHAIC). Based on subsequent discussions with the Regulatory Staff, it was concluded that an equivalent reduction could be realized by increasing the minimum required waiting time after shutdown before reactor fuel could be moved. Accordingly, the Staff informed Consolidated Edison that based on X/Q of 5.1×10^{-4} sec/m^3, an increase in the minimum waiting time to greater than 130 hours (present technical specifications require 90 hours) would be acceptable. Thus, the proposed changes to section 3.8 contained in this Application require that no movement of reactor fuel be made unless the reactor has been subcritical for at least 131 hours.

NRC's generic August 15, 1980 letter requested that technical specification changes be made to require at least 23 feet of water above the reactor pressure vessel flange during movement of control rods or fuel assemblies with the reactor vessel head removed. The changes proposed to Specification 3.8 address this request.

In addition to responding to the above requests, Consolidated Edison has also included the following proposed changes in this request:

1) Modifications to specification 3.2 to conform the Indian Point Unit 2 specifications more closely with the STS regarding boric acid addition capabilities. Such changes would obviate the need for unnecessary and unwarranted initiations of plant shutdown as have occurred in the past.

2) Modifications to specification 3.1.B and 4.2 to relocate requirements for the reactor vessel surveillance capsule programs. No changes to the program itself are proposed.

3) Modifications to specification 4.5 to satisfy the technical specifications by including a flow rate for testing the post accident containment venting system. This flow was determined during the recently completed Cycle 5/6 refueling/maintenance outage. In addition, based on other testing performed during the outage, the required Fuel Storage Buildng Air Filtration System testing flow rate is also proposed to be modified.

Since a large number of sections of the technical specifications have been modified in this Application, a significant amount of retyping of the technical specifications was necessary. In many cases, simple changes to both page numbers and specification section numbers were necessary to properly incorporate the requested revisions. Thus, many technical specification pages are included in this Application for solely editorial reasons. While many sections of the technical specifications appear different, only those sections marked with change bars contain wording changes.

The proposed changes have been reviewed by the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.