October 23, 1985

Docket No. 50-247

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Distribution	Docket file
NRC PDR	L PDR
ORB#1 RDG	Gray file
HThompson	ORAB
CParrish	MSlosson
DNeighbors	OELD
EJordan	BGrimes
JPartlow	TBarnhart (4)
ACRS (10)	OPA, CMiles
SECY	RDiggs

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated February 14, 1983 as supplemented February 28, 1985. In addition, by letters dated August 1, 1985 and September 13, 1985 Consolidated Edison provided changes which were inadvertently left out of the February 28, 1985 submittal. These changes were administrative in nature and in no way changed the meaning or technical content of the earlier submittals.

The amendment revises the Technical Specifications to incorporate requirements for redundant decay heat removal capability during all modes of plant operation. The amendment revises the number of reactor coolant pumps required to be operating when the reactor coolant system is shutdown and above 350°F. This change was made to achieve consistency between the safety analysis for the Uncontrolled Rod Withdrawal from Subcritical transient and the Technical Specifications concerning the number of reactor coolant pumps in operation. The amendment revises the Technical Specifications to incorporate specifications on the Overpressure Protection System at Indian Point Unit 2. This system was installed to provide added assurance that the plant can operate without violating pressure limitations. Certain editorial and format changes have also been made.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/MSlosson

Marylee M. Slosson, Project Manager
Operating Reactors Branch #1
Division of Licensing

8511070292 851023
PDR ADOCK 05000247
P PDR

Enclosures:
1. Amendment No. 101 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

*SEE PREVIOUS WHITE FOR CONCURRENCE

ORB#1:DL*	~~ORB#1:DL~~	BC-ORB#1:DL	~~OELD~~	~~AD:OR:DL~~
CParrish	MSlosson/ts	SVarga		GLainas
08/23/85	10/4/85	10/9/85	10/2/85	10/22/85

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Sincerely,

~~Joseph D. Neighbors~~, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. to DPR-26
2. Safety Evaluation

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See next page

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CParrish	MSlosson/ts	DNeighbors	SVarga		GLainas
07/22/85	07/ /85	07/ /85	07/ /85	07/ /85	07/ /85

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.101
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated February 14, 1983, as supplemented Feburary 28, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

8511070297 851023
PDR ADOCK 05000247
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 101 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 60 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 23, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
iii	iii
v	v
vii	vii
3.1-1	3.1.A-1
3.1-2	3.1.A-2
3.1-2a	3.1.A-3
3.1-3	3.1.A-4
----	3.1.A-5
----	3.1.A-6
----	Table 3.1.A-1 (4 pages)
----	Table 3.1.A-2 (2 pages)
----	Figure 3.1.A-1
----	Figure 3.1.A-2
----	Figure 3.1.A-3
3.1-4	3.1.B-1
3.1-4a	3.1.B-2
3.1-5	3.1.B-3
3.1-6	3.1.B-4
3.1-7	3.1.B-5
3.1-8	3.1.B-6
3.1-8a	Table 3.1.B-1
3.1-8b	----
Figure 3.1-1	Figure 3.1.B-1
Figure 3.1-2	Figure 3.1.B-2
3.1-9	3.1.C-1
3.1-10	3.1.C-2
3.1-11	3.1.D-1
3.1-12	3.1.D-2
3.1-13	3.1.D-3
3.1-14	3.1.E-1
3.1-15	3.1.E-2
3.1-16	----
3.1-17	3.1.F-1
3.1-18	3.1.F-2
3.1-19	3.1.F-3
3.1-20	3.1.F-4
3.1-21	3.1.F-5
3.1-22	3.1.F-6
3.1-23	3.1.F-7
----	3.1.F-8
3.3-1	3.3-1
3.3-2	3.3-2

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.3-3	3.3-3
3.3-4	3.3-4
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
3.3-8	3.3-8
3.3-9	3.3-9
3.3-10	3.3-10
3.3-11	3.3-11
3.3-12	3.3-12
3.3-13	3.3-13
3.3-14	3.3-14
3.3-15	----
3.3-16	----
3.3-17	----
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
3.5-4	3.5-4
3.5-5	3.5-5
3.5-6	----
Table 3-1	Table 3.5-1
Table 3.2 (3 pages)	Table 3.5-2 (3 pages)
Table 3-3 (2 pages)	Table 3.5-3 (3 pages)
Table 3-4	Table 3.5-4
Table 3-5 (2 pages)	Table 3.5-5 (2 pages)
Table 4.1-1 (continued)	Table 4.1-1 (continued) Items 31-36
----	Table 4.1-1 (continued) Item 37-41
4.3-1	4.3-1
4.3-2	4.3-2
----	4.9-1
----	4.18-1
5.3-1	5.3-1
5.3-2	5.3-2
6-20	6-20

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	Radioactive Effluents	3.9-1

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	Radioactive Effluents	4.10-1
4.11	Radiological Environmental Monitoring	4.11-1
4.12	Shock Supressors (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
4.19	Meteorological Monitoring System	4.19-1
4.20	Reactor Coolant System Vents	4.20-1
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	6-5
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) Alternatives	6-5
	4) Meeting Frequency	6-5
	5) Quorum	6-5
	6) Responsibilities	6-6
	7) Authority	6-6
	8) Records	6-7

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 FIGURES

Safety Limits Four Loop Operation 100% Flow	2.1-1
PORV Opening Pressure for Operation Less Than or Equal to 310°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
Unrestricted Areas for Radioactive Gaseous and Liquid Effluents	5.1-1
Facility Management and Technical Support Organization	6.2-1
Facility Organization	6.2-2

3 LIMITING CONDITIONS FOR OPERATION

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

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

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

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

A. OPERATIONAL COMPONENTS

1. Coolant Pump

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

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

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

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

2. Steam Generator

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

3. Safety Valves

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

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 ±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 ≤ 295°F. When OPS is required to be operable, the PORV will 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. Pressurizer Heaters

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

b. If the requirements of specification 3.1.A.6.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or the reactor shall be placed in the hot shutdown 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 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 the peak RCS pressure from 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 hotter than the RCS water temperature.

2. Letdown isolation with three charging pumps operating.

3. Startup of one safety injection pump.

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

5. Inadvertant activation of the pressurizer heaters.

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

The RCS is protected against overpressure transients when RCS temperature is less than or equal to 295°F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the 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 267°F and 295°F. The purpose of the restriction is to assure that the temperature rise resulting from the transient will not be outside the temperature limits for OPS actuation.

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

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

(a) Assuming that the secondary side water temperature is at the saturation temperature corresponding to the secondary side steam pressure indicated on the Control Room 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 the saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperatures and pressure.[2]

If not residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

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

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

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

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

<u>Reference</u>

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

Table 3.1.A-1

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

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

Table 3.1.A-1

Reactor Coolant (RC) Pumps/Residual Heat Removal (RHR) and 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 $\leq$ 350°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°F below saturation temperature. If a pump has not been restored to operating status within the one hour permitted, take action as listed below for no operable pumps. With only one pump (RHR or RCP) operable, either restore a second pump to operable status or be in cold shutdown within 20 hours. With no pumps operable, suspend all operations involving a reduction in boron concentration and immediately initiate action to restore at least one pump to operable status.

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

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

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

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

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

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

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

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

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

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

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

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

Table 3.1.A-2

OPS Operability Requirements

Safety Injection and Charging Pumps

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

OPS is not required to be operable at or below 295°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

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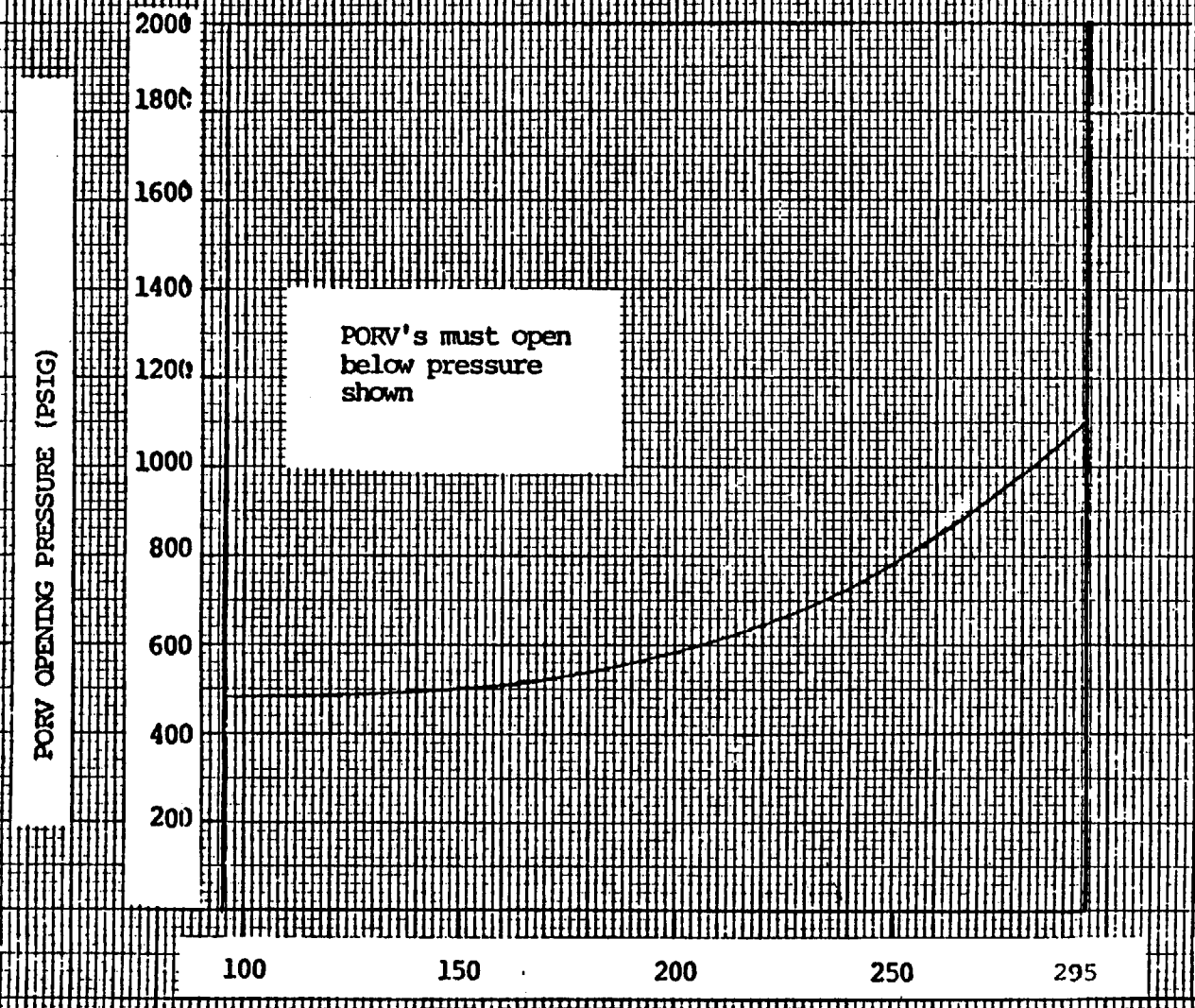
Figure 3.1.A-1

PORV OPENING PRESSURE FOR OPERATION LESS THAN OR EQUAL TO 295°F



Notes:

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

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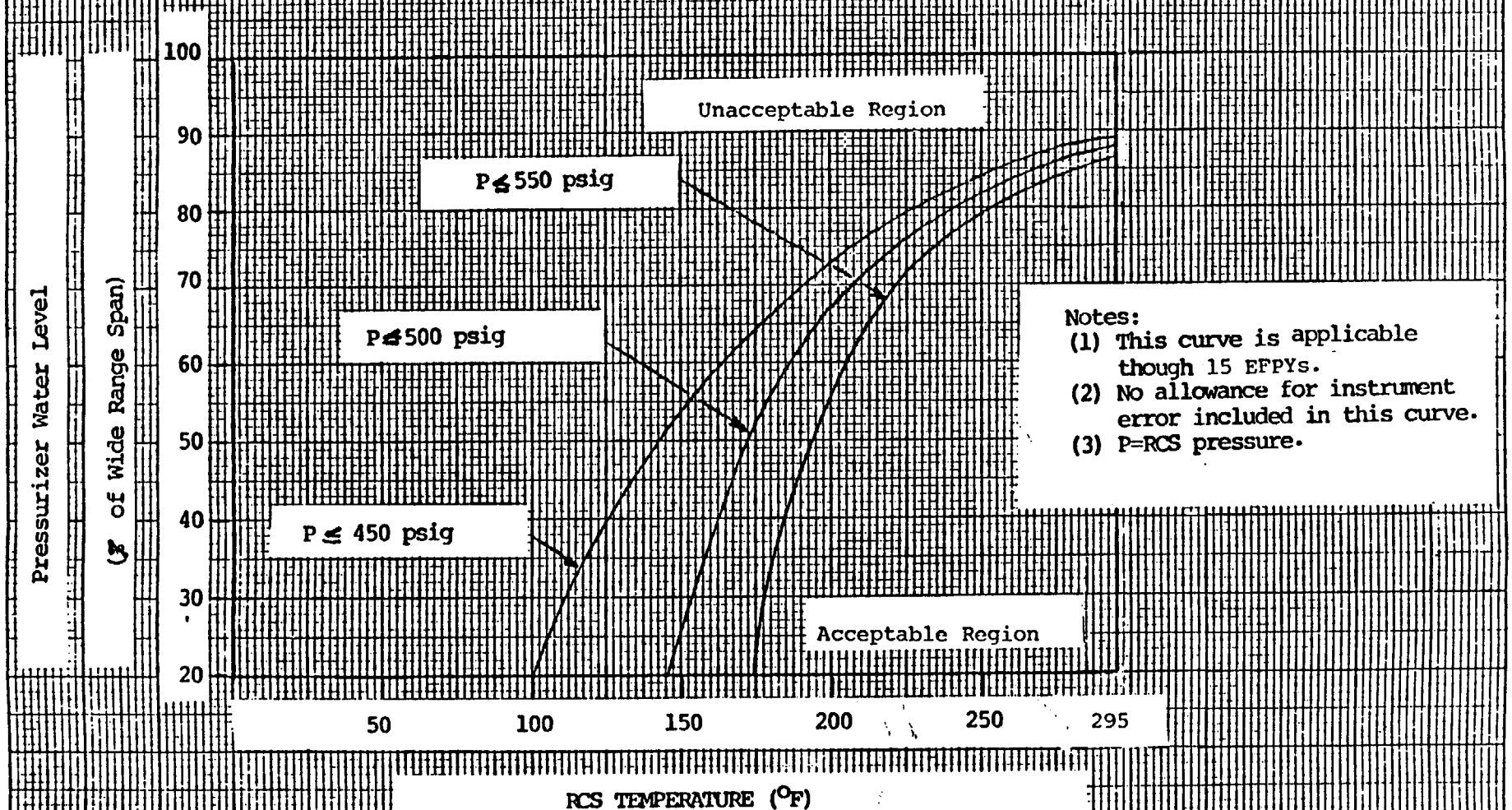
Figure 3.1.A-2

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



Notes:
(1) This curve is applicable though 15 EFPYs.
(2) No allowance for instrument error included in this curve.
(3) P=RCS pressure.

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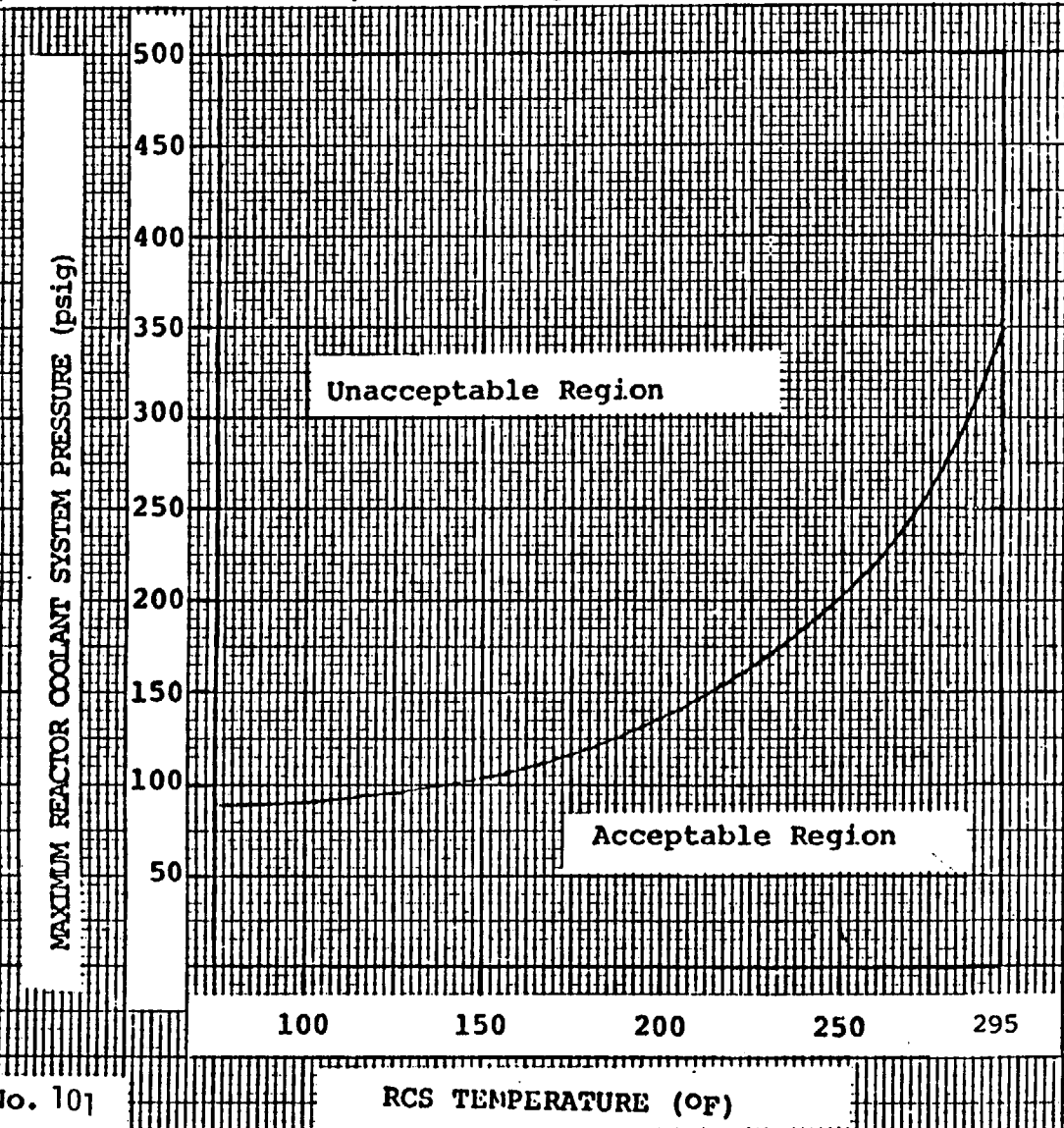
Figure 3.1.A.-3

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 15 EFPYs.

B. HEATUP AND COOLDOWN

Specifications

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

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

   b. Figure 3.1.B-1 and Figure 3.1.B-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the 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.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:

* 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 function.(3)

As a result of fast neutron irradiation in the region of the core, there will be an increase in the Reference Nil-Ductility Transition Temperature (RT_{NDT}), with nuclear operation. The techniques used to measure and predict the integrated fast neutron ($E > 1$ Mev) fluxes at the sample location are described in Appendix 4A of the 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. That capsule has been tested by SWRI and the results have been evaluated and reported.[10] The third vessel material surveillance capsule was removed during the 1982 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported.[11] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed for up to fifteen (15) effective full power years (EFPYs) of reactor operation.

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

Heatup and Cooldown Curves

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

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

TABLE 3.1.B-1

Indian Point Unit No. 2
Reactor Vessel Core Region Material

Plate	Copper Content	Initial RT NDT
B 2002-1	0.25	34°F
B 2002-2	0.14	21°F
B 2002-3*	0.14	21°F
HAZ	-	0°F
Weld Material	-	0°F

References:

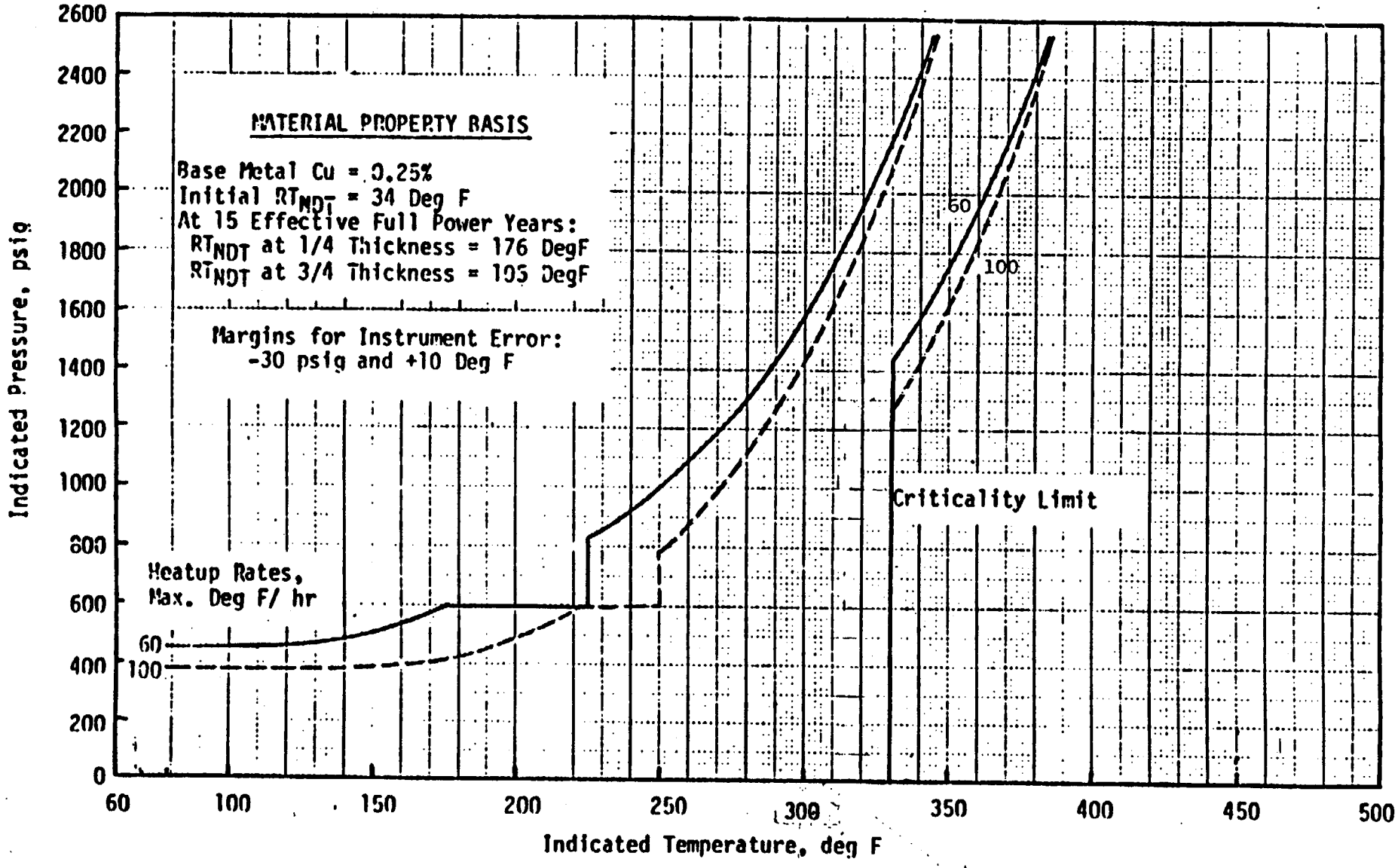
(1) Letter No. IPP-75-50, Westinghouse to Con Edison Dated May 16, 1975

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

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

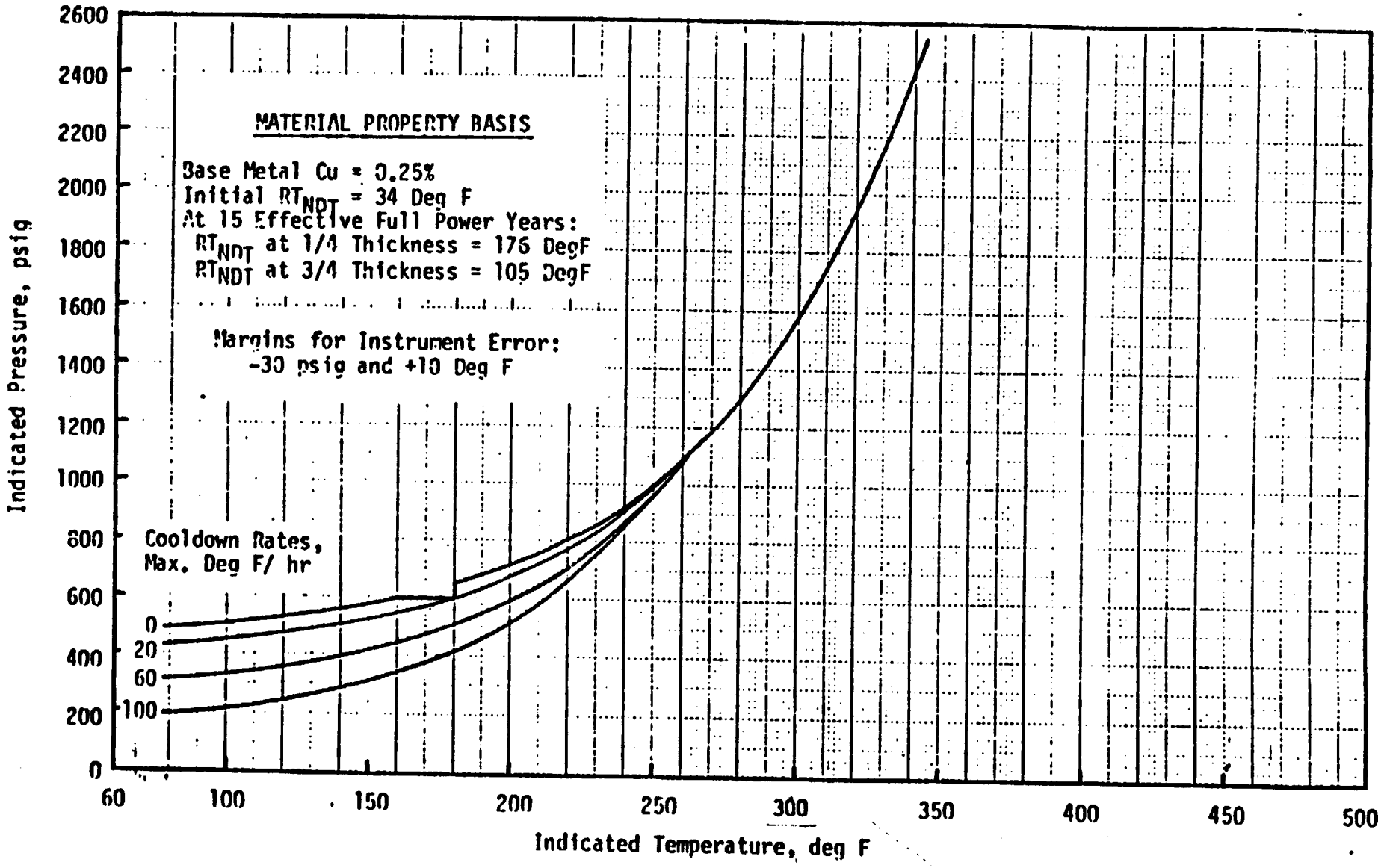
Notes:

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



INDIAN POINT UNIT NO. 2 COOLANT HEATUP LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

Figure 3.1.B-1



INDIAN POINT UNIT NO. 2 COOLANT COOLDOWN LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

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.

*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:

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

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

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

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

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

3.7×10^{10} dis/sec - Ci

$1.33 \times 10^{-11}\ rem/Mev/m^3$

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

If the air ejector discharge is diverted to the containment, the only activity released to atmosphere is that contained in the steam flow to the turbine gland seal (5000 lb/hr). For this case the activity release to 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 $\bar{E}$ will consist of the following:

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

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

3. A calculation of $\bar{E}$ by appropriate weighting of each 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. OPERATIONAL LEAKAGE LIMITS

a. Primary to Secondary Leakge:

(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. RCS/RHR Pressure Isolation Valves Leakage:

(1) Whenever the reactor is above cold shutdown, leakage through each of the RCS/RHR pressure isolation valves 897A, B, C & 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. Total Reactor Coolant System Leakage:

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

(a) No PRESSURE BOUNDARY LEAKAGE,

(b) 1 gpm UNIDENTIFIED LEAKAGE, and

(c) 10 gpm IDENTIFIED LEAKAGE.

(2) With any PRESSURE BOUNDARY LEAKAGE, the reactor must be placed in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

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

d. Leakage Into The Containment Free Volume:

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

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

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

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

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; 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} μc/cc to 10^{-6} μc/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

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 295°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.

[redacted]

C. Isolation Valve Seal Water System (IVSWS)

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

   a. The IVSWS shall be operable.

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

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

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

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

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

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

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

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

D. Weld Channel and Penetration Pressurizaton System (WC & PPS)

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

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

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

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

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

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

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

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

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°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. <u>Component Cooling System</u>

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. Control Room Air Filtration System

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

2. From the date that the control room air filtration system becomes and remains inoperable for any reason, operations requiring containment integrity are permissible only during the succeeding 3.5 days. At the end of this 3.5 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. Cable Tunnel Ventilation Fans

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

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

Basis

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

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

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. 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 295°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

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.

3.5 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.

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]

Instrument Operating Conditions

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

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The nuclear 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.

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	≤ 2.0 psig
2.	High Containment Pressure (Hi-Hi level)	a. Containment Spray b. Steam Line Isolation	≤ 30 psig
3.	Pressurizer Low Pressure	Safety Injection	≥ 1700 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	≤ 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	≤ 40% of full steam flow at zero load ≤ 40% of full steam flow at 20% load ≤ 110% of full steam at full load ≥ 540°F Tavg ≥ 600 psig steam line pressure
6.	Steam Generator Water Level (low-low)	Auxiliary Feedwater	≥ 5% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	≥ 40% nominal voltage
8a.	480v Emergency Bus Undervoltage (Loss of Voltage)	---	220V + 100V, -20V 3 sec ± 1 sec
8b.	480v Emergency Bus Undervoltage (Degraded Voltage)	---	403V ± 5V 180 sec ± 30 sec

Amendment No. 101

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 REDUN-DANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1.	Manual	2	1	1	0	Maintain hot shutdown
2.	Nuclear 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

Table 3.5-2 (2 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
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. 101

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 REDUN-DANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1	SAFETY INJECTION					
a.	Manual	2	1	1	0	Cold Shutdown
b.	High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
c.	High Differential Pressure Between steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
d.	Pressurizer Low Pressure*	3	2	2	1	Cold Shutdown
e.	High Steam Flow in 2/4 Steam Lines Coincident	2/line	1/2 in any 2 lines	1/line in each of 3 lines	2	Cold Shutdown
	With Low 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.

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 REDUN-DANCY	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 (De-graded 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	$T < 350^{o}F$

TABLE 3.5-3 (3 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 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 3.5-5

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED (1)	3 INDICATOR/ RECORDER (1)
1. Pressurizer Water Level	3	2	Indicator/One Channel is recorded
2. Reactor Coolant System Subcooling Margin Monitor (2)	1	1	Indicator
3. PORV Position Indicator (Limit Switch)	1/Valve	1/Valve	Indicator and alarm
4. PORV Block Valve Position Indicator (Limit Switch)	1/Valve (3)	1/Valve (3)	Indicator (3)
5. Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indicator
6. Auxiliary Feedwater Flow Rate	1/S.G.	1/S.G.	Indicator

Footnotes:

(1) Except as specified in another footnote, columns 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or recorder(s) to be inoperable for up to 14 days. If the minimum number of channels required is not restored to meet the above requirements within the time periods specified, then:

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

b. If the requirements of Columns 2 and 3 are not satisfied within an additional 48 hours, the reactor shall be cooled to below 350°F utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

TABLE 3.5-5 (CONTINUED)

Footnotes (Cont'd):

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

(3) Except at times when the valve operator is deenergized in accordance with technical specification 3.1.A.4.c.

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 Feedwater Pumps	N.A.	N.A.	R	
31.	Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.	
32.	PORV Position Indicator (Limit Switch)	M	R	R	
33.	PORV Block Valve Position Indicator (Limit Switch)	M*	R	R	
34.	Safety Valve Position Indicator (Acoustic Monitor)	M	R	R	
35.	Auxiliary Feedwater Flow Rate	M	R	R	
36.	PORV Actuation/ Reclosure Setpoints	N.A.	R	N.A.	

*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	**	

** 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.

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°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

4.9 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 NRC Regional Administrator of the Region I Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate test since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

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

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

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C).

e. Radioactive effluents (Specification 3.9).

f. Radiological environment monitoring (Specification 4.11)

g. Meteorological monitoring instrumentation (Specification 3.15).

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

DECAY HEAT REMOVAL AND ROD BANK WITHDRAWAL

Introduction

A number of events have occurred at operating Pressurized Water Reactor (PWR) facilities where decay heat removal capability has been seriously degraded due to inadequate administrative controls during shutdown modes of operation. One of these events, described in IE Information Notice 80-20 occurred at the Davis-Besse Station, Unit No. 1, on April 19, 1980. In IE Bulletin 80-22 dated May 9, 1980, licensees were requested to immediately implement administrative controls which would ensure that proper means are available to provide redundant methods of decay heat removal. While the function of the bulletin was to effect immediate action with regard to this problem, the NRC considered it necessary that an amendment be made to each PWR license to provide for permanent long-term assurance that redundancy in decay heat removal capability will be maintained. By the letter dated June 11, 1980, all PWR licensees were requested to propose Technical Specification changes that provide for redundancy in decay heat removal capability in all modes of operation; to use the NRC model technical specifications to provide an acceptable solution to the concern; to include an appropriate safety analysis as a basis; and to submit the proposed Technical Specification changes along with the bases by October 11, 1980.

Consolidated Edison Company of the New York (Consolidated Edison) responded to the NRC request for amended Technical Specifications for the Indian Point Nuclear Generating Unit 2, with their letter dated February 14, 1983, and proposed revisions to the Technical Specifications for the plant. In a letter dated September 14, 1984, Consolidated Edison responded to NRC staff's request for additional information and indicated that the existing and previously proposed Technical Specifications, supplemented by the surveillance requirements contained in the operational procedures presently available, satisfied the NRC requirements of assuring redundant decay heat removal for all modes of reactor operation.

By letter dated February 28, 1985, Consolidated Edison submitted a supplemental application revising the Technical Specifications to change the number of reactor coolant pumps required to be operating while control

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rods were being withdrawn in operating Mode 3. This revision is being made to be consistent with Consolidated Edison's reanalysis of the Uncontrolled Rod Cluster Assembly (RCCA) bank withdrawal event initiated from a just critical hot zero power condition.

Discussion and Evaluation

The proposed technical specifications provide for redundant means of decay heat removal in all modes of operation except during refueling when a large mass of water is above the core. These redundant means are outlined as follows:

Power Operation and Startup

All reactor coolant loops must be in operation with all four reactor coolant pumps operating. However, operation with three reactor coolant pumps is permitted if the nuclear overpower trip setpoint is appropriately reduced.

Hot Standby

All reactor coolant loops are required to be operable. However, only one of the loops is required to be in operation.

Hot Shutdown

At least two of the heat removal coolant loops are required to be operable, (i.e., either two RC loops or two DHR loops, or a combination of one each of these coolant loops). Two of the loops must be in operation.

Cold Shutdown and Refueling with the Water Level Above the Core Less Than 23 Feet

Two decay heat removal loops are required to be operable with at least one loop in operation.

Refueling with the Water Level Above the Core Greater Than 23 Feet

At least one decay heat removal loop is required to be in operation. The other loop need not be operable.

In addition to the above requirements for operability, the Technical Specification revisions specify surveillance intervals for heat removal systems that are consistent with the Standard Technical Specifications. The revised Technical Specifications provide an improvement over the existing ones since redundant decay heat removal will now be provided in MODES 3, 4, and 5. In MODE 6, with a large mass of water above the core, only a single heat removal path is required. The surveillance requirements that would identify any inoperable equipment or degrading performance are performed during each shift. The staff therefore concludes that the proposed Technical Specifications meet the intent of the Standard Technical Specifications with respect to redundant means of decay heat removal capability.

Additionally, Consolidated Edison has reanalyzed the uncontrolled RCCA event initiated from a just critical hot zero power condition with the assumption of two reactor coolant pumps in operation. The staff has examined the methods used to determine the peak fuel rod response and the input to that analysis, such as reactivity insertion rate, moderator on fuel temperature feedback effects, initial power distributions and reactor trip assumptions. The results of the analysis show that the event is terminated by the power range high neutron flux trip low setting of 35% of nominal power. The minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the 1.30 limit at all times during the transient, thus assuring that the core and the reactor coolant systems are not adversely affected. Since the fuel rods do not experience DNB, there is no clad damage and, therefore, there are not radiological consequences associated with the event. Therefore, the staff concludes that the proposed Technical Specifications are acceptable.

OVERPRESSURE PROTECTION

Introduction

By letter dated August 13, 1976, the NRC requested Consolidated Edison to propose system modifications to provide overpressure protection for the reactor vessel. As discussed in our June 22, 1984 Safety Evaluation, Consolidated Edison has installed an acceptable overpressure protection system (OPS). By letter dated February 14, 1983, Consolidated Edison proposed Technical Specification changes to incorporate limiting conditions of operation and surveillance requirements associated with the overpressure protection system by the addition of new specifications that define the low temperature reactor coolant overpressure protection range, incorporate a limit on the maximum primary-to-secondary differential temperature that is permitted prior to starting a reactor coolant pump and incorporates new requirements on the operability of Power Operated Relief Valves (PORV's). Consolidated Edison also proposed changes to limit the maximum number of energized safety injection and/or charging pumps for various pressurizer pressure, pressurizer level, and temperature conditions in the reactor coolant system.

By letter dated February 28, 1985 Consolidated Edison submitted a supplement to the February 14, 1983 submittal which revised the proposed overpressure protection Technical Specifications. The revisions were required due to changes in the pressure-temperature limitations for heatup and cooldown which were also proposed in the February 28, 1985 submittal and were issued by Amendment No. 96. By letters dated August 1, 1985 and September 13, 1985 Consolidated Edison provided additional page changes which were inadvertently omitted from the February 28 submittal. These changes were administrative in nature and in no way changed the meaning or technical content of the previous submittals.

Consolidated Edison's February 14, 1983 application for amendment also proposed editorial and format changes to the Technical Specifications.

Discussion and Evaluation

By letter dated February 14, 1983, Consolidated Edison submitted proposed Technical Specifications for the OPS. The Technical Specification states that the OPS is required to be operable when the RCS temperature is less than or equal to 295°F, and it gives conditions on the operation and operability of the system. The proposed basis for this Technical Specification is as follows:

> The Overpressure Protection System is designed to relieve the RCS pressure for certain unlikely overpressure transients to prevent these incidents from causing a peak RCS pressure exceeding 10 CFR 50, Appendix G limits. When the OPS is "armed", MOVs 535 and 536 are in the open position, and the PORVs will open upon receipt of the appropriate signal. This OPS arming can be accomplished either automatically by the OPS when the RCS is below a prescribed temperature or manually by the operator.
>
> The OPS will be set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G limits for the following events:
>
> 1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature 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. Inadvertent activation of the pressurizer heaters.
>
> Consideration of the above events provides bounding PORV setpoints for other potential overpressure conditions caused by heat or mass additions at low temperature.
>
> The RCS is protected against overpressure transients when RCS temperature is less than or equal to 295°F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the 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 limits for OPS actuation.

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

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

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

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

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

The proposed technical specifications also contain surveillance requirements on the OPS. Specifically system testing requirements are included in Table 4.1-1 (Item 37) of the February 14 submittal. Valve operability and cycling tests for PORV's and MOV's associated with the OPS

are performed in accordance with the requirements of 10 CFR 50.55a. This is identified in the basis for Specification 4.18. The staff concludes that with the inclusion of the Technical Specification revisions the Indian Point 2 overpressure protection system meets GDC 15 and 31 and the Consolidated Edison has implemented the guidelines of NUREG-0224. Therefore, the Technical Specification changes are acceptable.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 23, 1985

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