SITE MANAGEMENT MANUAL CONTROLLED DOCUMENT TRANSMITTAL FORM - PROCEDURES

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TRANSMITTAL FORM - PROCEDURES

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Document Name: ITS/BASES/TRM

CC_NAME	NAME	DEPT	LOCATION
1	OPS PROCEDURE GROUP SUPV.	OPS PROCEDURE GROUP	IP2
3	PLANT MANAGER'S OFFICE	UNIT 3(UNIT 3/IPEC ONLY)	IP2
5	CONTROL ROOM & MASTER	OPS(3PT-D001/6(U3/IPEC)	IP3(ONLY)
11	RES DEPARTMENT MANAGER	RES (UNIT 3/IPEC ONLY)	45-4-A
19	STEWART ANN	LICENSING	GSB-2D
20	CHEMISTRY SUPERVISOR	CHEMISTRY DEPARTMENT	45-4-A
21	TSC(IP3)	EEC BUILDING	IP2
22	SHIFT MGR.(LUB-001-GEN)	OPS (UNIT 3/IPEC ONLY)	IP3
23	LIS	LICENSING & INFO SERV	OFFSITE
25	SIMULATOR	TRAIN(UNIT 3/IPEC ONLY)	48-2-A
28	RESIDENT INSPECTOR	US NRC 88' ELEVATION	IP2
32	EOF	E-PLAN (ALL EP'S)	EOF
47	CHAPMAN N	BECHTEL	OFFSITE
50	TADEMY L. SHARON	WESTINGHOUSE ELECTRIC	OFFSITE
55	GSB TECHNICAL LIBRARY	A MCCALLION/IPEC & IP3	GSB-3B
61	SIMULATOR	TRAIN(UNIT 3/IPEC ONLY)	48-2-A
69	CONROY PAT	LICENSING/ROOM 205	GSB-2D
99	BARANSKI J (ALL)	ST. EMERG. MGMT. OFFICE	OFFSITE
106	SIMULATOR INSTRUCT AREA	TRG/3PT-D001-D006 ONLY)	#48
164	CONTROL ROOM & MASTER	OPS(3PT-D001/6(U3/IPEC)	IP3(ONLY)
207	TROY M	PROCUREMENT ENG.	1A
273	FAISON CHARLENE	NUCLEAR LICENSING	WPO-12
319	L.GRANT (LRQ-OPS TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
354	L.GRANT(LRQ-OPS/TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
357	L.GRANT(ITS/INFO ONLY)	TRAINING - ILO CLASSES	48-2-A
424	GRANT LEAH (9 COPIES)	(UNIT 3/IPEC ONLY)	#48
474	OUELLETTE P	ENG., PLAN & MGMT INC	OFFSITE
483	SCHMITT RICHIE	MAINTENANCE ENG/SUPV	45-1-A
484	HANSLER ROBERT	REACTOR ENGINEERING	72'UNIT 2
489	CLOUGHNESSY PAT	PLANT SUPPORT TEAM	GSB-3B
491	ORLANDO TOM (MANAGER)	PROGRAMS/COMPONENTS ENG	45-3-G
492	FSS UNIT 3	OPERATIONS	K-IP-I210
493	OPERATIONS FIN TEAM	33 TURBIN DECK	45-1-A
494	AEOF/A.GROSJEAN(ALL EP'S)	E-PLAN (EOP'S ONLY)	WPO-12D
495	JOINT NEWS CENTER	EMER PLN (ALL EP'S)	EOF
496	L.GRANT(LRQ-OPS/TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
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518	DOCUMENT CONTROL DESK	NRC (ALL EP'S)	OFFSITE
527	MILIANO PATRICK	NRC/SR. PROJECT MANAGER	OFFSITE
529	FIELDS DEBBIE	OUTAGE PLANNING	IP3/OSB

<u>Distribution of IP3 Technical Specification Amendment 224 and 225</u>
(Approved by NRC March 22, 2005 for Amend 224 and March 24 for Amend 225)

Pages are to be inserted into your controlled copy of the IP3 Technical Specifications following the instructions listed below. The **TAB** notation indicates which section the pages are located.

REMOVE PAGES	INSERT PAGES
TAB - Facility Operating License	
Remove all 7 pages	Insert 8 new pages
TAB - List of Effective Pages	
Pages 1 through 3, (Amendment 223)	Pages 1 through 3, (Amendment 225)
TAB - List of Amendments	
Page 13	Page 13
TAB 1.0 – Use and Application	
Page 1.1-3 (Amendment 205)	Page 1.1-3 (Amendment 224)
Page 1.1-5 (Amendment 213)	Page 1.1-5 (Amendment 225)
TAB 2.0 – Safety Limits	
Page 2.0-1 (Amendment 205)	Page 2.0-1 (Amendment 225)
Page 2.0-1 (Amendment 213)	N/A ; page is deleted
TAB 3.3 - Instrumentation	
Page 3.3.1-13 (Amendment 205)	Page 3.3.1-13 (Amendment 225)
Page 3.3.1-15 (Amendment 205)	Page 3.3.1-15 (Amendment 225)
Page 3.3.1-19 (Amendment 205)	Page 3.3.1-19 (Amendment 225)
Page 3.3.1-20 (Amendment 205)	Page 3.3.1-20 (Amendment 225)
Page 3.3.2-8 (Amendment 213)	Page 3.3.2-8 (Amendment 225)
Page 3.3.2-11 (Amendment 213)	Page 3.3.2-11 (Amendment 225)
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TAB 3.4 – Reactor Coolant System

Page 3.4.1-1 (Amendment 205)	Page 3.4.1-1 (Amendment 225)
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TAB 3.7 – Plant Systems

Page 3.7.1-3 (Amendment 213)	Page 3.7.1-3 (Amendment 225)
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TAB 5.0 – Administrative Controls

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ENTERGY NUCLEAR INDIAN POINT 3, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDED FACILITY OPERATING LICENSE

Amendment No. 203
License No. DPR-64

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

   A. The application for amendment by the Power Authority of the State of New York (PASNY) and Entergy Nuclear Indian Point 3, LLC (ENIP3) and Entergy Nuclear Operations, Inc. (ENO), submitted under cover letters dated May 11 and May 12, 2000, as supplemented on June 13, June 16, July 14, September 21, October 26, and November 3, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I; Amdt. 203 11/27/00

   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. ENIP3 and ENO are financially and technically qualified to engage in the activities authorized by this amendment; Amdt. 203 11/27/00

   E. ENIP3 and ENO have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; Amdt. 203 11/27/00

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

G. The receipt, possession and use of source, byproduct and special nuclear material as authorized by this amendment will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70 including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31; and

H. 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, Facility Operating License No. DPR-64 (previously issued to Consolidated Edison Company of New York, Inc., and the Power Authority of the State of New York) is hereby amended in its entirety and transferred to ENIP3 and ENO on November 21, 2000, to read as follows: Amdt. 203 11/27/00

A. This amended license applies to the Indian Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by ENIP3 and operated by ENO. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "Final Facility Description and Safety Analysis Report" as supplemented and amended, and the Environmental Report, as amended. Amdt. 203 11/27/00

B. Subject to the conditions and requirements incorporated herein, the Commission licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENIP3 to possess and use, and (b) ENO to possess, use and operate, the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this amended license; Amdt. 203 11/27/00

(2) ENO pursuant to the Act and 10 CFR Part 70, to receive, possess, and use, at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Facility Description and Safety Analysis Report, as supplemented and amended; Amdt. 203 11/27/00

(3) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, at any time, any byproduct source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; Amdt. 203 11/27/00

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 203 11/27/00

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 203 11/27/00

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

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 225 are hereby incorporated in the License. ENO shall operate the facility in accordance with the Technical Specifications.

(3) (DELETED) Amdt. 205 2-27-01

(4) (DELETED) Amdt. 205 2-27-01

D. (DELETED) Amdt.46 2-16-83

E. (DELETED) Amdt.37 5-14-81

F. This amended license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter of May 2, 1975, to Consolidated Edison Company of New York, Inc., granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

G. ENO shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and to the authority of 10 CFR 50.90 and CFR 50.54(p). The combined set of plans[1] for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0," and was submitted by letter dated October 14, 2004. — Letter of 10-28-04

H. ENO shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for Indian Point Nuclear Generating Unit No. 3 and as approved in NRC fire protection safety evaluations (SEs) dated September 21, 1973, March 6. 1979, May 2, 1980, November 18, 1982, December 30, 1982, February 2, 1984, April 16, 1984, January 7, 1987, September 9, 1988, October 21, 1991, April 20, 1994, January 5, 1995, and supplements thereto, subject to the following provision:

ENO may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

I.	(DELETED)	Amdt. 205 2/27/01
J.	(DELETED)	Amdt. 205 2/27/01
K.	(DELETED)	Amdt.49 5-25-84
L.	(DELETED)	Amdt. 205 2/27/01
M.	(DELETED)	Amdt. 205 2/27/01
14.	(DELETED)	Amdt. 49 5-25-84

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

O.	Evaluation, status and schedule for completion of balance of plant modifications as outlined in letter dated February 12, 1983, shall be forwarded to the NRC by January 1, 1984.	Amdt. 47 5-27-83
P.	Entergy Nuclear IP3 and ENO shall take no action to cause Entergy Global Investments, Inc. or Entergy International Ltd. LLC, or their parent companies to void, cancel, or modify the $70 million contingency commitment to provide funding for the facility as represented in the application for approval of the transfer of the license from PASNY to ENIP3 and ENO, without the prior written consent of the Director, Office of Nuclear Reactor Regulation.	Amdt. 203 11/21/00
Q.	The decommissioning trust agreement shall provide that the use of assets in the decommissioning trust fund, in the first instance, shall be limited to the expenses related to decommissioning of the facility as defined by the NRC in its regulations and issuances, and as provided in this license and any amendments thereto.	Amdt. 203 11/27/00
R.	The decommissioning trust agreement shall provide that no contribution to the decommissioning trust fund that consists of property other than liquid assets shall be permitted.	Amdt. 203 11/27/00
S.	With respect to the decommissioning trust fund, investments in the securities or other obligations of PASNY, Entergy Corporation, ENIP3, Entergy Nuclear FitzPatrick, LLC, ENO, or affiliates thereof, or their successors or assigns, shall be prohibited. Except for investments that replicate the composition of market indices or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear plants is prohibited.	Amdt. 203 11/27/00
T.	The decommissioning trust agreement shall provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the	Amdt. 203 11/27/00

trustee has first given the NRC 30 days prior written notice of the payment. In addition, the trust agreement shall state that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the Director, Office of Nuclear Reactor Regulation.

(21) The decommissioning trust agreement shall provide that the trust agreement shall not be modified in any material respect without the prior written consent of the Director, Office of Nuclear Reactor Regulation. Amdt. 203 11/27/00

V. The decommissioning trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investment" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations. Amdt. 203 11/27/00

W. For purposes of ensuring public health and safety, ENIP3, upon the transfer of this license to it, shall provide decommissioning funding assurance for the facility by the prepayment or equivalent method, to be held in a decommissioning trust fund for the facility, of no less than the amount required under NRC regulations at 10 CFR 50.75. Any amount held in any decommissioning trust maintained by PASNY for the facility after the transfer of the facility license to ENIP3 may be credited towards the amount required under this paragraph. Amdt. 203 11/27/00

X. ENIP3 shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for the transfer of this license to ENIP3 and ENO and the requirements of the order approving the transfer, and consistent with the safety evaluation supporting such order. Amdt. 203 11/27/00

AA. The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications: Amdt. 205 2/27/01

1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications

from the CTS," and Table LA, "Removed Details and Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.

2. The following is a schedule for implementing surveillance requirements (SRs):

   For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

   For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after the date of implementation of this amendment.

   For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the date of implementation of this amendment.

   For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the date of implementation of this amendment.

AB. With the reactor critical, Entergy shall maintain the reactor coolant system cold leg at a temperature (T_{cold}) greater than or equal to 525 F. Entergy shall maintain a record of the cumulative time that the plant is operated with the reactor critical while T_{cold} is below 525 F. Upon determination by Entergy that the cumulative time of plant operation with the reactor critical while T_{cold} is below 525 F has exceeded one (1) year, Entergy must:
(a) within one (1) month, inform the NRC, in writing, and
(b) within six (6) months submit the results of an analysis of the impact of the operation with T_{cold} below 525 F on the pressurized thermal shock reference temperature (RT_{PTS}).

3. This amended license is effective at 12:01 a.m., November 21, 2000, and shall expire at midnight December 12, 2015.

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment: Changes to the Technical Specifications

Date of Issuance: March 8, 1978

INDIAN POINT 3

TECHNICAL SPECIFICATIONS - APPENDIX A

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The latest amendment reflected in this list is: **Amendment 225**

AMENDMENT	SUBJECT	LETTER DATE
217	Use of Best-Estimate Large-Break Loss of Coolant Accident analysis methodology (WCAP 12945)	05/06/2003
218	Revise City Water surveillance to reflect addition of (backflow preventer) valves	08/04/2003
219	Revise Ventilation Filter Testing Program to adopt ASTM D3803 charcoal filter testing requirements per GL 99-02.	10/30/2003
220	Extension of the RCS pressure/temperature limits and corresponding OPS limits from 16.17 to 20 EFPY.	12/03/2003
221	Extension of RCP flywheel inspection interval (from 10 years to 20 years) per TSTF 421.	07/02/2004
222	Inoperable accumulator time extended from 1 hour to 24 hours per TSFT-370.	08/18/2004
223	Extension of the allowed outage time to support the placement of the CRVS in an alternate configuration for tracer gas testing.	01/19/2005
224	Full-scope adoption of alternate source term for dose consequence analysis of DBAs.	03/22/2005
225	Stretch Power Uprate (4.85%) from 3067.4 MWt to 3216 MWt, and adoption of TSTF-339.	03/24/2005

1.1 Definitions (continued)

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that amount of I-131 (curies) that alone would produce the same committed effective dose equivalent (CEDE) dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used for this calculation shall be those listed in Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total noniodine activity in the coolant.

L_a

The maximum allowable primary containment leakage rate, L_a, shall be 0.1% of primary containment air weight per day at the calculated peak containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except for leakage into closed systems and reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

   (Leakage into closed systems is leakage that can be accounted for and contained by a

(continued)

1.1 Definitions

MODE (continued)	vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE-OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in FSAR Chapter 13, Initial Tests and Operations; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3216 MWt.

(continued)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Vessel inlet temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, 5, and in MODE 6 when the reactor vessel head is on, the RCS pressure shall be maintained $\leq$ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6, restore compliance within 5 minutes.

Table 3.3.1-1 (page 1 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	$4^{(j)}$	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11	≤111% RTP
b. Low	$1^{(b)}$,2	$4^{(j)}$	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤25% RTP
3. Intermediate Range Neutron Flux	$1^{(b)}, 2^{(c)}$	1	F	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	NA

(continued)

(a) With Rod Control System capable of rod withdrawal and one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(j) Only 3 channels required during Mode 2 Physics Tests, LCO 3.1.8

Table 3.3.1-1 (page 3 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Pressurizer Pressure					
a. Low	$1^{(e)}$	4	H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥1900 psig
b. High	1,2	3	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤2400 psig
8. Pressurizer Water Level - High	$1^{(e)}$	3	H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤97%
9. Reactor Coolant Flow - Low	$1^{(e)}$	3 per loop	H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥90%

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 7 of 8)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.8% of ΔT span :

$$\Delta T \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[T - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ [*]°F.

P is the measured pressurizer pressure, psig
P′ is the nominal RCS operating pressure, ≤ [*] psig

K_1 ≤ [*]　　K_2 ≥ [*]/°F　　K_3 ≥ [*]/psig
τ_1 ≥ [*] sec　　τ_2 ≤ [*] sec

$f_1(\Delta I)$ =	[*] {[*] + (q_t - q_b)}	when q_t - q_b ≤ - [*]% RTP
	0% of RTP	when -[*]% RTP < q_t - q_b ≤ [*]% RTP
	-[*] {(q_t - q_b) - [*]}	when q_t - q_b > [*]% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

The values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (page 8 of 8)
Reactor Protection System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.8% of ΔT span:

$$\Delta T \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s}{(1 + \tau_3 s)} T - K_6 (T - T'') - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} at RTP, ≤ [*]°F.

$K_4 \leq$ [*]

$K_5 \geq$ [*] /°F for increasing T_{avg}
[*] /°F for decreasing T_{avg}

$K_6 \geq$ [*] /°F when T > T″
[*] /°F when T ≤ T″

$\tau_3 \leq$ [*] sec

$f_2(\Delta I)$ = [*]

*The values denoted with [*] are specified in the COLR.

Table 3.3.2-1 (page 1 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection					
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure-Hi	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤4.80 psig
d. Pressurizer Pressure-Low	1,2,3[(b)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥1710 psig
e. High Differential Pressure Between Steam Lines	1,2,3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	NA
f. High Steam Flow in Two Steam Lines	1,2[(d)],3[(d)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with T_{avg}- Low	1,2[(d)],3[(d)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥540.5°F

(continued)

(a) Not used

(b) Above the Pressurizer Pressure interlock.

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 120% full steam flow at 100% load, and corresponding to 120% full steam flow above 100% load. Time delay for SI ≤6 seconds.

(d) Except when all MSIVs are closed.

Table 3.3.2-1 (page 4 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	1,2(d),3(d)	2 per steam line	F	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(d),3(d)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure (Hi-Hi)	1,2(d), 3(d)	2 sets of 3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤24 psig
d. High Steam Flow in Two Steam Lines	1,2(d), 3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with T_{avg}-Low	1,2(d), 3(d)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥540.5°F
e. High Steam Flow in Two Steam Lines	1,2(d), 3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with Steam Line Pressure-Low	1,2(d), 3(d)	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥500 psig

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 120% full steam flow at 100% load, and corresponding to 120% full steam flow above 100% load. Time delay for SI ≤6 seconds.

(d) Except when all MSIVs are closed.

3.3 INSTRUMENTATION

3.3.7 Control Room Ventilation System (CRVS) Actuation Instrumentation

LCO 3.3.7 The CRVS actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1	Place CRVS in CRVS Mode 3.	7 days
B. One or more Functions with two channels or two trains inoperable.	B.1.1	Place CRVS in CRVS Mode 3.	72 hours
C. Required Action and associated Completion Time for Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 5.	36 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;

b. RCS average loop temperature is less than or equal to the limit specified in the COLR; and

c. RCS total flow rate ≥ 354,400 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

-----------------------------NOTE-----------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limits.	2 hours
B. Required action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average loop temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 354,400 gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE---------------- Not required to be performed until 24 hours after ≥ 90% RTP. --------------------------------------- Verify by precision heat balance that RCS total flow rate is ≥ 354,400 gpm and greater than or equal to the limit specified in the COLR.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 54.3% in MODES 1 and 2 or ≤ 90% in MODE 3; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 54.3% in MODES 1 and 2 OR ≤ 90% in MODE 3.	12 hours
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	24 months

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Neutron Flux Trip Setpoint in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE Neutron Flux Trip Setpoint (% RTP)
4	≤ 57
3	≤ 38
2	≤ 20

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each CRVS train for ≥ 15 minutes.	31 days
SR 3.7.11.2	Perform required CRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.11.3	Verify each CRVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.11.4	Verify one CRVS train can maintain a slight positive pressure relative to the adjacent enclosed area during CRVS Mode 3 operation at a makeup flow rate of ≥ 1500 cfm.	24 months on a STAGGERED TEST BASIS

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

This program provides controls for implementation of required testing of the ventilation filter function for the Control Room Ventilation System and Containment Fan Cooler Units.

Applicable tests described in Specifications 5.5.10.a, 5.5.10.b, 5.5.10.c and 5.5.10.d shall be performed:

1) After 720 hours of charcoal adsorber use since the last test; and,

2) Every 24 months for the Control Room Ventilation System, and Containment Fan Cooler Units; and,

3) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; and,

4) After any structural maintenance on the system housing that could alter system integrity; and,

5) After significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

SR 3.0.2 is applicable to the Ventilation Filter Testing Program.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each system that an inplace test of the high efficiency particulate air (HEPA) filters shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.

Ventilation System	Removal Efficiency	Flowrate (cfm)	Reference Standard
Control Room Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c
Containment Fan Cooler Units	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each system that an inplace test of the charcoal adsorber shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.

Ventilation System	Removal Efficiency	Flowrate (cfm)	Reference Standard
Control Room Ventilation System	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d
Containment Fan Cooler Units	≥ 99%	80% to 120% of design accident rate	Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each system that a laboratory test of a sample of the charcoal adsorber shows the methyl iodide removal efficiency specified below when tested in accordance with ASTM D3803-1989, subject to clarification below, at a temperature of 86°F and a relative humidity of 95%.

Ventilation System	Methyl iodide removal efficiency (%):	ASTM D3803-1989 Clarification
Control Room Ventilation System	$\geq$ 95.5	78 ft/min face velocity
Containment Fan Cooler Units	$\geq$ 85	59 ft/min face velocity

Note: For the 1" beds, the Control Room Ventilation System methyl iodide removal efficiency is verified greater than or equal to 93% rather than 95.5% at a face velocity of 50 ft/min under the above requirements. This is done prior to fuel movement in Refuel Outage 12 and every 6 months after Refuel Outage 12 until the end of Refuel Outage 13 or the 2" beds are installed.

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each system that the pressure drop across the combined HEPA filters, the demisters and prefilters (if installed), and the charcoal adsorbers is less than the value specified below when tested at the flowrate specified below.

Ventilation System	Delta P (inches wg)	Flowrate (cfm):
Control Room Ventilation System	6	≥ 90% of design accident rate
Containment Fan Cooler Units	6	≥ 90% of design accident rate

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

cooler unit when pressurized at ≥ 1.1 Pa. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.

The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 42.0 psig. The containment design pressure is 47 psig.

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Specification 2.1, Safety Limits (SL);
2. Specification 3.1.1, Shutdown Margin;
3. Specification 3.1.3, Moderator Temperature Coefficient;
4. Specification 3.1.5, Shutdown Bank Insertion Limits;
5. Specification 3.1.6, Control Bank Insertion Limits;
6. Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));
7. Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor;
8. Specification 3.2.3, AXIAL FLUX DIFFERENCE (AFD);
9. Specification 3.3.1, Reactor Protection System Instrumentation;
10. Specification 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits; and
11. Specification 3.9.1, Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Specifications 3.1.5, Shutdown Bank Insertion Limits, 3.1.6, Control Bank Insertion Limits, and 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor);

2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES, TOPICAL REPORT," September 1974 (W Proprietary). (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control);

2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission,Section 4.3,Nuclear Design,July 1981. Branch

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

3a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant-Accident Analysis," March 1998 (Westinghouse Proprietary);

3b. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 (Specification 2.1, Safety Limits (SL)) and Specification 3.4.1, (RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits);

3c. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Specification 2.1, Safety Limits (SL));

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));

3e. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code; Safety Injection into the Broken Loop and Cosi Condensation Model," July 1997 (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3f. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z))); and

3g. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided for each reload cycle to the NRC.

5.6.6 NOT USED

(continued)