December 24, 1992

Docket No. 50-461

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Mr. Frank A. Spangenberg
Manager - Licensing and Safety
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, Illinois 61727

Dear Mr. Spangenberg:

SUBJECT: AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. NPF-62 (TAC NO. 83213)

The Commission has issued the enclosed Amendment No. 68 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment revises the Technical Specifications in response to your application dated February 7, 1992.

The amendment deletes from Technical Specifications (TSs), numerous component lists as recommended in NRC Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications," and the reactor vessel specimen withdrawal schedule as recommended in GL 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications." Proposed TSs and Bases changes regarding primary containment and drywell average air temperature are not included in this amendment and will be handled as a separate request as discussed in a conversation with your staff on November 2, 1992.

Technical specification pages B 3/4 4-4a, 3/4 3-21, and 6-6 are included to correct typographical errors contained in amendment nos. 65, 66, and 67 respectively.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By:
Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 68 to License No. NPF-62
2. Safety Evaluation
3. Corrected TS pages

300054

cc w/enclosures: See next page

*See previous concurrence

LA:PD32:DRPW	PM:PD32:DRPW	PM:PD32:DRPW	BC:OTSB	OGC-OWFN
PKreutzer	DVPickett	JDyer	CGrimes*	BMBordenick*
12/24/92	12/24/92	12/24/92	11/17/92	11/17/92

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PDR ADOCK 05000461
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 68
License No. NPF-62

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

   A. The application for amendment by Illinois Power Company* (IP), and Soyland Power Cooperative, Inc. (the licensees) dated February 7, 1992, 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. NPF-62 is hereby amended to read as follows:

*Illinois Power Company is authorized to act as agent for Soyland Power Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

9212300275 921224
PDR ADOCK 05000461
P PDR

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 68 , are hereby incorporated into this license. Illinois Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: December 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 68

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert	Remove	Insert
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iii	iii	3/4 8-26	----
iv	iv	through 44	----
viii	viii	3/4 8-45	3/4 8-26
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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

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

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY, including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM RESPONSE TIME

1.7 The CONTAINMENT AND REACTOR VESSEL ISOLATION AND CONTROL SYSTEM (CRVICS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

CORE ALTERATION

1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Movement, including undervessel replacement, of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the Clinton-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these operating limits is addressed in individual Specifications.

CRITICAL POWER RATIO

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of an approved General Electric Critical Power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.12 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:
   1. Capable of being closed by an OPERABLE drywell automatic isolation system or
   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.
b. The drywell equipment hatch is closed and sealed.
c. The drywell airlock is OPERABLE pursuant to Specification 3.6.2.3.

MEMBER(S) OF THE PUBLIC (Continued)

persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.25 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.26 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

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

OPERATIONAL CONDITION - CONDITION

1.28 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.29 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as 1) described in Chapter 14 of the FSAR, 2) as authorized under the provisions of 10 CFR 50.59, or 3) as otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.30 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.31 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

   1. Capable of being closed by an OPERABLE containment automatic isolation system or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Specification 3.6.4.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

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

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each primary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

1.32 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71 and Federal and State regulations, burial ground requirements and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

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

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2894 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

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

ROD DENSITY

1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted are equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper as applicable secured in its closed position, except as provided in Specification 3.6.6.2.

b. All secondary containment equipment hatches are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.6.3.

d. At least one door in each access to the secondary containment is closed, except for normal entry and exit.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.6.1.a.

SELF TEST SYSTEM

1.39 The SELF TEST SYSTEM (STS) shall be that automatic test system designed to continually monitor the solid state nuclear system protection system (NSPS) functional circuitry by injecting short-duration pulses into circuits and verifying proper circuit response to various input combinations. The SELF TEST

SELF TEST SYSTEM (Continued)

SYSTEM is designed to maintain surveillance over all NSPS cabinet circuitry essential to the Reactor Protection System, Emergency Core Cooling Systems, Reactor Core Isolation Cooling System, and the Nuclear Steam Supply Shutoff System on a continuous cyclic basis.

The SELF TEST SYSTEM may be used to perform various surveillance testing functions to satisfy technical specifications requirements for those components it is designed to monitor. The STS may be used to augment conventional testing methods to perform CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS, CHANNEL CALIBRATIONS, RESPONSE TIME TESTS AND LOGIC SYSTEM FUNCTIONAL TESTS provided that OPERABILITY of the STS has first been verified.

SHUTDOWN MARGIN

1.40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted, except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn, and the reactor is in the shutdown condition; cold, i.e., 68°F; and xenon-free.

SITE BOUNDARY

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

1.42 DELETED

1.43 DELETED

STAGGERED TEST BASIS

1.44 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

1.46 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two separate time intervals: a) time from initial movement of the main turbine stop valve or control valve until 80% of turbine bypass capacity is established, and b) the time from initial movement of the main turbine stop valve or control valve

TABLE 3.3.2-1

CRVICS INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (q)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY AND SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	B(b)(f)	2(a)	1, 2, 3 #	20 25
b. Reactor Vessel Water Level-Low Low, Level 2 (ECCS Div. I and II)	B	2(a)	1, 2, 3	29
c. Reactor Vessel Water Level-Low, Low, Level 2 (HPCS-NSPS Div. III and IV)	B	2(a)(m)	1, 2, 3	29
d. Drywell Pressure - High	L(b)(f)	2(a)	1, 2, 3	20
e. Drywell Pressure - High (ECCS Div. I and II)	L	2(a)	1, 2, 3	29
f. Drywell Pressure - High (HPCS-NSPS Div. III and IV)	L	2(a)(n)	1, 2, 3	29
g. Containment Building Fuel Transfer Pool Ventilation Plenum Radiation - High	Z(b)(f)	2(a)	##	25
h. Containment Building Exhaust Radiation - High				
1) Outboard (Div. I) Valve Isolation	M(b)(f)	2(a)(o)	1, 2, 3 #	29 25
2) Inboard (Div. II) Valve Isolation	M(b)(f)	2(a)(o)	1, 2, 3 #	29 25
i. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation - High	5(b)(f)	2(a)	1, 2, 3 #	29 25
j. Reactor Vessel Water Level-Low Low Low, Level 1	U	2(k)	1, 2, 3 #	29 25
k. Containment Pressure-High	P	1(k)(l)	1, 2, 3 #	29 25

TABLE 3.3.2-1 (Continued)

CRVICS INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (q)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
l. Main Steam Line Radiation - High	C	2(a)	1, 2, 3	23
m. Fuel Building Exhaust Radiation - High	(b)(f)(j)	2(a)	1, 2, 3 #	25 25
n. Manual Initiation	R(b)(f)	1	1, 2, 3 #	26 25

2. MAIN STEAM LINE ISOLATION †	ISOLATION SIGNAL	APPLICABLE TABLE NOTES	APPLICABLE OPERATIONAL CONDITIONS	ACTION
a. Reactor Vessel Water Level-Low Low Low, Level 1	U	NA	1, 2, 3	20
b. Main Steam Line Radiation - High	C	d	1, 2, 3	23
c. Main Steam Line Pressure - Low	H	NA	1	23
d. Main Steam Line Flow - High	D	NA	1, 2, 3	23
e. Condenser Vacuum - Low	J	NA	1, 2**, 3**	23
f. Main Steam Line Tunnel Temp. - High	E	NA	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temp. - High	F	NA	1, 2, 3	23
h. Main Steam Line Turbine Bldg. Temp. - High	G	p	1, 2, 3	23
i. Manual Initiation	R	NA	1, 2, 3	22

TABLE 3.3.2-1 (Continued)

CRVICS INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (g)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	1	1(k)	1, 2, 3	27
b. Δ Flow Timer	X	1(k)	1, 2, 3	27
c. Equipment Area Temp. - High				
1. Pump Rooms - A, B, C	N	1/room(k)	1, 2, 3	27
2. Heat Exchanger Rooms - East, West	N	1/room(k)	1, 2, 3	27
d. Equipment Area Δ Temp. - High				
1. Pump Rooms - A, B, C	2	1/room(k)	1, 2, 3	27
2. Heat Exchanger Rooms - East, West	2	1/room(k)	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	B	2(k)	1, 2, 3	29
			#	25
f. Main Steam Line Tunnel Ambient Temp. - High	E	1(k)	1, 2, 3	27
g. Main Steam Line Tunnel ΔTemp. - High	F	1(k)	1, 2, 3	27
h. SLCS Initiation(e)	X	1	1, 2, 5*	27
i. Manual Initiation(g)	R	1	1, 2, 3	26
			#	25
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	V	1(k)	1, 2, 3	27
b. RCIC Steam Line Flow - High Timer	X	1(k)	1, 2, 3	27
c. RCIC Steam Supply Pressure - Low	V(h)	1(k)	1, 2, 3	27
d. RCIC Turbine Exhaust Diaphragm Pressure - High	V	2(k)	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

CRVICS INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (q)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)				
e. RCIC Equipment Room Ambient Temp. - High	V	1(k)	1, 2, 3	27
f. RCIC Equipment Room Δ Temp. - High	V	1(k)	1, 2, 3	27
g. Main Steam Line Tunnel Ambient Temp. - High	E	1(k)	1, 2, 3	27
h. Main Steam Line Tunnel Δ Temp. - High	F	1(k)	1, 2, 3	27
i. Main Steam Line Tunnel Temperature Timer	X	1(k)	1, 2, 3	27
j. Drywell Pressure - High	L(h)	1(k)	1, 2, 3	27
k. Manual Initiation	R	1(i)	1, 2, 3	26
l. RHR/RCIC Steam Line Flow - High	V	1(k)	1, 2, 3	27
m. RHR Heat Exchanger A, B Ambient Temperature - High	T	1/room(k)	1, 2, 3	28
n. RHR Heat Exchanger A, B Δ Temp. - High	S	1/room(k)	1, 2, 3	28

TABLE 3.3.2-1 (Continued)

CRVICS INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (g)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITION	ACTION
5. RHR SYSTEM ISOLATION				
a. RHR Heat Exchanger A, B Ambient Temperature - High	T	1/room[k]	1, 2, 3	28
b. RHR Heat Exchanger A, B ΔTemp. - High	S	1/room[k]	1, 2, 3	28
c. Reactor Vessel Water Level - Low, Level 3	A	2[a]	1, 2, 3	28
d. Reactor Vessel Water Level - Low Low Low, Level 1	U	2[k]	1, 2, 3	28
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	X	2[a]	1, 2, 3	28
f. Drywell Pressure - High				
1) RHR Test Lines	L	2[k]	1, 2, 3	28
2) Fuel Pool Cooling	L	2[a]	1, 2, 3	28
g. Manual Initiation	R	1	1, 2, 3	26

TABLE 3.3.2-1 (Continued)
CRVICS INSTRUMENTATION

TABLE NOTATIONS

\# When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\## When handling irradiated fuel in the primary containment (building) and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

* With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** When any turbine stop valve is greater than 95% open or the reactor mode switch is in the run position.

† Main steam line isolation trip functions have 2-out-of-4 isolation logic except for the main steam line flow - high trip function which has 2-out-of-4 isolation logic for each main steam line.

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the standby gas treatment system.

(c) Deleted

(d) Also trips and isolates the mechanical vacuum pumps.

(e) Isolates RWCU valves 1G33-F001 and 1G33-F004 only.

(f) Also actuates secondary containment ventilation isolation dampers.

(g) Manual Switch closes RWCU system inboard isolation valves F001, F028, F053, F040 and outboard isolation valves F004, F039, F034 and F054.

(h) Vacuum breaker isolation valves require RCIC system steam supply pressure low coincident with drywell pressure high for isolation of vacuum breaker isolation valves.

(i) A single manual isolation switch isolates outboard steam supply line isolation valve (F064) and the RCIC pump suction from suppression pool valve (F031) only following a manual or automatic (Reactor Vessel Water Level 2) RCIC system initiation.

(j) Only actuates secondary containment ventilation isolation dampers.

(k) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the trip condition provided that the redundant trip system is OPERABLE and monitoring that parameter.

TABLE 3.3.2-1 (Continued)
CRVICS INSTRUMENTATION

TABLE NOTATIONS

(l) Not required to be OPERABLE when valves 1VR002A,B and 1VQ006A,B are sealed closed in accordance with Specification 3.6.4.

(m) Four reactor vessel water level trip channels are logically combined in a one-out-of-two-twice configuration. For the purposes of the associated ACTION, each one-out-of-two logic is defined as a separate trip system.

(n) Four drywell pressure trip channels are logically combined in a one-out-of-two-twice configuration. For the purposes of the associated ACTION, each one-out-of-two logic is defined as a separate trip system.

(o) One trip system is associated with the A and B monitors; the other trip system is associated with the C and D monitors.

(p) Each channel consists of five temperature modules and their associated sensors. A channel is OPERABLE if and only if five temperature modules and their associated sensors are OPERABLE.

(q) Isolation signal descriptions for the symbols appearing in the Isolation Signal column are provided below:

Symbol	Description
A	Reactor Vessel Water Level Low (Level 3)
B	Reactor Vessel Water Level Low (Level 2)
C	Main Steam Line Rad. - High and Inop
D	Main Steam Line High Flow
E	Main Steam Tunnel Temp. High
F	Main Steam Tunnel Differential Temp. High
G	Main Steam in Turbine Building Temp. High
H	Turbine Inlet Pressure Low
J	Condenser Vacuum Low
L	Drywell Pressure High
M	Containment Exhaust Duct High Rad.
N	RWCU High Temp.
P	Containment Pressure-High
R	CRVICS Manual Initiation Pushbuttons
S	RHR Heat Exchanger Rooms A, B High Differential Temp
T	RHR Heat Exchanger Rooms A, B High Temp.
U	Reactor Water Level Low (Level 1)
V	RCIC High Steam Line Space Temp.
	RCIC Low Steam Line Pressure
	RCIC High Steam Flow
	High Turbine Exhaust Pressure
	RCIC Area High Temp.
	RCIC Area High Differential Temp.
X	Permissively Interlocked with Other Equipment
Z	High Rad. in Containment Refueling Pool Exhaust Duct
1	RWCU Equipment High Differential Flow
2	RWCU Vent High Differential Temp.
5	Containment Purge Duct High Radiation

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls* shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more required remote shutdown system instrumentation channels inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more required remote shutdown system controls inoperable, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown system instrumentation channels shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

4.3.7.4.2 Each of the above required remote shutdown control switches and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

* The scope of these controls excludes those associated with the RHR steam condensing mode.

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1, 2, 3	80
2. Reactor Vessel Water Level	2	1	1, 2, 3	80
3. Suppression Pool Water Level	4	2	1, 2, 3	80
4. Suppression Pool Water Temperature	2/quadrant†	1/quadrant†	1, 2, 3	80
5. Drywell Pressure	2	1	1, 2, 3	80
6. Drywell Air Temperature	2	1	1, 2, 3	80
7. Drywell/Containment Hydrogen and Oxygen Concentration Analyzer and Monitor	2	1	1, 2, 3	83
8. Containment Pressure ##	2/division	1/division	1, 2, 3	80
9. Containment Temperature	2	1	1, 2, 3	80
10. Safety/Relief Valve Acoustic Monitor	1/valve***	1/valve***	1, 2, 3	80
11. Containment/Drywell High Range Gross Gamma Radiation Monitors	4**	2*	1, 2, 3	81
12. HVAC Stack High Range Radioactivity Monitor#	1	1	1, 2, 3	81
13. SGTS Exhaust High Range Radioactivity Monitor#	1	1	1, 2, 3	81
14. Primary Containment Isolation Valve Position Indication ††	2/valve###	1/valve###	1, 2, 3	82

TABLE NOTATIONS

* One each for containment and drywell.

** Two each for containment and drywell.

*** Thermocouples in the SRV discharge line can serve as backup to the acoustic tail pipe monitors indication should one channel of the position indication become inoperable.

\# High range noble gas monitors and iodine/particulate sampler.

For Divisions I and II only.

Not applicable if valve position indication is unavailable because the valve was deliberately deactivated, provided the valve is in the isolated position and administrative controls are in place to ensure that the control room operators can determine the valve's position, if needed. Valves closed in accordance with these conditions may be reopened on an intermittent basis under administrative controls.

† These instruments monitor suppression pool water temperature when pool water level is below instruments of Specification 3.6.3.1.

†† One channel consists of the open limit switch, and the other channel consists of the closed limit switch for each automatic isolation valve of Specification 3.6.4.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.3 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.4 The reactor vessel material specimens shall be removed and examined to determine changes in reactor pressure vessel material properties as a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to adjust the curves of Figure 3.4.6.1-1.

4.4.6.1.5 DELETED.

4.4.6.1.6 The reactor vessel flange and head flange temperature shall be verified to be ≥ 70°F when vessel head bolting studs are under full tension:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 90°F, at least once per 12 hours.
   2. ≤ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs except 10 percent of the bolting studs may be fully tensioned at ≥ 10°F but ≤ 70°F.

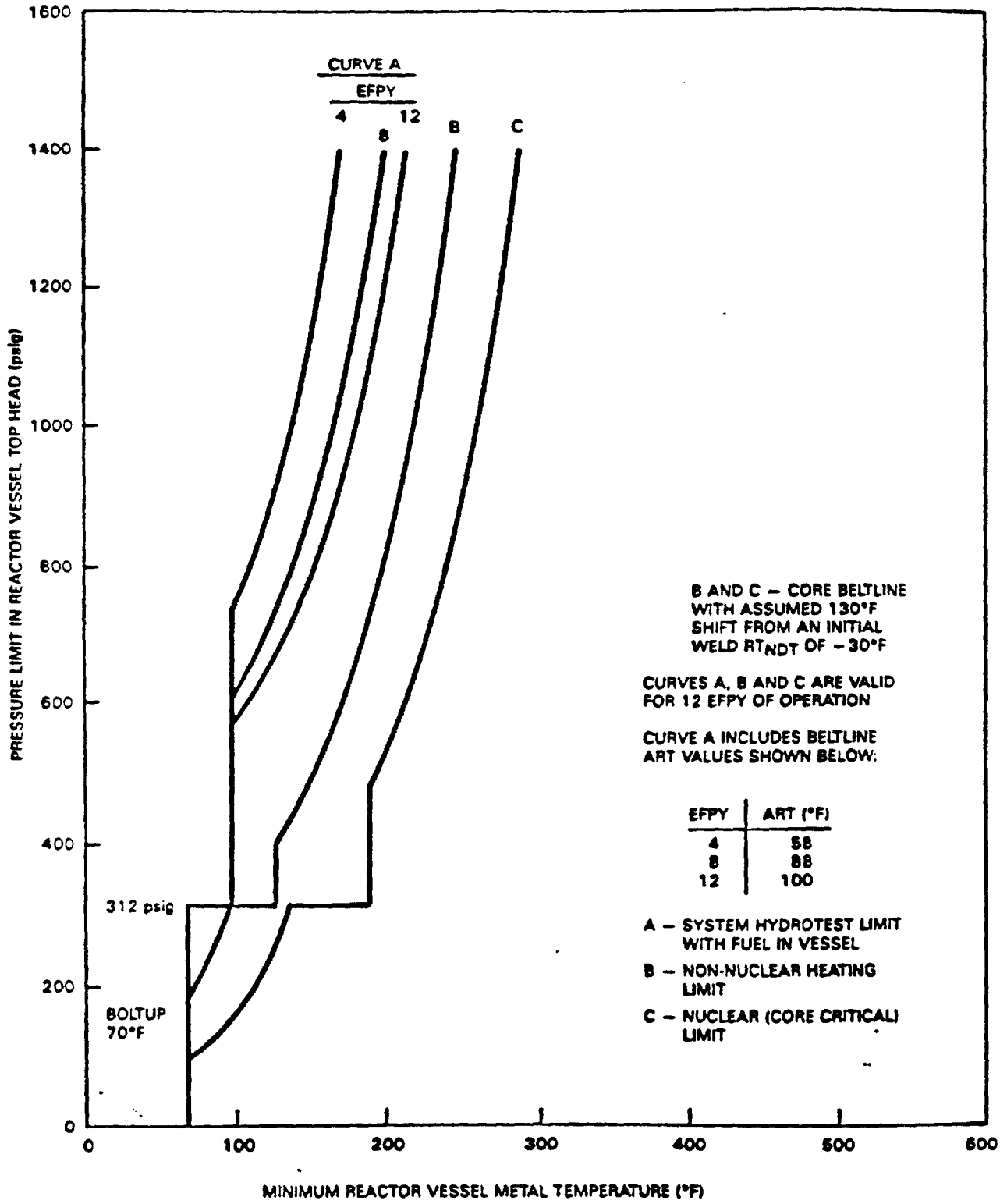

Figure 3.4.6.1-1. Reactor Vessel Pressure Versus Minimum Reactor Vessel Metal Temperature

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REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1045 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

*Not applicable during anticipated transients.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 9.0 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Specification 3.6.4.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1

**Except valves 1HG016 and 1HG017 and valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, steam tunnel, or drywell, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to:

   1. La, 0.65% by weight of the containment air per 24 hours at Pa, 9.0 psig.

b.# A combined leakage rate of less than or equal to 0.60 La, for all penetrations and all valves subject to Type B and C tests when pressurized to Pa, 9.0 psig.

c.* Less than or equal to 28 scf per hour for any one main steam line through through the isolation valves when tested at Pa, 9.0 psig.

d.## A combined leakage rate of less than or equal to 0.08 La, for all penetrations that are secondary containment bypass leakage paths when pressurized to Pa, 9.0 psig.

e. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 Pa, 9.9 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2**, and 3.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 La, or

b. The measured combined leakage rate for all penetrations and all valves subject to Type B and C tests exceeding 0.60 La, or

c. The measured leakage rate exceeding 28 scf per hour for any one main steam line through the isolation valves, or

d. The combined leakage rate for all penetrations which are secondary containment bypass leakage paths exceeding 0.08 La; or

e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

*Exemption to Appendix J of 10 CFR 50.
**See Special Test Exception 3.10.1.
#The leakage rates of valves 1B21-F032A and B are not required to be included until startup from the third refueling outage in accordance with an approved exemption to Appendix J of 10 CFR50.
##The leakage rates of valves 1B21-F032A and B are not required to be included until startup from the third refueling outage.

LIMITING CONDITION FOR OPERATION (Continued)

3.6.1.2 ACTION (Continued):

a. The overall integrated leakage rate(s) to less than or equal to 0.75 La, and

b. The combined leakage rate for all penetrations and all valves subject to Type B and C tests to less than or equal to 0.60 La, and

c. The leakage rate to less than 28 scf per hour for any one main steam line through the isolation valves, and

d. The combined leakage rate for all penetrations which are secondary containment bypass leakage paths to less than or equal to 0.08 La, and

e. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972 and BN-TOP-1 and verifying the result by the Mass Point Methodology described in ANSI/ANS N56.8-1981.

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at Pa, 9.0 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 La the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 La. The formula to be used is : $[Lo + Lam - 0.25\ La] \leq Lc \leq [Lo + Lam + 0.25\ La]$ where Lc = supplemental test result, Lo = superimposed leakage and Lam = measured Type A leakage.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.2 (Continued)

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the primary containment or bled from the primary containment during the supplemental test to be between 0.75 La and 1.25 La.

d. Type B and C tests shall be conducted*** with gas at Pa, 9.0 psig, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves,

3. Penetrations using continuous leakage monitoring systems,

4. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested with gas at Pa, 9.0 psig, at least once per 18 months.

g. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at Pa, 9.0 psig, at every other reactor shutdown for refueling, but in no case at intervals greater than 3 years.

h. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at 1.10 Pa, 9.9 psig, at least once per 18 months.

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.8.3.

j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.d, and 4.6.1.2.g.

***Except as provided in NRC-approved exemption to Appendix J to 10 CFR 50 for containment penetration 1MC-44.

3/4.6.4 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each primary containment isolation valve shall be OPERABLE#.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 for all primary containment isolation valves; and
OPERATIONAL CONDITIONS 1, 2, 3, and ** for those valves that isolate secondary containment bypass leakage paths; and
OPERATIONAL CONDITIONS 1, 2, 3, and 5*** for those valves that isolate the reactor water cleanup system suction containment penetration.

ACTION:

a. With one or more of the primary containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,*† or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*†

   The provisions of Specification 3.0.4 are not applicable provided the affected penetration is isolated in accordance with ACTION a.2 or a.3 above, and provided the associated system, if applicable, is declared inoperable or appropriate ACTION statements for that system are performed.

   Otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   Otherwise, in OPERATIONAL CONDITION **, suspend all operations involving CORE ALTERATIONS, handling irradiated fuel in the secondary containment, or with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

---

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

***With any control rod withdrawn. Not applicable to any control rod removed per Specification 3.9.10.1 or 3.9.10.2.

† Containment Isolation Valves can have dual functions in that they provide both containment isolation and Emergency Core Cooling functions. Any inoperable dual function valve could degrade the valve's other function.

Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

LIMITING CONDITION FOR OPERATION (Continued)

3.6.4 ACTION (Continued):

b. With one or more of the instrumentation line excess flow check primary containment isolation valves inoperable, operation may continue provided that within 4 hours either:

1. The inoperable valve is returned to OPERABLE status, or

2. The instrument line is isolated and the associated instrument is declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the isolation time.

4.6.4.2 Each automatic primary containment isolation valve shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on an isolation test signal each automatic primary containment isolation valve actuates to its isolation position.

4.6.4.3 The isolation time of each power operated or automatic primary containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.4.4 Each instrumentation line excess flow check primary containment isolation valve shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve actuates within the required differential pressure range.

3/4.6.5 DRYWELL POST-LOCA VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5 All drywell post-LOCA vacuum relief valves shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

a. With one drywell post-LOCA vacuum relief valve inoperable for opening but known to be closed, restore the inoperable vacuum relief valve to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one drywell post-LOCA vacuum relief valve open, restore the open vacuum relief valve to the closed position within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the position indicator of an OPERABLE drywell post-LOCA vacuum relief valve inoperable, verify the vacuum relief valve to be closed at least once per 24 hours by visual inspection. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 Each drywell post-LOCA vacuum relief valve shall be:

a. Verified closed at least once per 24 hours.

b. Demonstrated OPERABLE:

1. At least once per 31 days by:

   a) Cycling the vacuum relief valve through at least one complete cycle of full travel.

   b) Verifying the position indicator OPERABLE by observing expected valve movement during the cycling test.

2. At least once per 18 months by:

   a) Verifying the pressure differential required to open the vacuum relief valve, from the closed position, to be $\leq$ 0.2 psid, and

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.5 (Continued)

b) Verifying the position indicator OPERABLE by performance of a CHANNEL CALIBRATION.

3/4.6.6 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is greater than or equal to 0.25 inches of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All secondary containment equipment hatches are closed and sealed.

   2. At least one door in each access to the secondary containment is closed, except during normal entry and exit.

   3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position, except as provided in Specification 3.6.6.2.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.6.1 (Continued)

c. At least once per 18 months by verifying that:

1. Each standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge from the test start pressure in less than or equal to the time period corresponding to the measured flow rate specified in Figure 4.6.6.1-1.

2. By operating each standby gas treatment subsystem for one hour and maintaining $\geq$ 0.25 in. of vacuum water gauge in the secondary containment at a flow rate not exceeding 4000 cfm $\pm$ 10%.

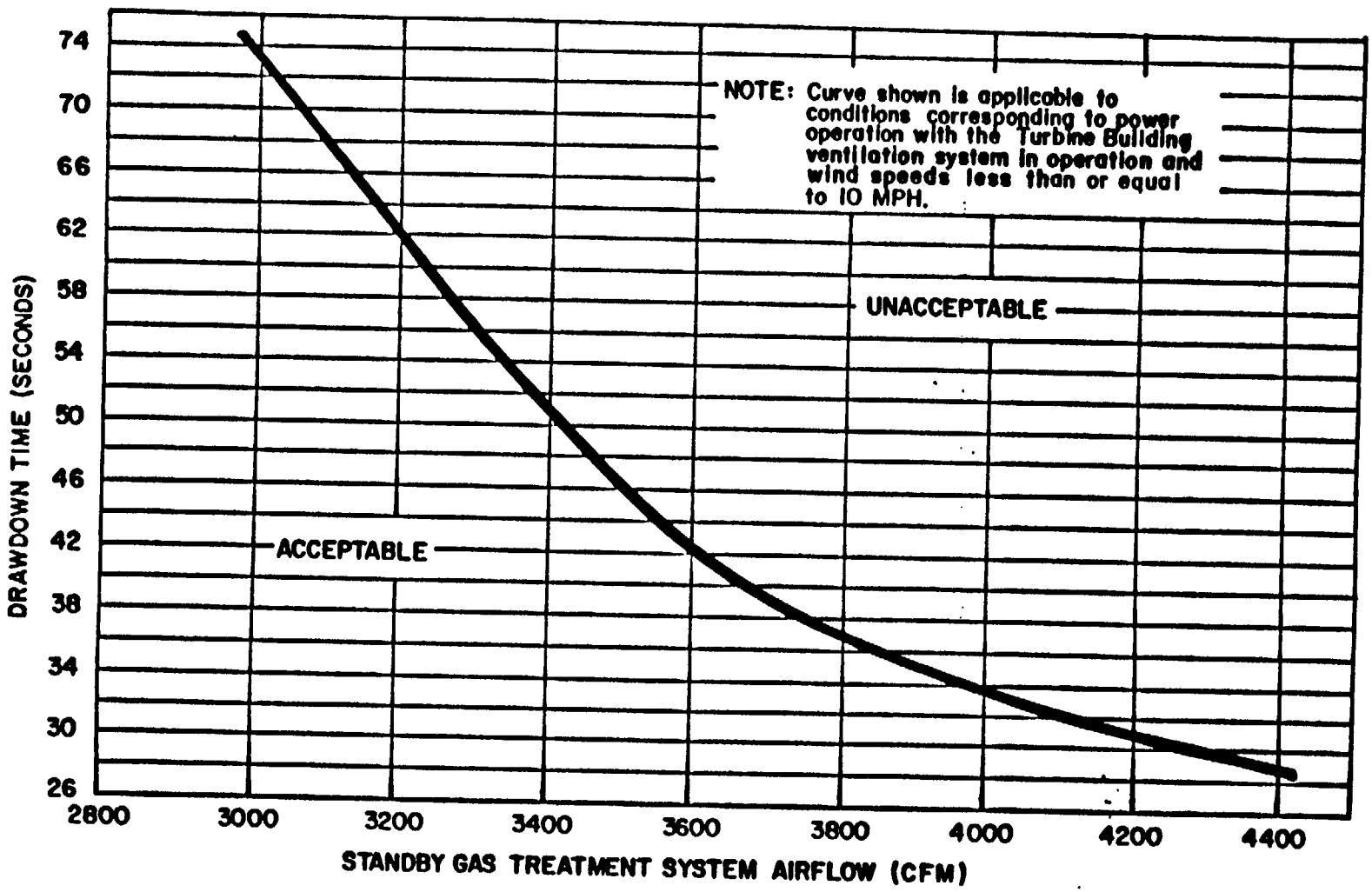

FIGURE 4.6.6.1-1
SECONDARY CONTAINMENT DRAWDOWN TIME
FOR 1500 CFM BOUNDARY LEAKAGE

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.6.2 Each secondary containment ventilation system automatic isolation damper shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open, and within 8 hours either:

a. Restore the inoperable damper to OPERABLE status, or

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

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

The provisions of Specification 3.0.4 are not applicable provided the affected penetration is isolated in accordance with ACTION b and/or c above, and provided the appropriate system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

Otherwise, in OPERATIONAL CONDITIONAL 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each secondary containment ventilation system automatic isolation damper shall be demonstrated OPERABLE:

a. Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifying the isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a secondary containment isolation test signal each secondary containment automatic isolation damper actuates to its isolation position.

c. At least once per 92 days by verifying the isolation time to be within its limit when tested.

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable in OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

4.6.6.3 (Continued)

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the system flow rate is 4000 cfm ± 10%.

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*, for a methyl iodide penetration of less than 0.175%; when tested in accordance with ASTM D3803-79 methods, with the following parameters:

   a) Bed Depth - 4 inches
   b) Velocity - 40 fpm
   c) Temperature - 80° C
   d) Relative Humidity - 70%

   and

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

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*, for a methyl iodide penetration of less than 0.175%; in accordance with ASTM D3803-79 methods, with the following parameters:

   a) Bed Depth - 4 inches
   b) Velocity - 40 fpm
   c) Temperature - 80° C
   d) Relative Humidity - 70%

*ANSI N510-1980 shall be used in place of ANSI N510-1975 as referenced in Regulatory Guide 1.52, Revision 2, March 1978.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.6.3 (Continued)

d. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

   a) LOCA, and
   b) Fuel handling accident.

2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.0 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm ± 10%.

3. Verifying that the filter train starts and isolation dampers open on receipt of the following test signals:

   a) Manual initiation from the control room, and
   b) Simulated automatic initiation signal.

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5. Verifying that the heaters dissipate at least 18.0 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

3/4.6.7 ATMOSPHERE CONTROL

CONTAINMENT HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.7.1 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one containment hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.1 Each containment hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the heater sheath temperature increases to greater than or equal to 600° F within 60 minutes and maintains greater than or equal to 600° F for at least 2 hours.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e, loose wiring or structural connections, deposits of foreign materials, etc.

   3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 2 hours following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

   4. Verifying during a recombiner system functional test that the reaction chamber temperature increase to be $\geq$ 1150° F within 2 hours and is maintained between 1177° F and 1223° F for at least 2 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.7.1 (Continued)

c. By measuring the leakage rate:

1. As a part of the integrated leakage rate test required by Specification 3.6.1.2, or

2. Of the system outside of the containment isolation valve at Pa, 9.0 psig, on the schedule required by Specification 4.6.1.2, and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

CONTAINMENT/DRYWELL HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.2 Two independent containment/drywell hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one containment/drywell mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.2 Each containment/drywell hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   1. Starting the system from the control room, and
   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least 800 scfm.

PRIMARY CONTAINMENT/DRYWELL HYDROGEN IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.3 The primary containment/drywell hydrogen ignition system, consisting of two independent primary containment/drywell hydrogen ignition subsystems each consisting of six circuits, shall be operable with no more than two igniter assemblies inoperable per circuit, no more than five igniter assemblies inoperable per subsystem, and no adjacent igniter assemblies inoperable.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one primary containment/drywell hydrogen ignition subsystem and/or circuit inoperable, restore the inoperable subsystem and/or circuit to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With any adjacent igniter assemblies inoperable, restore all igniter assemblies adjacent to an inoperable igniter assembly to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.3 The primary containment/drywell hydrogen ignition system shall be demonstrated OPERABLE.

a. At least once per 184 days by energizing all the igniter assemblies and performing current/voltage measurements of each circuit.

   1. If more than three igniter assemblies on either subsystem are deter-mined to be inoperable, Surveillance Requirement 4.6.7.3.a shall be performed at least once per 92 days until this condition no longer exists.

   2. If more than one igniter assembly on each subsystem are determined to be inoperable, determine if the inoperable igniter assemblies are adjacent.

b. At least once per 18 months, by energizing each igniter assembly, verifying a surface temperature of at least 1700°F for each of the accessible igniters and verifying by measurement sufficient current/voltage to develop 1700°F surface temperature for those igniter assemblies in inaccessible areas.

LIMITING CONDITION FOR OPERATION (Continued)

3.8.3.2 ACTION (Continued):

2. With Division III or Division IV of the above required AC distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

b. For DC power distribution:

1. With both Division I and Division II of the above required DC distribution system not energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

2. With Division III or Division IV of the above required DC distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the buses/MCCs.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 Primary and backup containment penetration conductor overcurrent protective devices associated with each primary containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetrations' design ratings.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the required containment penetration conductor overcurrent protective devices inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 6.9-kV circuit breakers, de-energize the 6.9-kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For lower voltage circuit breakers, remove the inoperable circuit breaker(s) from service by racking out the breaker within 72 hours and verify the inoperable breaker(s) to be racked out at least once per 7 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 6.9-kV circuits which have their redundant circuit breakers tripped or to lower voltage circuits which have the inoperable circuit breaker racked out.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the medium voltage 6.9-kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing:

      a) A CHANNEL CALIBRATION of the associated protective relays, and

4.8.4.1 (Continued)

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting currents in excess of the breaker's nominal setpoint and measuring the response time of the long time delay and short time delay trip elements and setpoint of the instantaneous element where applicable. The measured response time shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve in safety systems with a bypass device(s) integral with the motor starter shall be bypassed continuously for those directions for which the valve performs an active safety function.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously in the valve's safety direction(s), continuously bypass the affected thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously in the valves' safety direction(s):

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection for the above required valves shall be verified to be bypassed in the valves' safety direction(s) following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 One RPS electric power monitoring channel for each inservice RPS special solenoid power supply or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With the RPS special solenoid electric power monitoring channel for an inservice RPS special solenoid power supply or alternate power supply inoperable, restore the power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS special solenoid power supply or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS special solenoid electric power monitoring channels shall be determined OPERABLE:

a. At least once per six months by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, undervoltage and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

		EPA-INVERTER A	EPA-INVERTER B
1.	Overvoltage	≤ 134.2 + 0,-3 VAC	≤ 133.6 + 0, - 3 VAC
2.	Undervoltage	≥ 114.2-0, + 3 VAC	≥ 113.2 - 0, + 3 VAC
3.	Underfrequency	> 57 - 0 + 1.2 Hz	> 57 - 0 + 1.2 Hz

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM (Continued)

by comparing the detector(s) output with data obtained during the previous LPRM calibrations.

3/4.3.7.8 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection system ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Automatic transfer to the chlorine mode is not required when chlorine containers having a capacity of 150 pounds or less are stored 100 meters or more from the control room or its fresh air inlets. Upon detection of a high concentration of chlorine, the control room ventilation system will automatically be placed in the chlorine mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January, 1977.*

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

Deleted

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

*This specification is not applicable after all chlorine containers having a capacity of 100 pounds or greater are removed from the site including the chlorine containers located at the site sewage treatment plant.

3/4.3.7.11 MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING INSTRUMENTATION

The main condenser offgas treatment system explosive gas monitoring instrumentation is provided to monitor and control the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system.

The intent of the * note attached to the CHANNEL CALIBRATION requirement is to specify that the CHANNEL CALIBRATION is to be performed using at least two separate gas samples of different, specific hydrogen concentrations appropriate for the sensor range. The balance of the sample gas mixture (normally nitrogen) is not necessarily restricted purely to nitrogen but must be in accordance with the requirements or recommendations provided by the manufacturer of the explosive gas monitoring instrumentation.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

DELETED

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided to initiate action to mitigate the consequences of accidents that are beyond the ability of the operator to control. The LPCI mode of the RHR system is automatically initiated on a high drywell pressure signal and/or a low reactor water level, level 1 signal. The containment spray system will then actuate automatically following high drywell and high containment pressure signals. Negative barometric pressure

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam line rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis of compliance are more fully discussed in FSAR subsection 5.3.1.5 entitled "Fracture Toughness."

3/4.4.6 PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT} of the core beltline region. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A, B, and C, includes an assumed shift in RT_{NDT} for the conditions at 12 Effective Full Power Years. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used to predict reactor vessel material transition temperature shift. Flux wires which were removed after the first fuel cycle and will be removed at later intervals with the surveillance specimens are analyzed and provide an improved neutron fluence estimate for the reactor vessel. This data is then used to modify Bases Figure B 3/4.4.6-1 and predictions of reactor vessel material transition temperature shift per Regulatory Guide 1.99, Revision 2. The operating limit curves of Figure 3.4.6.1-1 have been and will be adjusted, as required, on the basis of the specimen data and the recommendations of Regulatory Guide 1.99, Revision 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C and A for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment; however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.3 DEPRESSURIZATION SYSTEMS (Continued)

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the system is to ensure containment integrity following a LOCA by preventing excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related component systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the standby service water is accomplished in the RHR heat exchangers.

The suppression pool make-up system provides water from the upper containment pool to the suppression pool by gravity flow through two 100% capacity dump lines following a LOCA. The quantity of water provided is sufficient to account for all conceivable post-accident entrapment volumes, ensuring the long term energy sink capabilities of the suppression pool and maintaining the water coverage over the uppermost drywell vents. The minimum freeboard distance above the suppression pool high water level to the top of the weir wall is adequate to preclude flooding of the drywell in the event of an inadvertent dump. During refueling, neither automatic nor manual action can open the make-up dump valves.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50 "and the requirements of NUREG-0660 as clarified by NUREG-0737 as described in the FSAR, Appendix D, item II.E.4.2 (Containment Isolation Dependability)." OPERABILITY of the automatic containment isolation valves which isolate the reactor water cleanup system suction containment penetration is also required in OPERATIONAL CONDITION 5 with any control rod withdrawn to ensure that, in the event initiation of the standby liquid control system becomes necessary, the sodium pentaborate solution is not removed from the reactor coolant system. The requirements for OPERABILITY of these valves agree with those conditions for which the standby liquid control system is required to be OPERABLE per Specification 3/4.1.5.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Measurement of the closure time of automatic containment isolation valves is performed for the purpose of demonstrating PRIMARY CONTAINMENT INTEGRITY and system OPERABILITY (Specification 3/4.6.1).

3/4.6.4 CONTAINMENT ISOLATION VALVES (Continued)

The Maximum Isolation Times (MIT) for primary containment automatic isolation valves listed in this specification are either the analytical times used in the accident analysis; described in the FSAR; or times derived by applying margins to the test data obtained by performing testing in accordance with the Inservice Testing program (IST) outlined in Section XI of the ASME Code. For non-analytical automatic primary containment isolation valves, the MIT is derived as follows:

1) Valves with full stroke times less than or equal to 10 seconds,
   MIT = Initial Base Line Time X 2

2) Valves with full stoke time greater than 10 seconds,
   MIT = Initial Base Line Time X 1.5

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES (Continued)

circuit breakers by periodic surveillance. The low-frequency motor generator set electrical power supply to the reactor recirculation pumps is provided with one overcurrent protection circuit breaker since the generator's maximum output under fault conditions is less than the penetration's design rating. The surveillance requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The bypassing of the motor-operated valves thermal overload protection continuously ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection continuously are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, March 1977.

The reactor protection system (RPS) electric power monitoring assemblies provide protection to the RPS and other systems which receive power from the RPS buses by acting to disconnect the RPS from the power source in the presence of an electrical fault in the power supply.

TABLE 3.3.2-2 (Continued)

CRVICS INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY AND SECONDARY CONTAINMENT ISOLATION (Continued)		
k. Containment Pressure - High	≤ 2.56 psid	≤ 3.00 psid
l. Main Steam Line Radiation - High	≤ 3.0 x full power background#	≤ 3.6 x full power background#
m. Fuel Building Exhaust Radiation - High	≤ 10 mR/hr	≤ 17 mR/hr
n. Manual Initiation	NA	NA
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -145.5 in.*	≥ -147.7 in.
b. Main Steam Line Radiation - High	≤ 3.0 x full power background#	≤ 3.6 x full power background#
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 170 psid	≤ 178 psid
e. Condenser Vacuum - Low	≥ 8.5 in. Hg vacuum	≥ 7.6 in. Hg vacuum
f. Main Steam Line Tunnel Temp. - High	≤ 156°F	≤ 171°F
g. Main Steam Line Tunnel Δ Temp. - High	≤ 54.5°F	≤ 63°F
h. Main Steam Line Turbine Bldg. Temp. - High		
(1) 1E31 - N559 A, B, C, D 1E31 - N560 A, B, C, D 1E31 - N561 A, B, C, D 1E31 - N562 A, B, C, D	≤ 131.2°F	≤ 142°F
(2) 1E31 - N563 A, B, C, D	≤ 139.5°F	≤ 150°F
i. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

CRVICS INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 59 gpm	≤ 66.1 gpm
b. Δ Flow Timer	≥ 45 sec.	≤ 47 sec.
c. Equipment Area Temp. - High		
1. Pump Rooms - A, B, C	≤ 186.5°F	≤ 202°F
2. Heat Exchanger Rooms - East, West	≤ 190°F	≤ 205°F
d. Equipment Area Δ Temp. - High		
1. Pump Rooms - A, B, C	≤ 54.5°F	≤ 63°F
2. Heat Exchanger Rooms - East, West	≤ 54.5°F	≤ 63°F
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -45.5 in.*	≥ -47.7 in.
f. Main Steam Line Tunnel Ambient Temp. - High	≤ 156°F	≤ 171°F
g. Main Steam Line Tunnel Δ Temp. - High	≤ 54.5°F	≤ 63°F
h. SLCS Initiation	NA	NA
i. Manual Initiation	NA	NA
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 110 in. H_2O	≤ 118.5 in H_2O
b. RCIC Steam Line Flow - High Timer	≥ 3 sec.	≤ 13 sec.
c. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 52 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel.

The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION SINGLE UNIT FACILITY

	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
POSITION	CONDITION 1, 2, or 3	CONDITION 4 or 5
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

TABLE NOTATIONS

SS - Shift Supervisor with a Senior Operator license on Unit 1.
SRO - Individual with a Senior Operator license on Unit 1.
RO - Individual with a Operator license on Unit 1.
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2, or 3, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Manager-Licensing and Safety, Director-Nuclear Safety, and to members of the Nuclear Review and Audit Group (NRAG).

COMPOSITION

6.2.3.2 The ISEG shall be composed of a least five, dedicated, full-time members located on site. Each ISEG member shall have either:

(1) A bachelor's degree in engineering or related science and at least three years of professional-level experience in his field which shall include at least one year of nuclear power experience, or

(2) At least five years of nuclear experience and hold or have held a Clinton Power Station Senior Reactor Operator license.

As a minimum, four of the ISEG members shall have the qualifications specified in (1) above.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification[1] that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Director-Nuclear Safety, to the Manager-Licensing and Safety and to members of the NRAG.

[1] Not responsible for signoff function.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. NPF-62

ILLINOIS POWER COMPANY, ET AL.

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter dated February 7, 1992, the Illinois Power Company (IP), (the licensee), requested an amendment to Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed amendment would remove the technical specification (TS) tables that include lists of components referenced in individual specifications and TS Table 4.4.6.1-1, "Reactor Vessel Material Surveillance Program-Withdrawal Schedule." Additionally, the TSs have been modified such that all references to these tables have been removed. Finally, the TS have been modified to state requirements in general terms that include the components listed in the tables removed from the TS. Guidance on the proposed TS changes was provided by Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991, and GL 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," dated January 4, 1991.

2.0 DISCUSSION

In both GL 91-01 and GL 91-08, the staff provided guidance to licensees for implementing line-item TS improvements.

10 CFR Part 50, The Code of Federal Regulations, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," requires licensees to establish a surveillance program to monitor changes in the fracture toughness properties of ferretic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. GL 91-01 indicates that the TS include limiting conditions for operation (LCOs) that establish pressure and temperature limits for the reactor coolant system (RCS). These LCOs are typically defined by TS figures that provide an acceptable range of operating temperatures and pressures for heatup, cooldown, criticality, and inservice leak and hydrostatic testing. These TS figures are based on the results of the surveillance of reactor vessel material required by Appendix H. The placement of the schedule for reactor vessel specimen removal in TS duplicates the controls on changes to this schedule that are established by II.B.3 of Appendix H and 10 CFR 50.4, "Standards for Licenses and Construction Permits." Therefore, the staff concluded in GL 91-01 that, because this duplication is unnecessary, the removal of this TS schedule as a line-item improvement is consistent with the Commission Policy Statement on TS improvements.

Similarly, GL 91-08 provides specific guidance on the removal of component lists from TSs when removal of the component lists do not alter existing TS requirements or those components to which they apply.

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3.0 EVALUATION

In accordance with GL 91-01, GL 91-08, and 10 CFR 50.90, the licensee proposed the following changes to the CPS TS. The licensee's proposed changes are discussed in the order the associated TS appears in the CPS TS. The staff's evaluation and conclusion follow each proposed change.

(1) Proposed changes to TS index pages to make editorial corrections associated with the proposed TS changes, to reflect the deletion of tables which contain component lists, or to reflect renumbering of pages due to deletion of large lists are included in the licensee's request.

The staff concludes that the proposed changes are acceptable based on the fact that they are administrative in nature only (reflecting the TS changes evaluated below).

(2) TS Definition changes to 1.12, "DRYWELL INTEGRITY," 1.31, "PRIMARY CONTAINMENT INTEGRITY," and 1.38, "SECONDARY CONTAINMENT INTEGRITY," are proposed to delete references to removed TS Tables and provide clarification of revised TSs.

The proposed changes to the TS Definitions are administrative or editorial in nature (reflecting the TS changes evaluated below) or provide clarification to existing TSs (no technical change to the current requirements), therefore, the staff finds the licensee's proposed changes acceptable.

(3) The licensee proposed to replace footnote "††" to TS Table 3.3.2-1, "CRVIS Instrumentation," with footnote (q). This footnote is modified to relocate the list of isolation signal descriptions to footnote (q) rather than refer to TS 3.6.4, Table 3.6.4-1 for the list of isolation signal descriptions. The list of valves actuated by the isolation signals is contained in CPS procedures. Additionally, the licensee proposed to modify footnote (f) to delete the reference to TS Table 3.6.6.2-1 and footnote (j) to delete the references to TS Table 3.6.6.2-1 and footnote "††".

The proposed changes to footnote "††", (f), and (j) are editorial in nature (reflecting the TS changes evaluated below); therefore, the staff finds the licensee's proposed changes acceptable.

(4) A proposed revision to TS 3.3.7.4, "Remote Shutdown Monitoring Instrumentation," to delete Table 3.3.7.4-1, "Remote Shutdown Monitoring Instrumentation"; Table 3.3.7.4-2, "Remote Shutdown System Controls"; and Table 4.3.7.4-1, "Remote Shutdown Monitoring Instrumentation Surveillance Requirements" was also included. Revisions to the LCO are proposed which include deletion of references to Tables 3.3.7.4-1 and 3.3.7.4-2 and to attach a footnote to the LCO to incorporate footnote "*" from Table 3.3.7.4-2. Footnote "*" states that this TS excludes those controls associated with the residual heat removal (RHR) system steam condensing mode. Additionally, the licensee proposed to revise the Action Statements to delete references to Tables 3.3.7.4-1, 3.3.7.4-2 and to add specific LCO criteria, and Surveillance Requirement 4.3.7.4.1 to delete the reference to Table 4.3.7.4-1 and to specify the actual surveillance intervals. Further, the licensee proposed to incorporate the word "required" into the Action Statement and Surveillance Requirement 4.3.7.4.2 to denote those remote shutdown panel controls which are required to be operable and to exclude those controls associated with the steam condensing mode of the RHR system.

With the removal of TS Tables 3.3.7.4-1, 3.3.7.4-2, and 4.3.7.4-1 for remote shutdown monitoring instrumentation, the operability and surveillance requirements have been stated in general terms that apply to all required remote shutdown monitoring instrumentation. The remote shutdown instrumentation requirements are consistent with the regulatory requirements for remote shutdown panel instrumentation operability. The licensee confirmed that the remote shutdown instrumentation and controls listed in TS Tables 3.3.7.4-1, 3.3.7.4-2, and 4.3.7.4-1 are contained in CPS procedures (CPS Procedure Nos. 9000.10, 9027.01, 9054.05). Additionally, the remote shutdown instrumentation and controls are described in Section 7.4.1.4 of the CPS Updated Safety Analysis Report (USAR).

The staff concludes that the proposed changes are consistent with regulatory requirements and guidance contained in GL 91-08. Therefore, the licensee's proposed rewording to generalize the operability and surveillance requirements, removal of the TS Tables associated with remote shutdown monitoring instrumentation and controls, and addition of the word "required" to TS 3/4 3.7.4 are acceptable.

(5) The licensee proposed an administrative change to footnote "††" to TS Table 3.3.7.5-1, "Accident Monitoring Instrumentation," which would replace a reference to TS Table 3.6.4-1 with a reference to TS 3.6.4.

The proposed change to footnote "††" is administrative in nature (reflecting the TS changes evaluated below); therefore, the staff finds the licensee's proposed change acceptable.

(6) Removal of TS Table 4.4.6.1-1, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," and the reference to it in Surveillance Requirement 4.4.6.1.4, and the corresponding TS Bases was also proposed by the licensee.

TS 3/4.4.6, "Pressure/Temperature Limits," contains a limiting condition for operation (LCO) for the Reactor Coolant System (RCS) that limits the rate of change in temperature and pressure to values consistent with the fracture toughness requirements of the American Society of Mechanical Engineers (ASME) Code and 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." Changes in the values of these limits are necessary because the fracture toughness properties of ferretic materials in the reactor vessel change as a function of the reactor operating time (neutron fluence).

The TS include a surveillance requirement, TS 4.4.6.1.4, to require the removal and examination of the irradiated specimens of reactor vessel material. The licensee examines the specimens to determine the changes in material properties in accordance with the requirements of 10 CFR Part 50, Appendix H. Table 4.4.6.1-1 identifies the material specimens and specifies the schedule for removal of each specimen.

The removal of the schedule for withdrawing material specimens from the TS will eliminate the necessity of a license amendment to make changes to this schedule. However, 10 CFR Part 50, Appendix H, Section I.B.3 requires the submittal of a proposed withdrawal schedule for material specimens to the NRC and approval by the NRC before implementation. Hence, adequate regulatory controls exist to control changes to this schedule without the necessity of subjecting it to the license amendment process by including it in TS.

The licensee indicated that this schedule is currently contained in CPS Procedure No. 9846.01, Reactor Vessel Irradiation Surveillance Specimen Removal and Examination," and the CPS Updated Safety Analysis Report (USAR) Section 7.4.1.4. In addition, the licensee will include any subsequent NRC-approved revisions to this schedule in USAR updates. The inclusion of the withdrawal schedule in the USAR provides a source for this information that is readily available as a reference for NRC inspectors and other staff use. Finally, the surveillance requirements for removing material specimens and the bases section for this specification remain unchanged except for the removal of the reference to Table 4.4.6.1-1.

The licensee's proposal to modify TS 4.4.6.1.4, remove TS Table 4.4.6.1-1, and modify the corresponding Bases is consistent with the guidance provided in GL 91-01 for the removal of the schedule for the withdrawal of reactor vessel material specimens from the TS. The staff has reviewed this matter and finds that the proposed changes are acceptable.

(7) The licensee has proposed the removal of TS Table 3.6.4-1, "Containment Isolation Valves," that is referenced in TS 3.6.1.2, "Primary Containment Leakage," in Surveillance Requirements 4.6.1.2.d.4 and 4.6.1.2.h, and in TS 3.6.1.1, "Primary Containment Integrity." With the removal of this table, the licensee has proposed to modify the LCO on containment leakage rates to state the limit specified by TS 3.6.1.2.d as the following:

> "A combined leakage rate of less than or equal to 0.08 L_a, for all penetrations that are secondary containment bypass leakage paths when pressurized to Pa, 9.0 psig."

Additionally, the licensee has proposed to specify the current test pressures for Main Steam Isolation Valves (MSIVs) and hydrostatically tested valves in Surveillance Requirements 4.6.1.2.f and 4.6.1.2.h since they are removed from the TSs with the removal of TS Table 3.6.4-1.

The penetrations which constitute secondary containment bypass leakage paths are defined in the USAR (Table 6.2-47). CPS Procedure No. 9861.02F004, "Bypass Leakage Summary Sheet," identifies those penetrations which constitute secondary containment bypass leakage paths whose individual leakages are summed to verify that the total bypass leakage is within its limit. Similarly, those PCIVs in hydrostatically tested lines are identified on USAR Table 6.2-47 and the data sheets of CPS Procedure No. 9861.05, "Water Local Leak Rate Testing." As a result, the secondary containment bypass leakage paths and PCIVs in lines which are hydrostatically tested are addressed in surveillance procedures which are subject to the change control program of Section 6.8 of the CPS TS.

The remainder of the proposed changes to the TSs are administrative or editorial in nature (reflecting the TS changes evaluated herein); therefore, the staff finds the licensee's proposed changes reflect the staff's guidance of GL 91-08 and hence are acceptable.

(8) The licensee proposed to modify Surveillance Requirement 4.6.1.7 associated with TS 3.6.1.7, "Primary Containment Average Air Temperature," and Surveillance Requirement 4.6.2.6 associated with TS 3.6.2.6, "Drywell Average Air Temperature," to delete the method and list of instrument locations used to verify primary containment average air temperature and drywell average air temperature, respectively.

CPS Procedure No. 9000.01D001, "Control Room Operator Surveillance Log - Mode 1,2,3 Data Sheet," contains a list of the instruments and the method used to determine the primary containment average air temperature and drywell average air temperature. Additionally, the bases for these TSs are being revised to describe the method and controls on the number of operable instruments used to determine the average air temperatures.

The licensee's proposed changes result in a change to the TS requirements the staff has yet to review. Therefore, the staff requires more time to review this potentially generic request. These changes will be reviewed as a separate item.

(9) The licensee proposed the following changes to TS 3.6.4, "Containment Isolation Valves":

a. Revise the title to read "Primary Containment Isolation Valves."

b. Delete references to and the TS Tables 3.6.4-1, "Containment Isolation Valves," and 3.6.4-2, "Containment Isolation Trip Signals."

c. Delete the reference to TS Table 3.6.4-1 in footnote "#" to the LCO and add a statement which allows locked-closed or sealed-closed valves to be opened on an intermittent basis under administrative control.

d. Revise the Applicability Statement by deleting the reference to TS Table 3.6.4-1 and specify that all PCIVs are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, and 3; those PCIVs which isolate secondary containment bypass leakage pathways are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, 3, and "**" (when handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with the potential for draining the reactor vessel); and those PCIVs which isolate the reactor water cleanup system suction containment penetration are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, 3, and 5***. Additionally, footnote *** is relocated from TS Table 3.6.4-1 to TS 3.6.4.

e. Remove the references to TS Table 3.6.4-1 from Actions "a" and "b", to specify that the TS is applicable to PCIVs only, and add an exception from TS 3.0.4.

f. Remove the references to TS Table 3.6.4-1, make editorial changes, and clarify that only PCIVs are applicable to Surveillance Requirements 4.6.4.1, 4.6.4.2, 4.6.4.3, and 4.6.4.4.

g. Revise the Bases for TS 3/4.6.4 to add a discussion on the basis for the requirement that PCIVs which isolate the Reactor Water Cleanup (RWCU) system suction containment penetrations be OPERABLE in OPERATIONAL CONDITION 5 with any control rod withdrawn. Additionally, considerations that constitute acceptable administrative control for opening locked-closed or sealed-closed PCIVs are added to the Bases for TS 3/4.6.4 consistent with the guidance in GL 91-08.

The licensee states in its application that the list of PCIVs, PCIV isolation times, and the required differential pressure for actuation of instrument line excess flow check PCIVs are contained within CPS procedures which are subject to the procedural control requirements of CPS TS Section 6.8 and that the proposed changes comply with the guidance contained within GL 91-08. Bases changes proposed by the licensee regarding locked-closed or sealed-closed PCIVs are in conformance with the guidance provided by GL 91-08. The deletion of TS Table 3.6.4-1 and moving the appropriate footnotes and Applicable Operational Conditions column does not result in any technical changes to the operability or testing requirements for these penetrations with two exceptions. The Applicable Operational Condition column exceptions include:

a. The Applicable Operational Conditions for the RWCU system suction containment penetration are expanded to include OPERATIONAL CONDITION 3 and OPERATIONAL CONDITION 5 with any control rod withdrawn.

b. The requirement to maintain containment heating, ventilation, and air conditioning (HVAC) supply and exhaust isolation valves 1VR002A, B, and 1VQ006A, B "sealed-closed" in Operational Condition 4 as currently required by TS Table 3.6.4-1 is deleted.

The expansion of the Applicable Operational Conditions for the RWCU system containment penetration is an increase in the current conservatism contained in the CPS TSs. The requirement to maintain HVAC supply and exhaust valves closed was based on an overconservative interpretation of NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," Item II.E.4.2. Because the subject valves do not receive an automatic isolation signal in response to a loss-of-coolant accident of high radiation condition, TMI Action Plan Item II.E.4.2 requires that these valves be sealed-closed. The staff position regarding this requirement is that these valves be sealed-closed whenever the reactor is not in the cold shutdown or refueling mode, whenever containment integrity is required, and operational modes of power operation, startup, hot standby and hot shutdown (BTP CSB 6-4 and Staff Interim Position dated October 23, 1979). For pressurized water reactors these modes are 1, 2, 3, and 4 and boiling water reactors these modes are 1, 2, and 3. Therefore, the current CPS TS for HVAC supply and exhaust valves is overly restrictive. Based on the above discussions, the staff finds the proposed exceptions acceptable.

The remaining proposed changes are administrative or editorial in nature to reflect the deletion of TS Table 3.6.4-1 and to specify that particular specifications are applicable only to PCIVs. In addition, the proposed TSs conform to the recommendations contained within GL 91-08, with exceptions as noted. Therefore the staff finds the licensee's request acceptable.

(10) The licensee proposed to delete TS Table 3.6.6.2-1, "Secondary Containment Ventilation System Automatic Isolation Dampers." Additionally, the LCO Action Statements and Surveillance Requirements are modified to accommodate the removal of Table 3.6.6.2-1. Further, the licensee proposed to modify Surveillance Requirement 4.6.6.2.b to identify that this surveillance is applicable only to secondary containment isolation signals (as opposed to primary containment isolation signals) and that the isolation dampers to be verified are secondary containment isolation dampers (as opposed to the main control room isolation dampers, for example).

The licensee states in its application that the list of dampers and their isolation times are contained within CPS procedures which are subject to the procedural control requirements of CPS TS Section 6.8 and that the proposed changes are in conformance with the guidance contained within GL 91-08. The remaining proposed changes are administrative or editorial in nature to reflect the deletion of TS Table 3.6.6.2-1 and to specify that particular specifications are associated with secondary containment isolation dampers and isolation signals. These proposed changes do not change the original operability or testing requirements contained within the original specifications. Therefore, the staff finds the licensee's proposed changes acceptable.

(11) The licensee has proposed the removal of Table 3.8.4.1-1, "Containment Penetration Conductor Overcurrent Protective Devices," that is referenced in TS 3/4.8.4.1 With the removal of this table, the licensee has proposed to include the following statement for the LCO under TS 3.8.4.1:

> Primary and backup containment penetration conductor overcurrent protective devices associated with each primary containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetrations' design rating.

In addition, the licensee has proposed to revise TS 4.8.4.1 to remove the reference to Table 3.8.4.1-1. The surveillance requirement has been revised to specify that each of the required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE. Bases changes to accommodate exceptions already contained within TS Table 3.8.4.1-1 (low frequency motor-generator set power supply to the reactor recirculation pumps) are also included.

The licensee stated in its submittal that the total list of penetration circuits listed in the current TS Table 3.8.4.1-1 are contained within two CPS procedures which are subject to the procedural control requirements of CPS TS Section 6.8 and that the proposed changes reflect the guidance contained within GL 91-08. The remaining proposed changes are administrative or editorial in nature. These proposed changes do not change the original operability or testing requirements contained within the original specifications. Therefore, the staff finds the licensee's proposed changes acceptable.

(12) The licensee has proposed the removal of Table 3.4.8.2-1, "Motor-Operated Valves Thermal Overload Protection," that provides a list of valves with bypass devices that is referenced in TS 3.8.4.2. With the removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.8.4.2:

> The thermal overload protection of each valve in safety systems with a bypass device(s) integral with the motor starter shall be bypassed for those directions for which the valve performs an active safety function.

The Bases to TS 3.8.4.2 indicate that this TS is to address compliance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves [MOVs]." Since this Regulatory Guide addresses those thermal overload devices integral with the motor starter, the licensee proposes to apply this TS to only those types of MOVs which use this design. Also, since many MOV thermal overload devices are bypassed in specific safety directions (noted in TS Table 3.8.4.2-1), the licensee proposes to modify the LCO to state that the thermal overload devices are required to be bypassed in those directions for which the associated valves perform an active safety function.

The licensee's proposal is consistent with the guidance of GL 91-09 and does not technically change the intent of the TS as they currently specify. Other proposed changes to this TS are administrative in nature reflecting the removal of the TS Table, are editorial in nature, or provide clarification of the TS. Therefore, the staff finds the licensee's proposal acceptable.

(13) The remaining proposed administrative changes by the licensee include renumbering TS pages, deletion of TS Bases associated with TS 3/4.3.8, "Turbine Overspeed Protection," which was deleted by Amendment No. 60 dated October 9, 1991, and a revision to Surveillance Requirement 4.6.6.1.b.3.

On the basis of its review of this matter, the staff finds that the proposed changes to the TS for the Clinton Power Station Unit 1 are primarily an administrative change that does not alter the requirements set forth in the existing TS (exceptions noted above). However, the applicability of the operability requirements will extend to all primary containment isolation valves as noted in this evaluation. Overall, these changes will allow licensees to make corrections and updates to the list of components for which these TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TS. Therefore, the staff finds that the proposed TS changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change to a surveillance requirement. 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 (57 FR 40214). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative

procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). 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.

5.0 CONCLUSION

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: December 24, 1992