Serial No. 07-0450
Docket No. 50-423

ATTACHMENT 3

LICENSE AMENDMENT REQUEST
STRETCH POWER UPRATE

MARK-UP OF THE OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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

(1) DNC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in New London County, Connecticut in accordance with the procedures and limitations set forth in this license; Central Vermont Public Service Corporation and Massachusetts Municipal Wholesale Electric Company, pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in New London County, Connecticut in accordance with the procedures and limitations set forth in this renewed operating license;

(2) DNC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended,

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

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

(5) DNC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

3650

DNC is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

1.22 DELETED

PURGE - PURGING

1.23 PURGE or PURGING shall be any 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.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411~~ 3650 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, Reactor Coolant System highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT; and the following Safety Limits shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.17 for the WRB-1/WRB-2 DNB correlations.
1.14 WRB-2M

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

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the Reactor Core Safety Limit is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
8.	Overpower ΔT	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	1900 psia	≥ 1897.6 psia
10.	Pressurizer Pressure-High	2385 psia	≤ 2387.4 psia
11.	Pressurizer Water Level-High	89% of instrument span	≤ 89.3% of instrument span
12.	Reactor Coolant Flow-Low	90% of loop design flow*	≥ 89.8% of loop design flow*
13.	Steam Generator Water Level Low-Low	18.1% of narrow range instrument span	≥ 17.8% of narrow range instrument span
14.	General Warning Alarm	N.A.	N.A.
15.	Low Shaft Speed - Reactor Coolant Pumps	92.4% of rated speed	≥ 92.2% of rated speed

nominal loop flow

* Minimum Measured Flow Per Loop = 1/4 of the RCS Flow Rate Limit as listed in Section 3.2.3.1.a.

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT			NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
16.	Turbine Trip			
	a.	Low Fluid Oil Pressure	500 psig	≥ 450 psig
	b.	Turbine Stop Valve Closure	1% open	≥ 1% open
17.	Safety Injection Input from ESF		N.A.	N.A.
18.	Reactor Trip System Interlocks			
	a.	Intermediate Range Neutron Flux, P-6	1×10^{-10} amp	$\geq 9.0 \times 10^{-11}$ amp
	b.	Low Power Reactor Trips Block, P-7		
		1) Power Range Neutron Flux, P-10 input (Note 5)	11% of RTP**	≤ 11.6% of RTP**
		2) Turbine Impulse Chamber Pressure, P-13 input	10% RTP** Turbine Impulse Pressure Equivalent	≤ 10.6% RTP** Turbine Impulse Pressure Equivalent
	c.	Power Range Neutron Flux, P-8	$\boxed{37.5}$% of RTP** 50.0	≤ $\boxed{38.1}$% of RTP** 50.6

** RTP = RATED THERMAL POWER

(1)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\left(\frac{\Delta T}{\Delta T_0}\right)\frac{(1+\tau_1 s)}{(1+\tau_2 s)} \le K_4 \boxed{- K_5 \frac{(\tau_7 s)}{(1+\tau_7 s)} T} - K_6(T - T'')$$

Where: ΔT is measured Reactor Coolant System ΔT, °F;

ΔT_0 is loop specific indicated ΔT at RATED THERMAL POWER, °F;

$\frac{(1+\tau_1 s)}{(1+\tau_2 s)}$ is the function generated by the lead-lag compensator on measured ΔT;

τ_1 and τ_2 are the time constants utilized in the lead-lag compensator for ΔT, $\tau_1 \ge$ [*] sec, $\tau_2 \le$ [*] sec;

$K_4 \le$ [*];

$K_5 \ge$[*]/°F for increasing T_{avg} and $K_5 \le$ [*] for decreasing T_{avg};

$\frac{(\tau_7 s)}{(1+\tau_7 s)}$ is the function generated by the rate-lag compensator for T_{avg};

τ_7 is the time constant utilized in the rate-lag compensator for T_{avg}, $\tau_7 \ge$ [*] sec;

T is measured average Reactor Coolant System temperature, °F;

T" is loop specific indicated T_{avg} at RATED THERMAL POWER, $\le$ [*]°F;

$K_6 \ge$ [*]/°F when T > T" and $K_6 \le$ [*]/°F when $T \le$ T";

s is the Laplace transform operator, sec^{-1};

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

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate ≥ ~~371,920~~ 363,200 gpm and greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR), and

b. $F^N_{\Delta H} \le F^{RTP}_{\Delta H} \; [1.0 + PF_{\Delta H}(1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the COLR,

4) $PF_{\Delta H}$ = The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties of 2.4% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate to within the limits specified above and in the COLR and $F^N_{\Delta H}$ to within the above limit, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that the RCS total flow rate is restored to within the limits specified above and in the COLR and $F^N_{\Delta H}$ is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels.

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

Replace with Insert '1'

4.2.3.1.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.1.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.1.2, is assumed to exist.

4.2.3.1.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

INSERT 1 to Page 3/4 2-20

4.2.3.1.2 $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.1.3 The RCS total flow rate shall be determined to be within the acceptable range by:

a. Verifying by precision heat balance that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR within 24 hours after reaching 90% of RATED THERMAL POWER after each fuel loading, and

b. Verifying that the RCS total flow rate is ≥363,200 gpm and greater than or equal to the limit specified in the COLR at least once per 12 hours.

4.2.3.1.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.1.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.1.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Engineering Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	2	2	1, 2, 3	23
10. Emergency Generator Load Sequencer	2	1	2	1, 2, 3, 4	15
11 Cold Leg Injection Permissive, P-19	4	2	3	1,2,3	20

~~March 29, 2005~~

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

\# The Steamline Isolation Logic and Safety Injection Logic for this trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

* MODES 1, 2, 3, and 4~~, 5 and 6.~~
During fuel movement within containment or the spent fuel pool.

**** Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 7 days. After 7 days, or if no channels are OPERABLE, immediately suspend fuel movement, if applicable, and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power		
a. 4 kV Bus Undervoltage (Loss of Voltage)	2800 volts with a $\leq$ 2 second time delay.	$\geq$ 2720 volts with a $\leq$ 2 second time delay.
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	3730 volts with a $\leq$ 8 second time delay with ESF actuation or $\leq$ 300 second time delay without ESF actuation.	$\geq$ 3706 volts with a $\leq$ 8 second time delay with ESF actuation or $\leq$ 300 second time delay without ESF actuation.
9. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	1999.7 psia	$\leq$ 2002.1 psia
b. Low-Low T_{avg}, P-12	553°F	$\geq$ 552.6°F
c. Reactor Trip, P-4	N.A.	N.A.
10. Emergency Generator Load Sequencer	N.A.	N.A.
11. Cold Leg Injection Permissive, P-19	1900 psia	~~[illegible]~~ $\geq$ 1897.6 psia

5/26/98

~~February 20, 2002~~

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	S	R	Q	N.A.	N.A.	N.A.	N.A.	*
8. Loss of Power								
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	Q(1, 2)	N.A.	N.A.	1, 2, 3, 4
11 Cold Leg Injection Permissive, P-19	S	R	Q	N.A.	N.A.	N.A.	N.A.	1,2,3

TABLE 4.3-2 (Continued)

TABLE NOTATION

1. Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

2. This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated OPERABLE by the performance of an ACTUATION LOGIC TEST at least once per 92 days.

3. On a monthly basis, a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the CHANNEL CALIBRATION required once per 18 months.

4. For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP- 13900, the surveillance interval for slave relay testing is R.

* MODES 1, 2, 3, and 4, ~~5 and 6~~ e
During fuel movement within containment or the spent fuel pool.

Replace with the NEW FIGURE Next PAGE

PRESSURIZER LEVEL CONTROL

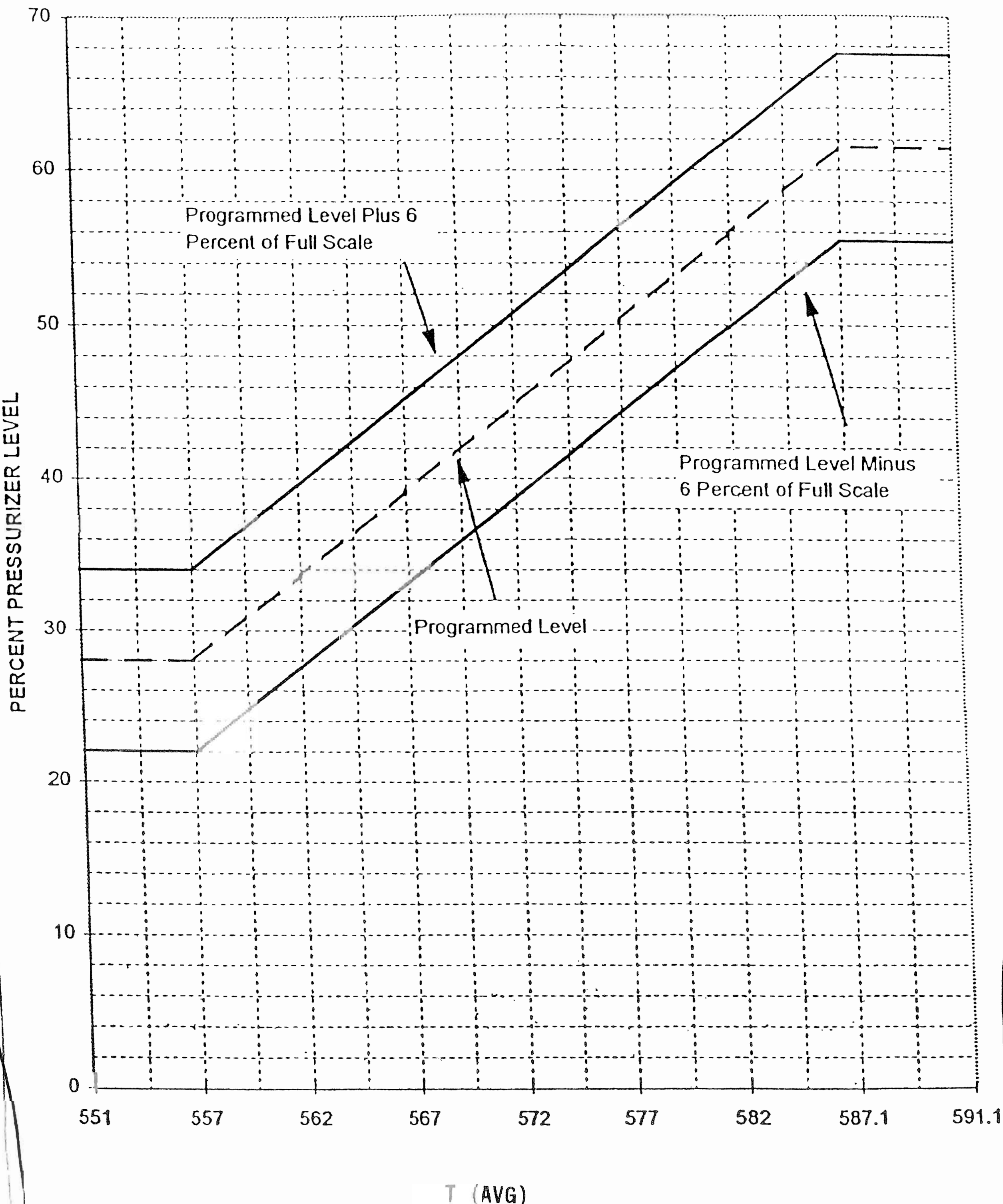

FIGURE 3.4-5

PRESSURIZER LEVEL CONTROL

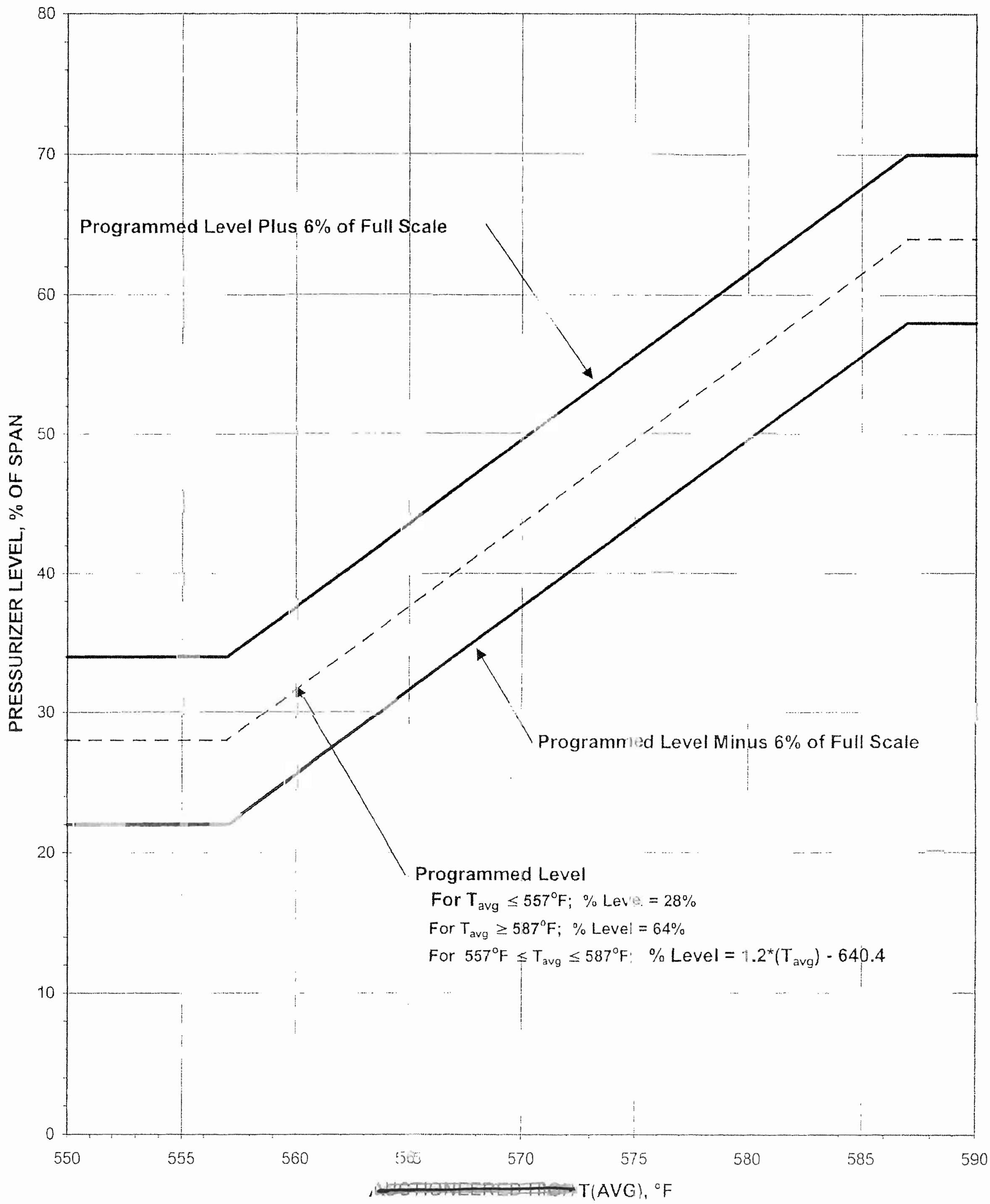

FIGURE 3.4-5

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves (MSSVs) shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION: [NOTE: separate condition entry is allowed for each MSSV.]

a. With one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT A

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

INSERT ‘A’ to Page 3/4 7-1

a. With one or more steam generators (SGs) with one MSSV inoperable, and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels, within 4 hours reduce THERMAL POWER to less than or equal to 60.1% RATED THERMAL POWER (RTP); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more SGs with two or more MSSVs inoperable, within 4 hours reduce THERMAL POWER to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs, and reduce the Power Range Neutron Flux High setpoint to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for number of OPERABLE MSSVs within the next 32 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more SGs with one MSSV inoperable and the MTC positive at any power level, within 4 hours reduce THERMAL POWER to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs and reduce the Power Range Neutron Flux High setpoint to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for number of OPERABLE MSSVs within the next 32 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or more SGs with four or more MSSVs inoperable, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

* Applicable only in MODE 1.

TABLE 3.7-1

OPERABLE MSSVs Versus Maximum Allowable Power

~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES~~

~~MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR~~ NUBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER ~~RANGE~~ ~~NEUTRON FLUX HIGH SETPOINT~~ (PERCENT OF RATED THERMAL POWER)
~~1~~ 4	~~65~~ 60.1
~~2~~ 3	~~46~~ 42.8
~~3~~ 2	~~28~~ 25.5

TABLE 3.7-2

DELETED

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Control Room Emergency Air Filtration Systems shall be OPERABLE.#

APPLICABILITY: MODES 1, 2, 3 ~~4, 5 and 6.~~ and 4.
During fuel movement within containment or the spent fuel pool.

ACTION:

MODES 1, 2, 3 and 4:

a. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Control Room Emergency Air Filtration Systems inoperable, except as specified in ACTION c., immediately suspend the movement of fuel within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With both Control Room Emergency Air Filtration Systems inoperable due to an inoperable Control Room boundary, immediately suspend the movement of fuel within the spent fuel pool and restore the Control Room boundary to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

~~MODES 5 and 6, and during~~ During fuel movement within containment or the spent fuel pool:

d. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE Control Room Emergency Air Filtration System in the recirculation mode of operation, or immediately suspend the movement of fuel.

e. With both Control Room Emergency Air Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Air Filtration System required to be in the recirculation mode by ACTION d. not capable of being powered by an OPERABLE emergency power source, immediately suspend the movement of fuel.

\# The Control Room boundary may be opened intermittently under administrative control.

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
7. FUEL BUILDING	
FB-02, Fuel Pool Pump Cubicles, El 24′6"	≤ 119
FB-03, General Area, El 52′4"	≤ 108
8. FUEL OIL VAULT	
FV-01, Diesel Fuel Oil Vault	≤ 95
9. HYDROGEN RECOMBINER BUILDING	
HR-01, Recombiner Skid Area, El 24′6"	≤ 125
HR-02, Controls Area, El 24′6"	≤ 110
HR-03, Sampling Area, El 24′6"	≤ 110
HR-04, HVAC Area, El 37′6"	≤ 110
10. MAIN STEAM VALVE BUILDING	
MS-01, Areas above El. 58′0"	≤ 140
MS-02, Areas below El. 58′0" Replace with 'Deleted'	≤ 140
11. TURBINE BUILDING	
TB-01, Entire Building	≤ 115
12. TUNNEL	
TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building	≤ 112
13. YARD	
YD-01, Yard	≤ 115

REFUELING OPERATIONS

REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL - REACTIVITY

LIMITING CONDITION FOR OPERATION

3.9.13 The Reactivity Condition of the Spent Fuel Pool shall be such that k_{eff} is less than or equal to 0.95 at all times.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION: With k_{eff} greater than 0.95:

a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and

b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3, ~~or~~ 3.9-4, or 3.9.5 to a location for which that fuel assembly is allowed.

SURVEILLANCE REQUIREMENTS

4.9.13.1.1. Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-3 by checking the fuel assembly's design, decay time and burn-up documentation.

4.9.13.1.3. Ensure that all fuel assemblies used exclusively in pre-uprate (3411 MWt) conditions which are to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel assembly's design, decay time, and burn-up documentation. Ensure that all Fuel assemblies used in post-uprate (3650 MWt) conditions which are to be placed in Region 3 Fuel storage are within the enrichment, decay time, and burn-up limits of Figure 3.9-5 by checking the fuel assembly's design, decay time, and burn-up documentation.

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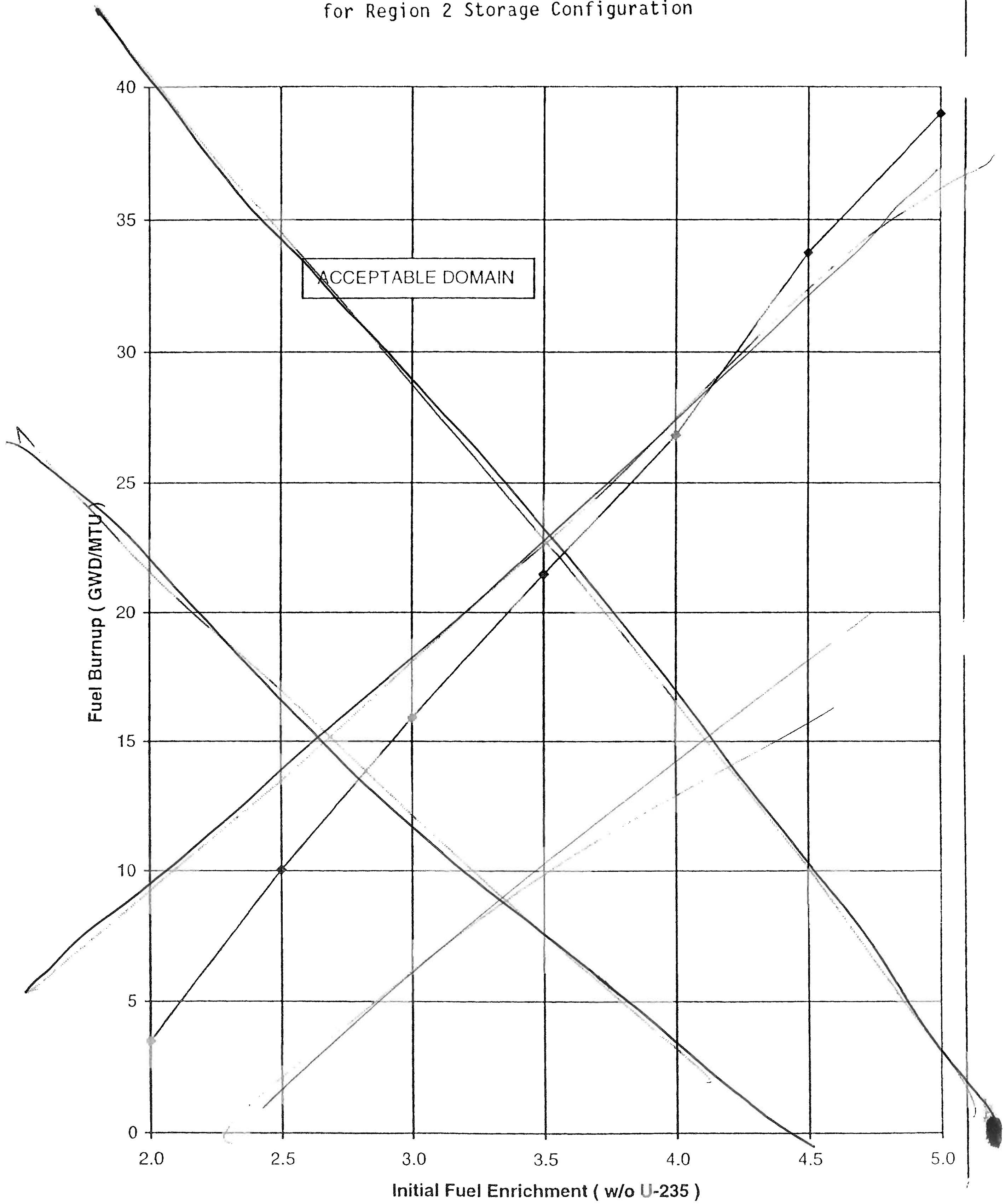
FIGURE 3.9-3 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 2 Storage Configuration



Minimum Fuel Assembly Burnup and Decay time Versus Nominal Initial Enrichment for Region 2 Storage Configuration.

Figure 3.9-3 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 2 Storage Configuration

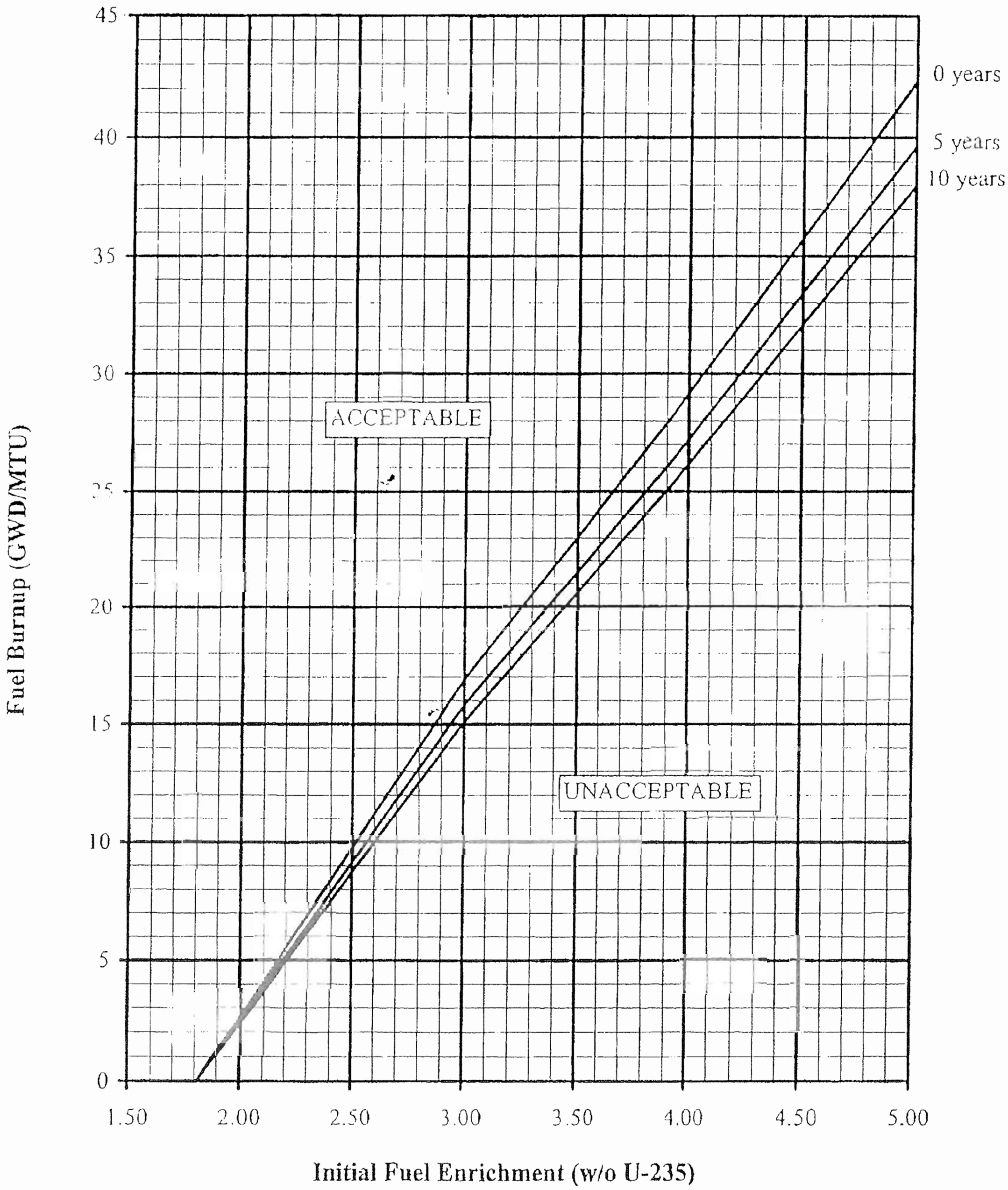


FIGURE 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Pre-Uprate (3411 MWt) Cores

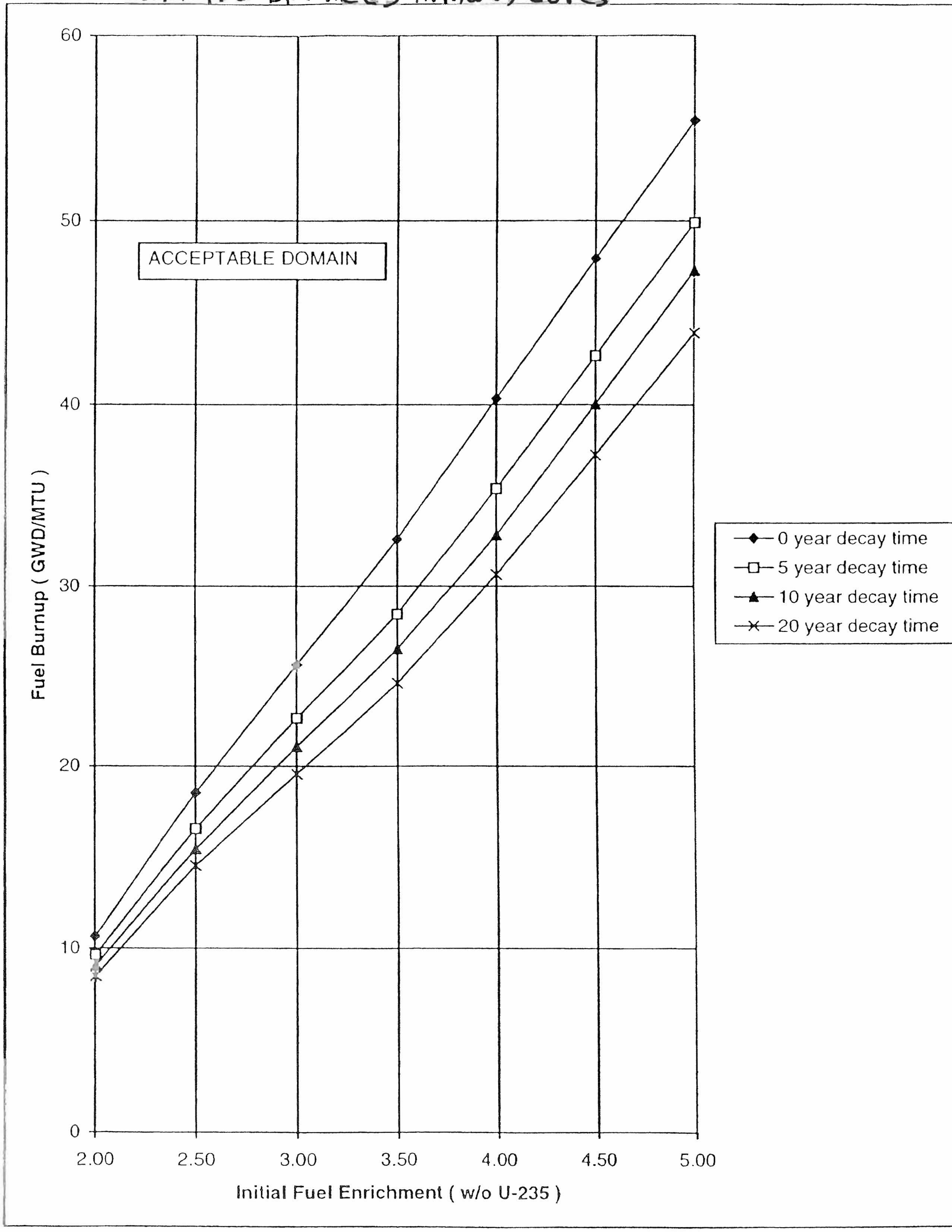


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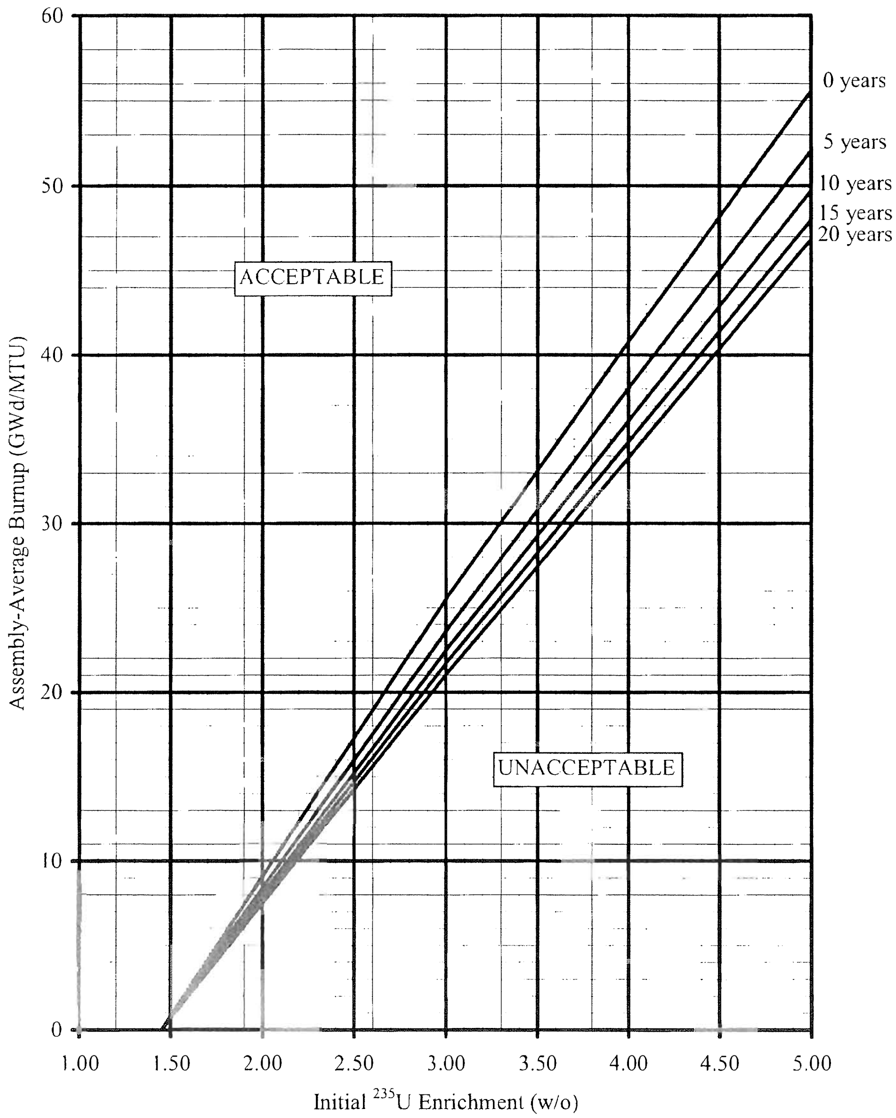
FIGURE 3.9- Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Post-Uprate (3650 MWt) Cores



5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water. The storage rack Regions are:

a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and can store fuel in 2 storage configurations:

(1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.

(2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.

and fuel decay time

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.

c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4. Fuel may be stored in all available Region 3 storage locations. The Boraflex contained inside these storage racks is not credited.

for assemblies used exclusively in pre-uprate (3411 MWt) cores or Figure 3.9-5 for assemblies used in post-uprate (3650 MWt) cores

DRAINAGE

5.6.2 The spent fuel storage pool is design and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions*. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performance Based Option of 10 CFR Part 50 Appendix J": The first Type A test performed after the January 6, 1998 Type A test shall be performed no later than January 6, 2013.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is $\boxed{38.57}$ psig. 41.4

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq\ 0.06\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

CORE OPERATING LIMITS REPORT (Cont.)

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary). (Methodology for Specifications 2.1.1.1--Departure from Nucleate Boiling Ratio, 2.1.1.2--Peak Fuel Centerline Temperature, 3.1.1.3--Moderator Temperature Coefficient, 3.1.3.5--Shutdown Bank Insertion Limit, 3.1.3.6--Control Bank Insertion Limits, 3.2.1--AXIAL FLUX DIFFERENCE, 3.2.2--Heat Flux Hot Channel Factor, 3.2.3--Nuclear Enthalpy Rise Hot Channel Factor, 3.1.1.1.1, 3.1.1.1.2, 3.1.1.2 -- SHUTDOWN MARGIN, 3.9.1.1-- Boron Concentration.)

2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), January 31, 1980--Attachment: Operation and Safety-Analysis Aspects of an Improved Load Follow Package.

3. NUREG-800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981 Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Revision 2, July 1981.

4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1--AXIAL FLUX DIFFERENCE [Relaxed Axial Offset Control] and 3.2.2--Heat Flux Hot Channel Factor [W(z) surveillance requirements for F_Q Methodology].)

WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis" (W Proprietary)

5. WCAP-9561-P-A, ADD. 3, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS--SPECIAL REPORT: THIMBLE MODELING ~~W ECCS EVALUATION MODEL,"~~ (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.)

6. ~~WCAP-10266-P-A, Addendum 1, "THE 1981 VERSION OF THE~~ WESTINGHOUSE ECCS EVALUATION MODEL USING THE BASH CODE," (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),"

7. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," (W Proprietary).

8. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL 17 USING THE NOTRUMP CODE," (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

9. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

10. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)