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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. DPR-23

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

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated February 1, 1974, as supplemented March 12, April 12 and 29, May 17 and June 4, 1974, December 29, 1977 and March 14 and 20, 1978, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 3.A and 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2300 megawatts thermal.

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 39 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

William P. Gammill, Acting Assistant
Director for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 29, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
1-1	1-1
2.1-4	2.1-4
2.1-5	2.1-5
2.1-6	2.1-6
2.1-7	2.1-7
2.1-8	2.1-8
2.3-2	2.3-2
2.3-3	2.3-3
2.3-5	2.3-5
3.1-13	3.1-13
3.1-14	3.1-14
3.1-15	3.1-15
	3.1-15a
3.4-2	3.4-2
3.4-3	3.4-3
4.1-7	4.1-7
4.1-8	4.1-8

TECHNICAL SPECIFICATIONS AND BASES

1.0 DEFINITIONS

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

1.1 Rated Power

A steady state nuclear steam supply output (reactor core thermal power) of 2300 MWt.

1.2 Reactor Operation

1.2.1 (Deleted by Change No. 21 issued 7/2/73)

1.2.2 Cold Shutdown Condition

When the reactor is subcritical and T_{avg} is $\leq$ 200 °F.

1.2.3 Hot Shutdown Condition

When the reactor is subcritical and T_{avg} is > 200 °F.

1.2.4 Reactor Critical

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

1.2.5 Power Operating Condition

When the reactor is critical and the neutron instrumentation indicates greater than 2% of rated power.

is assured. This latter limit is arbitrary but cannot be reached due to the permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculations of the curves shown in Figures 2.1-1 and 2.1-2. The figures also include the effects of uprating to 2300 MWt.[4]

The limits specified for one loop operation and natural circulation result in DNB ratios greater than 1.30.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Somewhat worse hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.10-1 ensure that the DNB ratio is always greater at part power than at full power.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of part length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required.[2] Rod withdrawal block and load runback occurs before reactor trip setpoints are reached.

The safety limit curves given in Figures 2.1-1 and 2.1-2 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 14.1 of the FSAR.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.30[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 45%.

To provide the Commission with added verification of the safety and reliability of pre-pressurized zircaloy clad nuclear fuel, a limited program of nondestruction fuel inspection will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, and other) of the two lead burnup fuel assemblies during the second and third refueling outages. Any condition observed by this inspection which could lead to unacceptable fuel performance may be the object of an expanded effort. The visual inspection program and, if indicated, the expanded program will be conducted in addition to that being performed in the Saxton and Cabrera reactors. If another domestic plant which contains pre-pressurized fuel of the same design as that used for H. B. Robinson Unit No. 2 and reaches the second and third refueling outages first, and if a limited inspection program is or has been performed there, then the program may not have to be performed at H. B. Robinson Unit No. 2. However, such action requires approval of the NRC.

References:
(1) FSAR, Section 3.2.2
(2) FSAR, Section 14.1.3
(3) FSAR, Section 7.2.1
(4) WCAP-8243, "H. B. Robinson Unit 2 - Justification for Operation at 2300 MWt," December, 1973

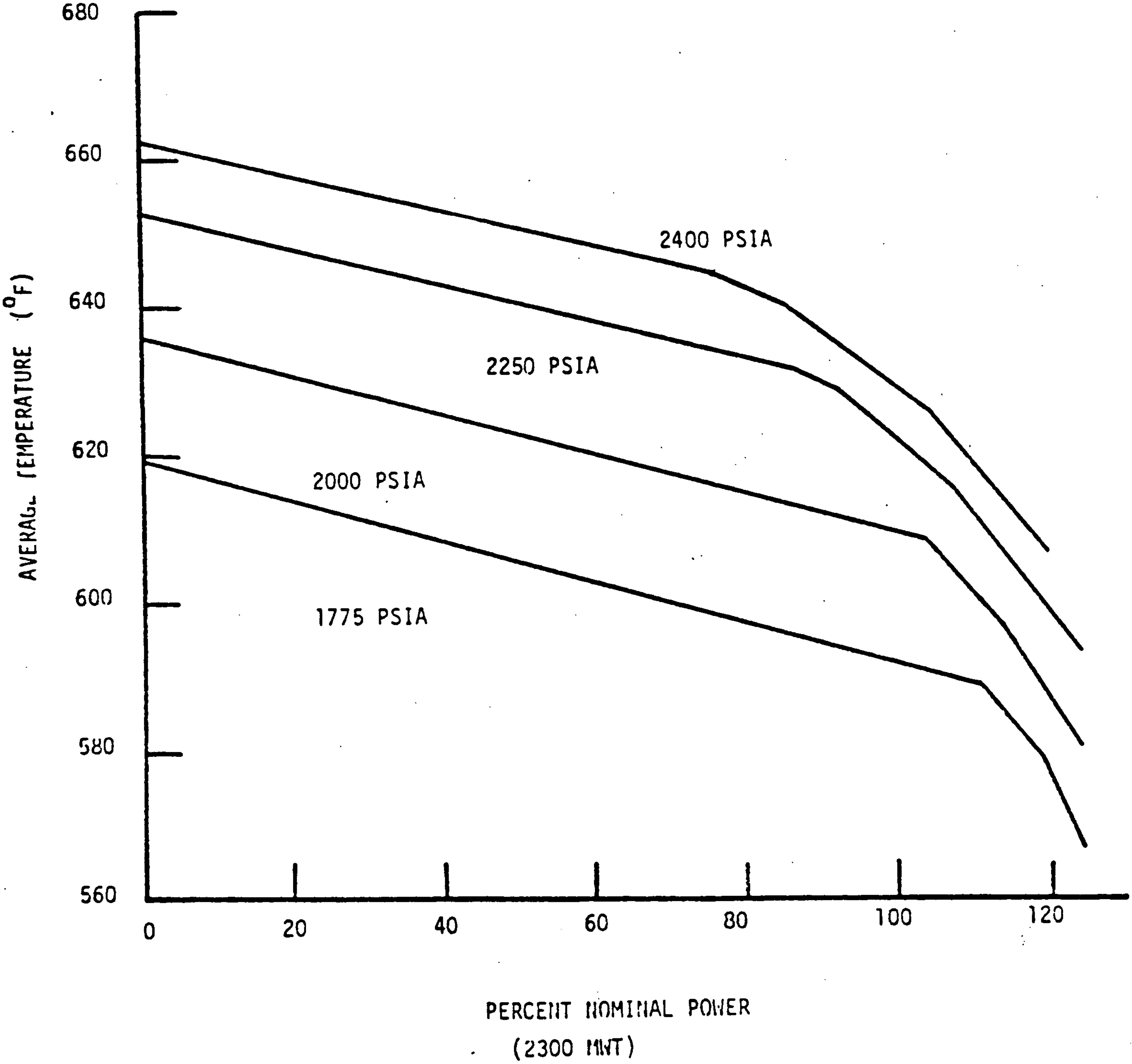
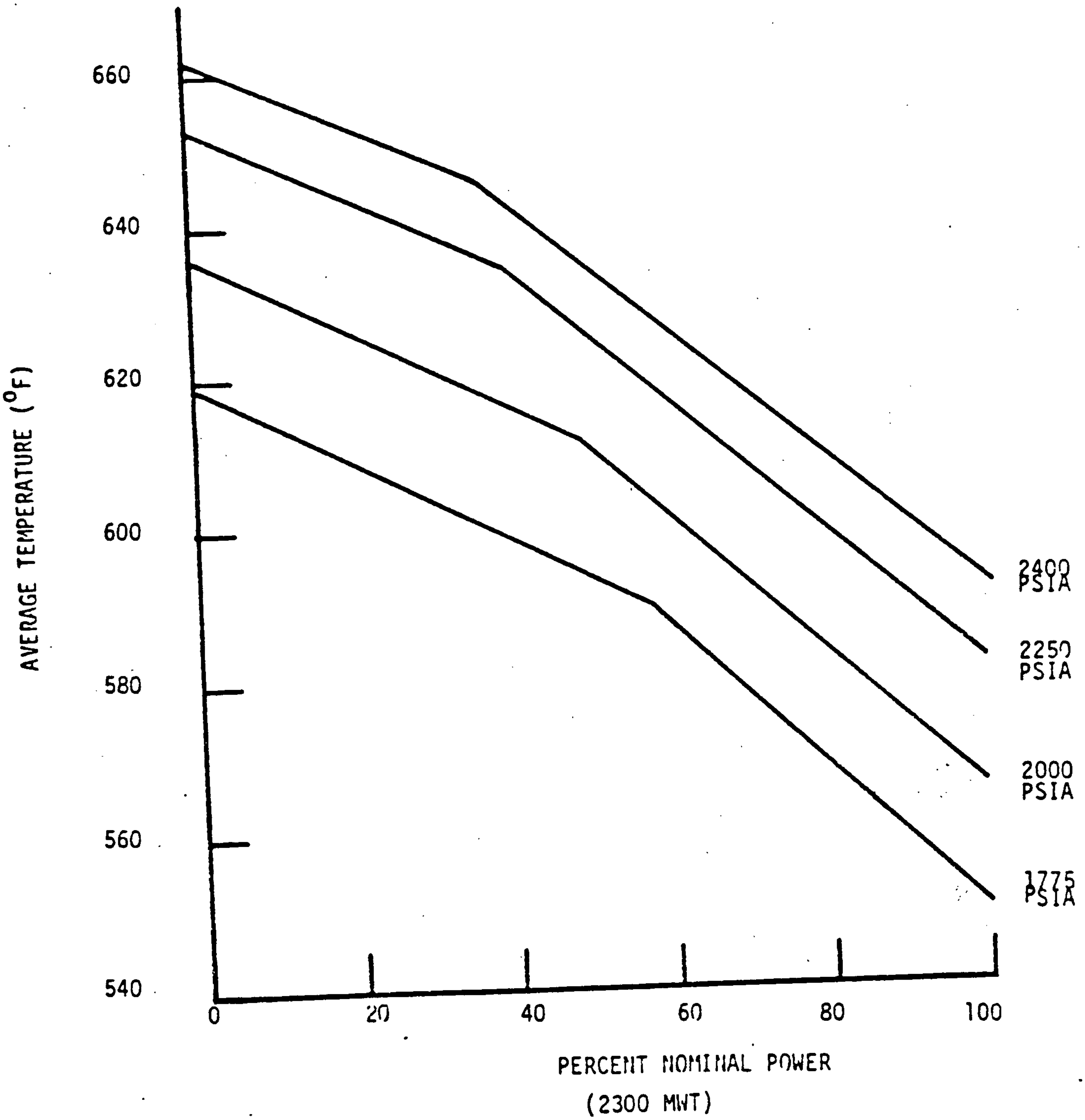

680
660
640
620
600
580
560
AVERAGE TEMPERATURE (^{o}F)
2400 PSIA
2250 PSIA
2000 PSIA
1775 PSIA
0
20
40
60
80
100
120
PERCENT NOMINAL POWER
(2300 MWT)

Figure 2.1-2

Protection Boundaries

N-1 Loop Operation

Flow = 171000 GPM



d. Overtemperature ΔT

$\leq \Delta T_o \{K_1 - K_2 (T - 575.4) + K_3 (P - 2235) - f(\Delta I)\}$

where:

ΔT_o = Indicated ΔT at rated power, °F

T = Average temperature, °F

P = Pressurizer pressure, psig

K_1 = 1.1619

K_2 = 0.01035

K_3 = 0.0007978

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $(q_t - q_b)$ within +12% and -17% where q_t and q_b are percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta I) = 0$. For every 2.4% below rated power level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power level, the permissible negative flux difference range is extended by -1 percent.

(2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power.

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17%, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power.

e. Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5 \frac{dT}{dt} - K_6 (T - T') - f(\Delta I) \right]$$

where:

ΔT_o = Indicated ΔT at rated power, °F

T = Average temperature, °F

T' = Indicated average temperature at nominal conditions and rated power, °F

K_4 = 1.07

$$K_5 = \begin{cases} 0 \text{ for decreasing average temperature} \\ 0.2 \text{ seconds per °F for increasing average temperature} \end{cases}$$

K_6 = 0.002235 for $T \geq T'$; $K_6 = 0$ for $T < T'$

$f(\Delta I)$ = as defined in d. above.

f. Low reactor coolant loop flow $\geq$90% of normal indicated flow

g. Low reactor coolant pump frequency $\geq$57.5 Hz

h. Under voltage $\geq$70% of normal voltage.

2.3.1.3 Other Reactor Trips

a. High pressurizer water level $\leq$92% of span

b. Low-low steam generator water level $\geq$5% of narrow range instrument span.

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power.

2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced.[5][6]

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 112% of design power density as discussed in Section 7.2.3 and 14.1.3 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification and the increase in rated thermal output to 2300 MWt on core safety limits. The revised setpoints in the Technical Specifications insure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figures 2.1-1 and 2.1-2.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[7] The undervoltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error[2] and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

3.1.4 Maximum Reactor Coolant Activity

The total specific activity in μCi/gram of the reactor coolant shall not exceed 1.0 μCi/gram dose equivalent I-131 and $100/\bar{E}$ μCi/gram under all modes of operation. ($\bar{E}$ is the average of beta and gamma energy (MEV) per disintegration of the specific activity.)

Whenever the reactor is critical or the average reactor coolant temperature is greater than 500°F, with the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit.

With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1.4-1, be in at least HOT SHUTDOWN with T_{avg} < 500°F within 6 hours.

With the specific activity of the primary coolant > $100/\bar{E}$ μCi/gram, be in at least HOT SHUTDOWN with T_{avg} <500°F within 6 hours.

In any operating mode, with the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of Item 1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/gram DOSE EQUIVALENT I-131.

The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.1-2.

Basis

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.1.4-1, accomodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.1.4-1 must be restricted to no more than 800 hours in any consecutive 12 month period, since the activity levels allowed by Figure 3.1.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to $<500°F$ prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

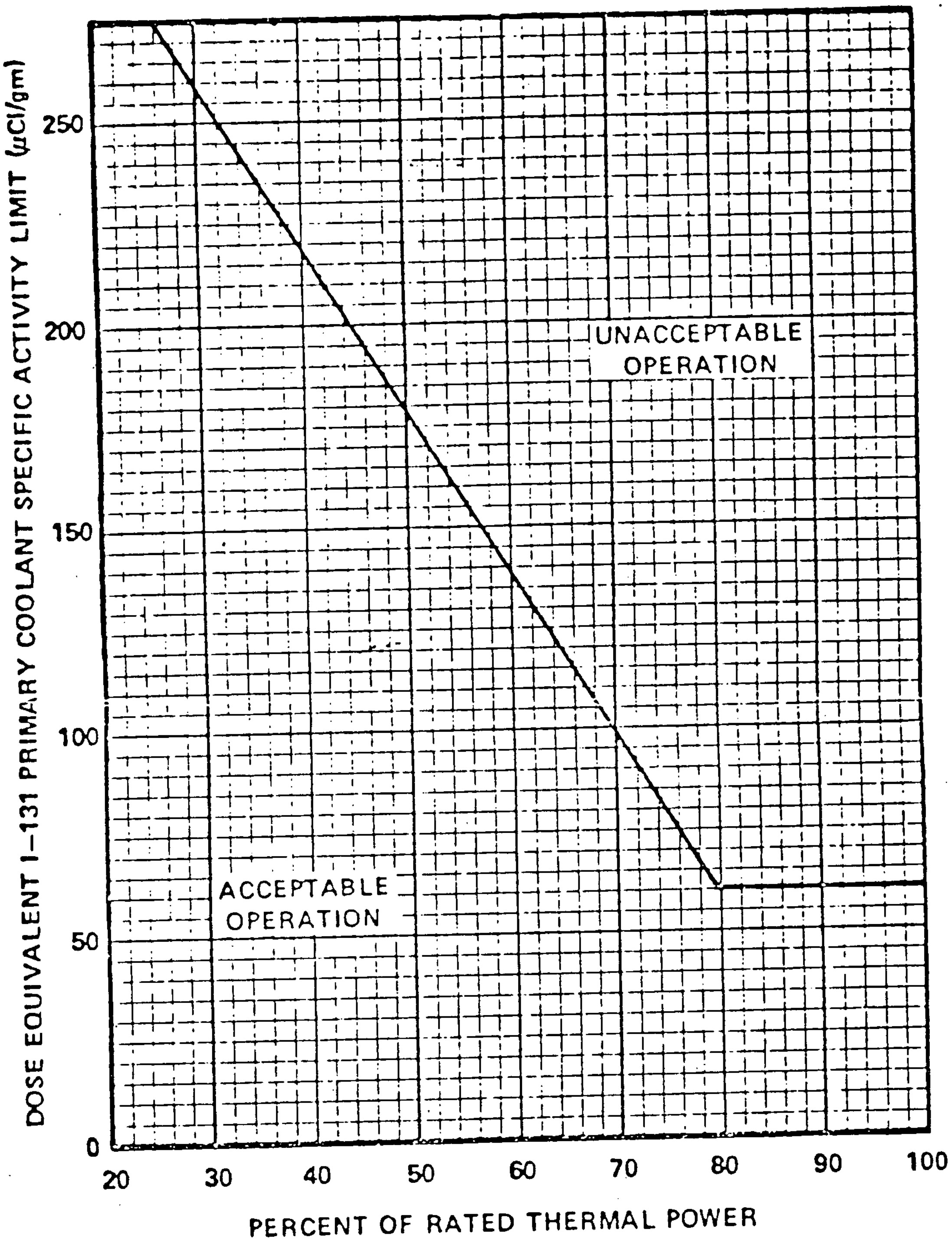


FIGURE 3.1.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

3.4.2 The specific activity of the secondary coolant system shall be ≤ 0.10 µCi/gram DOSE EQUIVALENT I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is >0.10 µCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

3.4.3 If, during power operations, any of the specifications in 3.4.1 above cannot be met within 24 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. If any of these specifications cannot be met within 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

The twelve main steam safety valves have a total combined rated capability of 10,068,845 lbs/hr. The total full power steam flow is 10,068,845 lbs/hr.; therefore, twelve (12) main steam safety valves will be able to relieve the total steam flow if necessary.(1) Following a loss of load, which represents the worst transient, steam flows are below the total capacity of the 12 safety valves. Therefore, overpressurization of the secondary system is not possible.

In the unlikely event of complete loss of turbine-generator and off-site electrical power to the plant, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps operated from the diesel generators and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant.[2] The minimum amount of water in the condensate storage tank is the amount needed for at least two-hour operation at hot standby conditions. If the outage is more than two hours, deep well or Lake Robinson water may be used.

An unlimited supply is available from the lake via either leg of the plant Service Water System for an indefinite time period.

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

References

(1) FSAR Section 10.3
(2) FSAR Section 14.2.5

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	-Gross Activity (1)	Minimum 1 Per 72 hrs.	3 days
		-Radiochemical (2)	Monthly	45 days
		-Radiochemical for $\bar{E}$ Determination	1 per 6 mos. (6)(7)	6 months
		-Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days (7)	14 days
		-Isotopic Analysis for Iodine Including I-131, I-133 and I-135	a) Once per 4 hours (8) b) One sample (9)	
		-Tritium Activity	Weekly	10 days
		-Cl & O_2	5 day/week	3 days
2.	Reactor Coolant Boron	Boron concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days
4.	Boric Acid Tank	Boron concentration	Twice/week	5 days
5.	Boron Injection Tank	Boron concentration	Weekly (5)	10 days
6.	Spray Additive Tank	NaOH concentration	Monthly	45 days
7.	Accumulator	Boron concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron concentration	Prior to Refueling	NA*
9.	Secondary Coolant	Gross activity Isotopic Analysis for Dose Equivalent I-131 Concentration	Minimum 1 Per 72 hrs. a) 1 per 31 days (10) b) 1 per 6 months (11)	3 days
10.	Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly (3)	10 days
11.	Steam Generator Samples	Primary to secondary tube leakage	5 days/week	3 days

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gram

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) When iodine or particulate radioactivity levels exceed 10% of the limit in Specification 3.9.2.1, the sampling frequency shall be increased to a minimum of once each day.

(5) The boron concentration in the boron injection tank shall be checked immediately after any actuation of the safety injection system that might result in dilution of the boron concentration in the boron injection tank.

(6) Sample to be taken after a minimum of 2EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

(7) Samples are to be taken in the power operating condition.

(8) Samples taken at all operating conditions whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or $100/\bar{E}$ μCi/gram. These samples are to be taken ur il the specific activity of the reactor coolant system is restored within ts limits.

(9) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one hour period. Samples are required when in the hot shutdown or power operating modes.

(10) Sample whenever the gross activity determination indicates iodine concentrations are greater than 10% of the allowable limit.

(11) Sample whenever the gross activity determination indicates iodine concentrations are below 10% of the allowable limit.

NA* - Not applicable.