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PLANT NAME: H. B. Robinson Unit #2

ENCLOSURES:
(1) Petition requesting Amdt of lic & Amdt of Tech Specs (25 cys)
(2) Revisions to Tech Specs (40 cys)
(3) Revisions to FSAR (40 cys)
(4) Cert of Sv: showing sv of the proposed changes upon H. L. Gardner chrmn of Darlington Co. Board of Comm, Darlington, S.C S. C. (30 cys)

FOR ACTION/INFORMATION 2-6-74 GC

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Carolina Power & Light Company

February 1, 1974

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File: NG-3514

Serial: NG-74-144

Mr. John F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

RECEIVED FEB 5 1974 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Dear Mr. O'Leary:

50-261

H. B. ROBINSON UNIT 2
FACILITY OPERATING LICENSE DPR-23
APPLICATION FOR OPERATION AT 2300 MWt CORE POWER

Enclosed are three originals and 25 copies of the Petition Requesting Amendment to Facility Operating License DPR-23 to allow operation of H. B. Robinson Steam Electric Plant Unit 2 at a core power level of 2300 MWt. Also enclosed for your review are 40 copies of proposed page changes to the Final Safety Analysis Report (FSAR) and the Technical Specifications to reflect operation at 2300 MWt.

As previously indicated in our letter of December 17, 1973, H. B. Robinson Unit 2 is presently licensed and operating at a core power level of 2200 MWt and is over halfway through the second fuel cycle. Operating data indicates that the original safety analyses were performed using extremely conservative design data. Hot channel factors and maximum linear heat output experienced during the first cycle of operation are significantly smaller than design values and are smaller than those required to meet the AEC Interim Acceptance Criteria for the loss of coolant accident analysis at a core power level of 2300 MWt. In addition, the fuel regions in which clad flattening can be postulated to occur are to be removed and replaced with fuel of improved design during the next refueling. This refueling outage is scheduled to commence April 20, 1974.

The reanalysis of postulated accidents and other relevant data in support of the application for operation at a core power level of 2300 MWt are presented in the report "H. B. Robinson Unit 2 - Justification for Operation at 2300 MWt," WCAP-8243 (Proprietary) and WCAP-8244 (Non-Proprietary) previously submitted to the AEC on December 17, 1973. These reanalyses have been incorporated into the proposed page changes to the FSAR.

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Operation of Unit 2 at a core power of 2300 MWt will result in a net gain of approximately 30 MWe to the CP&L system. If the Company is able to operate at the higher power level on the completion of the second refueling in May, 1974, its ability to meet anticipated load with sufficient reserves, particularly during the summer months will be enhanced. Additionally, the maximum utilization of nuclear generating capacity will have a favorable economic impact on the Company, and perhaps more importantly, serve to conserve the critical supply of fossil fuels.

The Company respectfully requests a timely review of this application.

Yours very truly,

J. A. Jones
Executive Vice-President

JAJ:mvp

Enclosure

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of)
) Docket No. 50-261
Carolina Power & Light Company)

PETITION REQUESTING AMENDMENT OF

LICENSE AND AMENDMENT OF

TECHNICAL SPECIFICATIONS

Carolina Power & Light Company, Licensee in the above-entitled cause, hereby files this petition to amend its Facility Operating License No. DPR-23 and incorporated Technical Specifications for its H. B. Robinson Unit 2, (the facility) located in Darlington County, near Hartsville, South Carolina. Pursuant to Condition A of the Facility Operating License, the Company is authorized to operate the facility at steady state core power levels not in excess of 2200 megawatts thermal. The Company hereby petitions to amend Condition A in order to authorize operation of the facility at a steady state core power level of 2300 megawatts thermal. In support of this request Licensee states as follows:

(1) Licensee received from the Atomic Energy Commission its Facility Operating License No. DPR-23 on July 31, 1970, and authorization for operation at a core power level of 2200 megawatts thermal was received on September 23, 1970. Commercial operation of the facility was achieved

in March 1971 and the facility operated commercially until March 16, 1973, when the unit was shut down for the first refueling. The facility, upon refueling, again achieved criticality on May 14, 1973, and was synchronized to the Carolina Power & Light Company system on May 16, 1973.

(2) The facility is presently operating in the second fuel cycle, and this period of operation will extend to April 20, 1974, at which time the facility is scheduled for a refueling outage. Although initial operation of the unit during the second fuel cycle was limited to 75 percent of rated power, the Company was authorized to operate the unit at a core power level not to exceed 2200 megawatts thermal on July 25, 1973, upon completion of the Atomic Energy Commission review of fuel densification and incore surveillance methodology.

(3) At the time of the design and construction of H. B. Robinson Unit 2 it was the intent of Carolina Power & Light Company as documented in both the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR) to obtain authorization to operate this facility at a core power level of 2300 megawatts thermal; accordingly, the Company had the components of the facility designed and constructed to accommodate operation of the facility at this level.

(4) Licensee has previously submitted for review a document entitled "H. B. Robinson Unit 2 - Justification for Operation at 2300 MWt" WCAP-8243, and such document is hereby incorporated by reference. This document contains, among other things: (a) a reanalysis of postulated accidents at the higher power level; (b) a review of the Power Test Program which will be used during escalation to 2300 megawatts thermal and after the maximum power level has been established; (c) a brief review of the initial start-up program for the facility; and (d) a summary of the operational performance of the facility for the period March 1971 to October 31, 1973. WCAP-8243, "H. B. Robinson Unit 2, Justification for Operation at 2300 MWt," Westinghouse, (December 1973). The reanalyses and other supportive data have been incorporated into the proposed page changes to the Final Safety Analysis Report (FSAR) also incorporated herein by reference.

(5) Since March 1971, the facility has been operated on a commercial basis for a period of time which exceeds two years. Since October 31, 1973, the date of the latest operational data covered in WCAP-8243, the facility has operated on a continuous basis with the exception of the

period November 22 - December 2, 1973. Operating data acquired from initial operation of the facility to the present date confirms that the initial analyses of postulated accidents were performed utilizing conservative data and design parameters. See also H. B. Robinson Unit 2 Semi-Annual Operating Report, Nos. 1-6 (1971-1973).

(6) Westinghouse Electric Corporation is presently analyzing the changes in evaluation models and the resultant impact on all Westinghouse facilities as a result of the Atomic Energy Commission decision in Rulemaking Proceeding RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors" rendered December 28, 1973. Reanalysis of postulated accidents for H. B. Robinson Unit 2 will be undertaken to reflect the Final Acceptance Criteria and the results of this reanalysis will be submitted by the Company in accordance with the schedule established by the Commission.

(7) Operation of the H. B. Robinson Unit 2 at 2300 megawatts thermal will result in a net gain of generation of 30 megawatts electrical representing an increase of over 4 percent in electrical generation capability for this facility. This increase in electrical generation will require no capital

expenditure since all steam and power conversion equipment, including the turbine generator, is designed and constructed to permit operation at the increased power level.

(8) On January 28, 1974, the Federal Energy Office allocated to the Company approximately 36 percent of the oil supplies previously utilized at facilities near Wilmington, North Carolina. This will require that the Company seek to burn coal rather than oil at these facilities. The Company is experiencing great difficulty in obtaining adequate coal supplies to fuel these fossil units and the addition of 30 megawatts electrical to the system from H. B. Robinson Unit 2 would help offset this problem, and the fuel costs for this generation would be approximately one-fourth (1/4) of fuel costs if coal is utilized, based on present costs.

WHEREFORE, Licensee prays as follows:

1. That Condition A, Facility Operating License DPR-23 be amended to authorize operation of H. B. Robinson Unit 2 at a steady state core power level of 2300 megawatts thermal, and

2. That the Technical Specifications and FSAR be amended to reflect operation at the higher power levels.

CAROLINA POWER & LIGHT COMPANY

BY: ______________________
J. A. Jones
Executive Vice President

ATTESTED: ______________________
J. L. Lancaster, Jr.
Secretary

Sworn to and subscribed to before me this 4th day of February, 1974.

Nancy A. Stephens (Yancey)

Notary Public

My commission expires: June 29, 1976

NANCY A. STEPHENS
NOTARY
PUBLIC
WAKE COUNTY, N.C.

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-261
H. B. Robinson, Unit 2	)	

CERTIFICATE OF SERVICE

This is to certify that a copy of the Petition Requesting Amendment to Facility Operating License DPR-23 and proposed page changes to the Final Safety Analysis Report and the Technical Specifications has this 4th day of February, 1974, been served upon the following by deposit of same in the United States mail, addressed as follows:

Mr. Harrell L. Gardner
Chairman - Darlington County Board of Commissioners
Route 2
Darlington, South Carolina 29532

W. Brian Howell

W. Brian Howell
Associate General Counsel
Carolina Power & Light Company

Business Address: 336 Fayetteville Street
Raleigh, N. C. 27602

Dated: February 4, 1974

Business Telephone: Area Code 919
828-8211

981

50-261

REVISIONS TO TECHNICAL SPECIFICATIONS

H. B. ROBINSON UNIT NO. 2

2300 MWt CORE POWER OPERATION

JANUARY, 1974

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

1.2.2 Cold Shutdown Condition

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

1.2.3 Hot Shutdown Condition

"When the reactor is subcritical and T_{avg} is $> 200^{o}F$."

1.2.4 Reactor Critical

When the neutron chain reaction is self-sustaining and $K_{eff} \doteq 1.0$.

1.2.5 Power Operating Condition

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists and shall not exceed the limits shown in Figure 2.1-2 when the full flow from two reactor coolant pumps exists.

b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 590°F.

c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 620°F.

d. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in Figures 2.1-1 or 2.1-2 or if the thermal power level, coolant pressure, or Reactor Coolant System average temperature violates the limits specified above.

(DELETE)

Basis:

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, thermal power, reactor coolant temperature and pressure, have been related to DNB through the L-Grid DNB correlation. The L-Grid DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during normal operational transients and anticipated transients (those transients listed on page 14.1-1 of the FSAR) is limited to 1.30. A DNB ratio of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1] The DNB ratio limit of 1.30 is a conservative design limit which is used at the basis for setting core safety limits. Based on rod bundle DNB tests, no fuel rod damage is expected at this DNB ratio or greater.

The curves of Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is not less than 1.30. The area where clad integrity is assured is below these lines. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperature are shown for each pressure at powers lower than approximately 75%. The temperature limits at low power are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30 but are such that the plant conditions required to violate the limits are precluded by the self actuated safety valves on the steam generators. An arbitrary upper safety limit of 120% for thermal power is shown. The upper limit is below the damage limit of 1.7% for maximum clad strain which is reached at 123% thermal power with design hot channel factors.

The curves of Figure 2.1-2 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperatures are shown for each pressure at powers lower than approximately 40%. The limits at low power as well as the limits based on the average enthalpy at the exit of the core are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30. The plant conditions required to violate these limits are precluded by the protection system and the self actuated safety valves on the steam generator. An upper limit of 70% for power is shown to completely bound the area where clad integrity 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.

The specified limits are based on $F^N_{\Delta H}$ of 1.55, a 1.55 cosine axial flux shape and a DNB analysis as described in Section 4 of the report Fuel Densification - H. B. Robinson Steam Electric Plant, Unit 2 (WCAP-8114).

The recalculated DNB core safety limits have been found to be less limiting than those previously presented in the FSAR, i.e., the reduction in DNB penalties due to densification and removal of the clad flattening penalty more than offsets the effects of the increased power level on the DNB ratio.

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 AEC.

(DELETE)

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

Figure 2.1-1

Protection Boundaries

N-Loop Operation

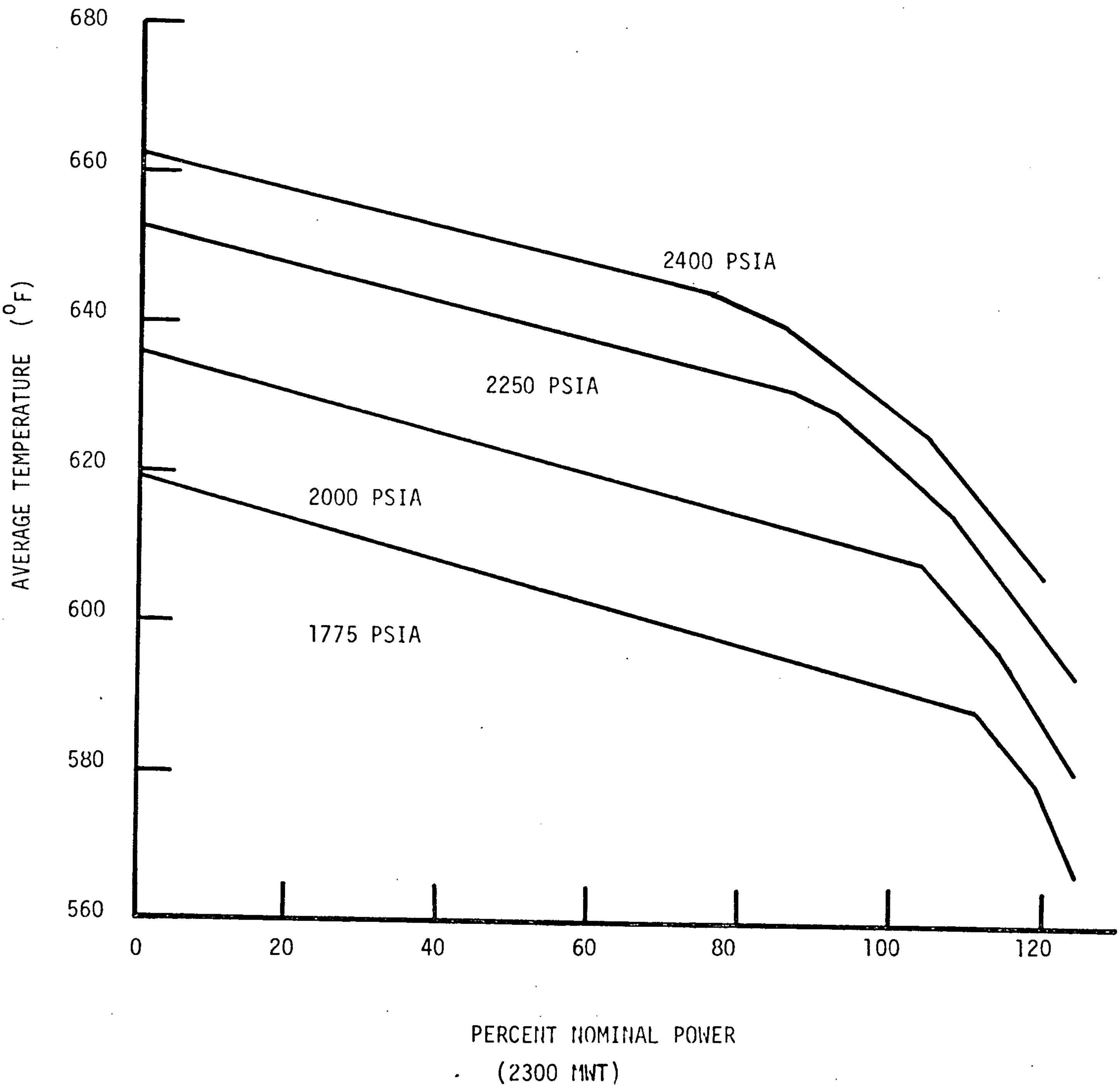
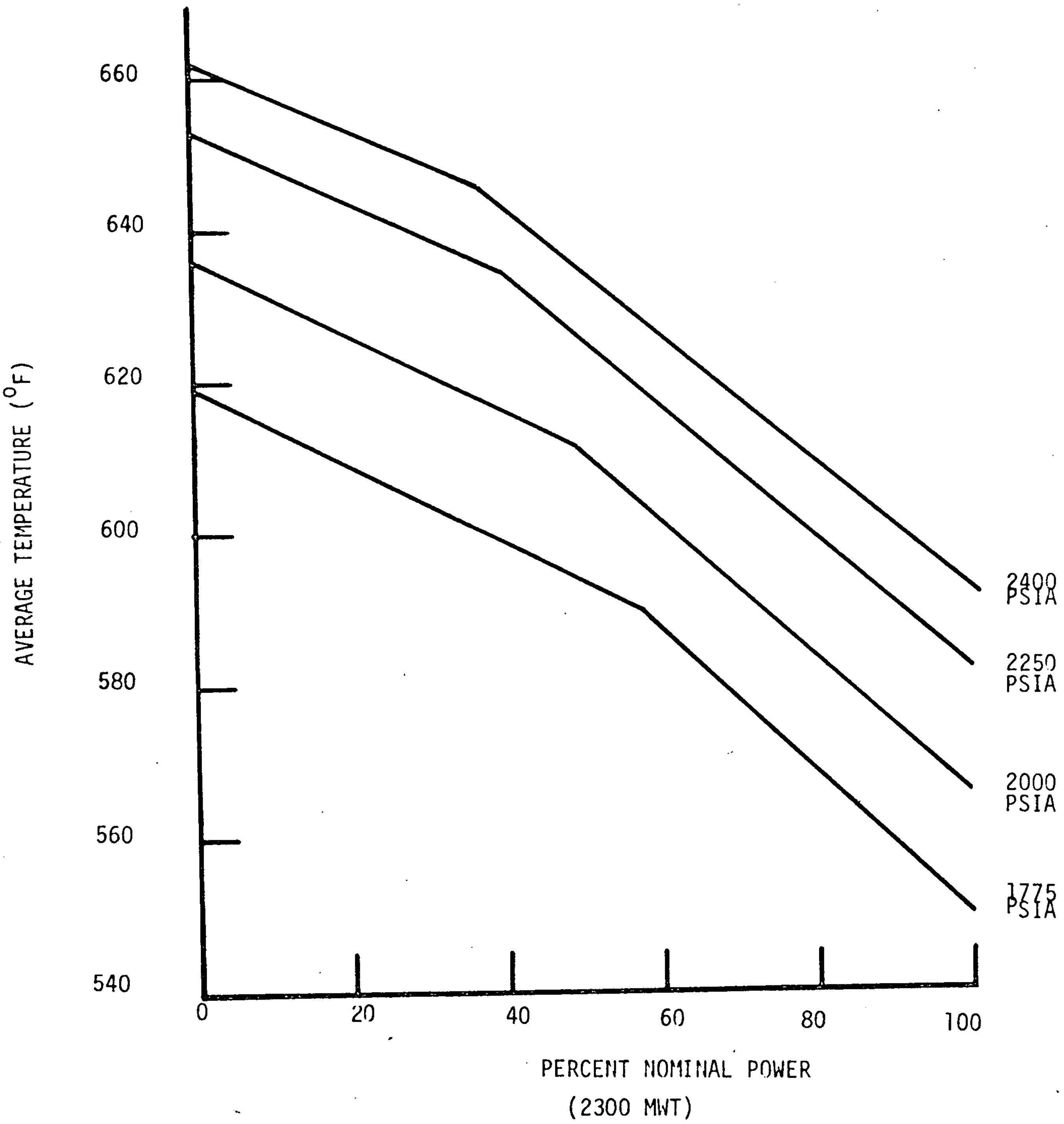


Figure 2.1-2

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 T at rated power, ^oF

T = Average temperature, ^oF

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

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

where

ΔT_o = Indicated ΔT at rated power, ^oF

T = Average temperature, ^oF

T' = Indicated average temperature at nominal conditions and rated power, ^{o}F

K_4 = 1.07

K_5 = {0 for decreasing average temperature
{0.2 seconds per ^{o}F for increasing average temperatur

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

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[4], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors, (2) is always below the core safety limit as shown in Figure 2.1-1 or 2.1-2. If axial peaks are 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 setpoint 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 under voltage 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.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.(8)

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 45% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.30 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14).

References

1) FSAR Section 14.4.1
2) FSAR Page 14-3
3) FSAR Section 14.3.1
4) FSAR Section 14.1.2
5) FSAR Section 7.2.2, 7.2.3
6) FSAR Section 3.2.1
7) FSAR Section 14.1.6
8) FSAR Section 14.1.11

3.1.5 LEAKAGE

Specification:

3.1.5.1 If the primary system leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.1.5.2 If the sources of leakage have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.1.5.3 If the primary to secondary leakage in a steam generator exceeds 1 gpm the reactor shall be placed in the hot shutdown condition within 8 hours utilizing normal operating procedures. If the leakage exceeds this limit for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Basis:

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System, and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage is a conservative limit on what is allowable before the guidelines of 10 CFR Part 20 would be exceeded. This is shown as follows: If the reactor coolant activity as $50/\bar{E}$ uCi/cc ($\bar{E}$ = average beta plus gama energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, the yearly whole body dose resulting from this activity at the site boundary, using an annual average X/Q = 2.00 x 10^{-5} sec/m^3 is about the 10 CFR Part 20 guideline of 0.5 R/yr[1,2].

With the limiting reactor coolant activity and assuming initiation of 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet by cooling coils of the main recirculation units. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks, from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 0.5 gpm to greater than 10 gpm can be detected and measured by this system. Condensate flow corresponding to coolant leakage of approximately 1 gpm can be detected within 10 minutes.

Leaks less than 1 gpm can be measured by periodic observation of the level changes in the condensate collection system.

If leakage is to another closed system, it will be detected by the plant radiation monitors and/or inventory control.

Steam generator tube leakage limits are based upon offsite dose considerations as limited by 10 CFR Part 20 in the event of a 112% overpower transient with the presence of collapsed rods.

(DELETE)

Operator action to start to place the reactor in the hot shutdown condition within 12 hours utilizing normal operating procedures provides adequate time for an orderly reduction of power. The hot shutdown condition allows personnel to enter the containment and inspect the pressure boundary for leaks. The 24 hours allowed prior to the operator starting to place the reactor in the cold shutdown conditon utilizing normal operating procedures allows reasonable time to correct small deficiencies. If major repairs are needed, a cold shutdown conditon would be in order.

Figure 3.1-4

(Deleted)

3.4.2 The iodine-131 activity on the secondary side of a steam generator shall not exceed 0.24 uCi/cc.

3.4.3 If, during power operations, any of the specifications in 3.4.1 or 3.4.2 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 2-hours operation at hot standby conditions. If the outage is more than 2 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.

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specifications:

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be no further inserted than the limits shown by the solid lines on Figure 3.10-1 for 3 loop or 2 loop operation.

3.10.1.4 After 50% of the second and subsequent cycles as defined by burnup, the limits shall be adjusted as a linear function of burnup toward the end-of-core life values as shown by the dotted lines on Figure 3.10-1.

3.10.1.5 Except for physics tests, if a part-length or full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1.1, or

c. Limit power to 70% of rated power for 3 loop operation or 45% of rated power for 2 loop operation.

(Delete)

3.10.1.6 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod inserted and part length rods fully withdrawn.

3.10.2 Power Distribution Limits

3.10.2.1 Limiting Values

3.10.2.1.1 Power distribution limits are expressed as hot channel factors. Limiting values at rated power are:

$F_q^N = 2.57\{1 + 0.2(1 - P)\}$ in the flux difference range -17 percent to +12 percent

$F_{\Delta H}^N = 1.55\{1 + 0.2(1 - P)\}$

Where P is the fraction of rated power at which the core is operating ($P \leq 1.0$).

If measured peaking factors exceed these values, the maximum allowable reactor power level and the nuclear overpower trip setpoint shall be reduced in direct proportion to the amount which F_q^N or $F_{\Delta H}^N$ exceeds the limiting values, whichever is more restrictive. If the $F_{\Delta H}^N$ or F_q^N cannot be reduced below the limiting values within twenty-four hours, the over-power ΔT and overtemperature ΔT trip setpoint shall be similarly reduced.

3.10.2.1.2 At all times the measured value of axial offset must either be between -17% and +12% or the overtemperature and overpower ΔT trip setpoints must be reduced as required in Specifications 2.3.1.2(d) and (e).

3.10.2.1.3 Following initial loading and each subsequent reloading, a power distribution map, using the Movable Detector System, shall be made to confirm that power distribution limits are met, in the full power configuration, before the plant is operated above 75 percent of rated power.

3.10.2.1.4 At regular monthly intervals, a power distribution map, using the Moveable Detector System, shall be made to confirm that power distribution limits are met.

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Whenever the indicated quadrant power tilt ratio exceeds 1.02*, the tilt condition shall be eliminated within two hours, or in the event the tilt condition cannot be eliminated within two hours, the following actions shall be taken:

a) Restrict core power level and reset the power range high flux setpoint to be less two percent for every percent of indicated power tilt ratio exceeding 1.0, and

b) If the tilt condition is not eliminated after 24 hours, the power range high flux setpoint shall be reset to 75% of allowed power. Subsequent reactor operation would be permitted up to 70% power for the purpose of measurement and testing to identify the cause of the tilt condition.

3.10.3.2 If the indicated quadrant power tilt ratio exceeds 1.10, the reactor would be permitted to operate up to 50% power for the purpose of measurement and testing to identify the cause of the tilt and correct it, except if the cause of the tilt is a misaligned rod and the power level is less than 75%, then two hours are allowed to recover the rod. If the tilt cannot be corrected within 24 hours the plant will be limited to 5% power for testing purposes only.

3.10.4 Rod Drop Time

3.10.4.1 The drop time of each control rod shall be no greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

3.10.5 Part Length Control Rod Banks

3.10.5.1 The eight (8) part length control rods shall be configured under administrative control into one of the following part length rod configurations.

*A greater power tilt is permitted for physics testing.

a. Four part length rods occupying core positions K-6, K-10, F-6, and F-10 shall constitute a part length control rod bank, hereafter designated bank P-1.

b. Four part length rods occupying core positions P-8, H-2, H-14, and B-8 shall constitute a part length control bank, hereafter designated part length bank P-2.

c. Combined Banks P-1 and P-2, hereafter designated Bank P-3.

3.10.5.2 The part length control rods will not be inserted. They will remain in the fully withdrawn position except for physics tests and for axial offset calibration which will be performed at 75% of permitted power or less.

3.10.6 Inoperable Full Length and Part Length Control Rods

3.10.6.1 A full length or part length control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) if the rod cannot be moved by its drive mechanism, or (c) if its rod drop time is not met in the case of a full length rod.

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation.

3.10.6.3 If a full length control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that shutdown margin equal to or greater than shown on Figure 3.10-2 results. If sustained power operation is anticipated, the rod insertion limit shall be adjusted to reflect the worth of the inoperable rod.

3.10.7 <u>Power Ramp Rate Limits</u>

3.10.7.1 Should a power level less than 95% be maintained continuously for more than 100 hours but less than 24 days, the rate of power increase shall be limited to 10% per hour.

3.10.7.2 Should a power level less than 95% be maintained continuously for more than 24 days, the rate of power increase shall be limited to 3% per hour from 25% to 100% of full power.

3.10.8 <u>Required Shutdown Margins</u>

3.10.8.1 When the reactor is in the hot shutdown condition, the shutdown margin shall be at least that shown in Figure 3.10-2.

3.10.8.2 When the reactor is in the cold shutdown condition, the shutdown margin shall be at least 1% $\Delta k/k$.

3.10.8.3 When the reactor is in the refueling operation mode, the shutdown margin shall be at least 10% $\Delta k/k$.

3.10.9 <u>Quadrant Power Tilt Monitor</u>

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

<u>Basis</u>:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection and provide for acceptable nuclear peaking factors. The solid lines shown in Figure 3.10-1 meet the shutdown requirement for the first 50% of Cycle 3. The end-of- cycle life limit is represented by the dotted lines. The end-of-cycle life limit may be determined on the basis of plant startup and operating data to provide a more realistic limit which will allow for more flexibility in plant operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to 1.77% reactivity(3) at end of life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin. The specified control rod insertion limits have been revised for Cycle 3 in order to meet the design basis criteria on (1) potential ejected control rod worth and peaking factor (4), (2) radial power peaking factors, $F_{\Delta H}$, and (3) required shutdown margin.

The various control rod banks (shutdown banks, control banks, and part length rods) are each to be moved as a bank, that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position(2). The 15-inch permissible misalignment provides an enforceable limit below which design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.1.5 are acceptable because complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power to 70% will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.1.6) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First the peak value of linear power density must not exceed 21.1 kW/ft. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the initial steady state conditions for the peak linear power for a loss-of-coolant accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the Interim Policy Statement for LOCA. To aid in specifying the limits on power distribution the following hot channel factors are defined.

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

F_Q^E, Nuclear Heat Flux Hot Channel Factor is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNBR at full power are met, provided:

$$F_Q^N \leq 2.57 \text{ and } F_{\Delta H}^N \leq 1.55$$

These quantities are measurable although there is not normally a requirement to do so. Instead it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met; these conditions are as follows.

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

2. Control rod banks are sequenced with overlapping banks as shown in Figure 3.10-1.

3. The control bank insertion limits are not violated.

4. Part length control rods are not inserted.

5. Axial power distribution guide lines, which are given in terms of flux difference control, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined on the difference in power between the top and bottom halves of the core. Calculation of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with axial offset. The correlation shows that the design power shape is not exceeded if the axial offset (flux difference) is maintained between +15 percent and -20 percent.

For operation at a fraction P of full power the design limits are met, provided,

$F_Q^N \leq 2.57\{1 + .2\ (1-P)\}$ in the flux difference range -17 percent to +12 percent

and $F_{\Delta H}^N \leq 1.55\{1 + .2\ (1-P)\}$,

where P is the fraction of full power at which the reactor is operating: $0 \leq P \leq 1.0$.

The permitted relaxation allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

For transient events the core is protected from exceeding 21.1 KW/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature. Only conditions 1 through 4, above, are mandatory since the flux difference is an explicit input to the protection system.

Measurements of the hot channel factors are required as part of start-up physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_q^N there is a 5% allowance for uncertainties (1) which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured $F_q^N \leq 2.57/1.05$, for example, at rated power even on a worst case basis. When a measurement is taken experimental error must be allowed for and 5% is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

The measured value of F_q^N must be additionally corrected by including a penalty as shown in Figure 3.10-3 (at the appropriate core location) to account for fuel densification effects before comparison with the limiting value above.

In the specified limit of $F_{\Delta H}^N$ there is an 8% allowance for design prediction uncertainties, which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$ at rated power. The uncertainty to be associated with a measurement of $F_{\Delta H}^N$ by the movable incore system on the other hand is 3.65% which means that the normal operation of the core shall result in a measured $F_{\Delta H}^N \leq 1.55/1.0365$ at rated power. The logic behind the larger design uncertainly in this case is that (a) abnormal perturbation in the radial power shape (e.g., rod misalighment) affect $F_{\Delta H}^N$ is in the most cases without necessarily affecting F_q^N through movement of part length rods and can limit it to the desired value, (b) while the operator has some control over F_q^N through F_z^N by motion of control rods, he has no direct control over $F_{\Delta H}^N$, and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

Quadrant power tilts are based upon the following considerations. The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that extended reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The two hour time interval in this specification is considered ample to identify a drop or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the movable detector system as well as thermocouple measurements of the reactor coolant. An additional 22 hours time interval is authorized to accomplish these measurements with full power and other control rod bank configurations. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event the tilt condition cannot be eliminated after 24 hours, the reactor power level will be reduced to a conservative level for physics testing. To avoid reset of a large number of protection set-points, the power range nuclear instrumentation would be reset to cause an automatic reactor scram at 75% of allowed power. A reactor scram at this power has been selected to prevent, with margin, exceeding core safety limits even with a ten percent tilt condition. If a tilt ratio greater than 1.10 occurs which is not due to a misaligned rod, the reactor power shall be further restricted to 50% for 24 hours while corrections are being made. If the correction is not made in 24 hours, critical operation is limited to low power physics testing.

The specified rod drop time is consistent with safety analyses that have been performed.[1]

Part length rod insertion has been limited to eliminate adverse power shapes (Section 3.10.5.2).

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

Analyses indicate the 10% and 3% per hour rates of power increase are slow enough to relax strains which may be induced by axial pellet expansion at the pinched down portion of the clad during power increases following moderate and prolonged periods of operation below 95%.

(Deleted)

References

(1) FSAR, Section 14 and WCAP-8243
(2) FSAR, Section 7.3
(3) WCAP-8243, Section 4.4.2
(4) WCAP-8243, Section 4.4.3

REQUIRED SHUTDOWN
VS. BORON CONCENTRATION
H. B. ROBINSON #2 - CYCLE 3

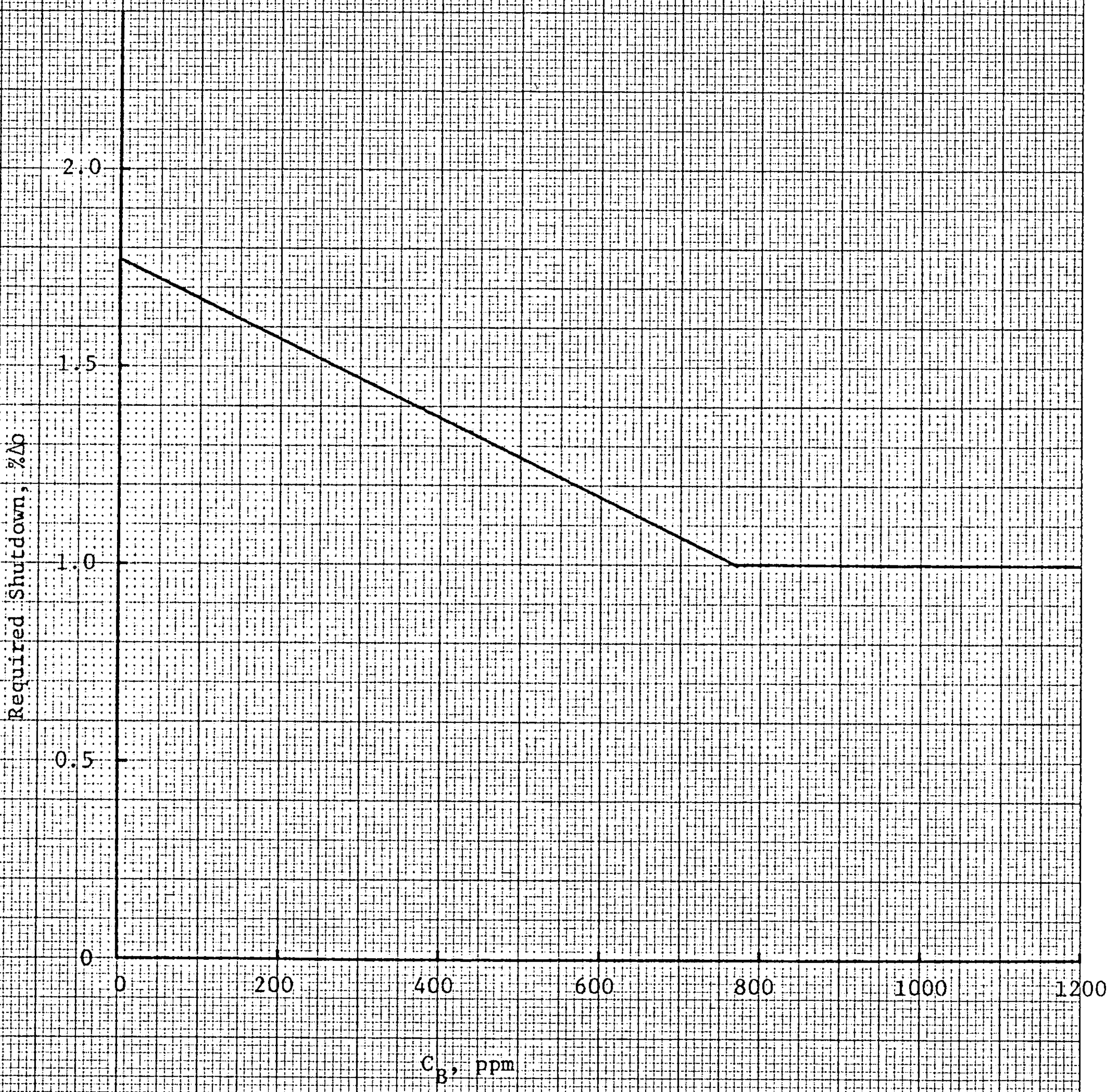


Figure 3.10-2

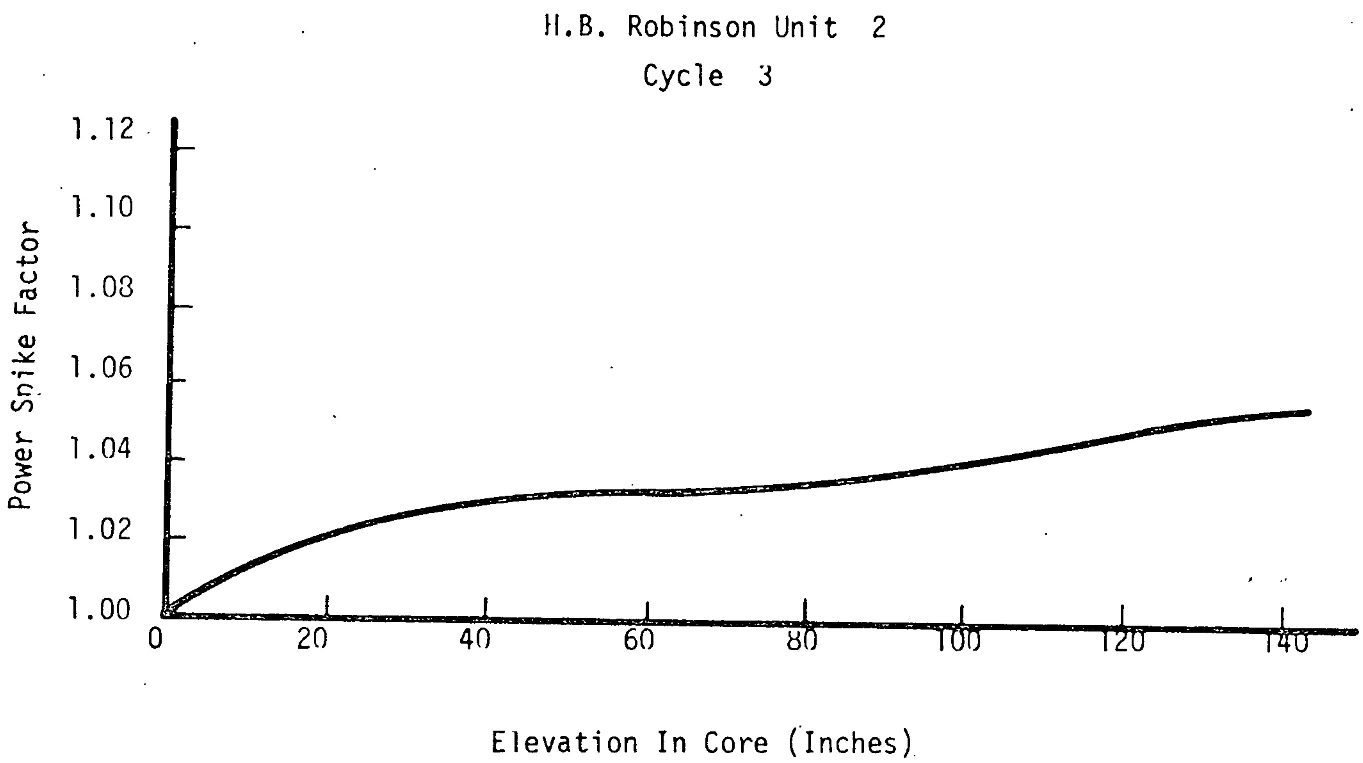


Figure 3.10-3

Power Spike Factor Versus Elevation

3" from the vessel wall at the axial midplace and are spaced radially at 0^{o}, 10^{o}, 20^{o}, 30^{o}, and 40^{o}.[1][2]

To date, Capsule S has been removed and the encapsulated specimens have been tested.[3] Based on the post irradiation test results of Capsule S, the recommended removal schedule for the remaining capsules in the H. B. Robinson Unit No. 2 reactor vessel is as follows:

Capsule Identification	Factor by Which Capsule Leads Vessel Maximum Exposure (1/4T Exposure)	Removal Time
V	0.79 (1.53)	End of 2nd Core Cycle
T	2.4 (4.66)	10 years
U	0.49 (0.95)	20 years
X	0.34 (0.66)	30 years
W	0.34 (0.66)	Standby
Y	0.49 (0.95)	Standby
Z	0.34 (0.66)	Standby

Capsules V, T, and X contain weld metal specimens.

References

(1) FSAR Section 4.4

(2) FSAR Volume 4, Tab VII, Question VI.C

(3) Letter, CPL to AEC, December 20, 1973

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