August 25, 1986

Docket Nos. 50-338
and 50-339

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Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment Nos. 84 and 71 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments revise the Technical Specifications (TS) and the licenses in response to your letters dated May 2, 1985, as supplemented February 6, April 30, June 4, and July 3, 1986.

The amendments revise the NA-1&2 Facility Licenses No. NPF-4 and No. NPF-7 and the Technical Specifications (TS) to increase the presently rated core power level of 2775 Megawatts-thermal (MWt) to 2893 MWt. The amendments allow NA-1&2 to operate at a Nuclear Steam Supply (NSSS) power of 2905 MWt. The amendments will increase the electrical power output for each unit by 32 Megawatts-electrical (MWe).

Amendment No. 84 to Facility Operating License No. NPF-4 for NA-1 is effective within 60 days from the date of issuance. Amendment No. 71 to Facility Operating License No. NPF-7 for NA-2 is effective within 30 days from the date of issuance.

It is noted that the change to the NA-1&2 TS Table 3.3-2 requiring a response time test of the source range, neutron flux trip at least once per 18 months shall be effective prior to restart after the forthcoming sixth (6th) refueling outage for NA-1 and prior to restart after the forthcoming fifth (5th) refueling outage for NA-2.

The Advisory Committee on Reactor Safeguards (ACRS), during its 316th meeting (August 7-9, 1986), concluded that the NA-1&2 power level increase did not warrant review by the ACRS.

Also enclosed is a copy of the related Federal Register notice.

Sincerely,

Leon B. Engle, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosure:
1. Amendment No. 84 to NPF-4
2. Amendment No. 71 to NPF-7
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

8/22

*SEE PREVIOUS CONCURRENCE

LA:PAD#2*	PM:PAD#2*	D:PAD#2*	OGC*	D/DPLA	DD/NRR	D/NRR
DMiller	LEngle	LRubenstein		for TNovak	RVollmer	HDenton for
7/ /86	7/ /86	7/ /86	7/ /86	8/22/86	8/22/86	8/22/86

Also enclosed is a copy of the related Federal Register notice.

Sincerely,

Leon B. Engle, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosure:
1. Amendment No. to NPF-4
2. Amendment No. to NPF-7
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

LA:PAD#2	PM:PAD#2	D:PAD#2	OGC
DMiller	LEngle	LRubenstein	No legal objection
7/31/86	8/5/86	8/6/86	8/7/86

RSB
J.Wilson
8/2/86

Mr. W. L. Stewart
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated May 2, 1985, as supplemented February 6, April 30, June 4, July 3, and August 20, 1986, 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 ... the application, the provisions of the Act, and the rule ... regulations of the Commission;

   C. There is reasonable assurance (i) that t... tivities authorized by this amendment can be conducted witho... langering the health and safety of the public, and (ii) that s... ctivities will be conducted in compliance with the Commissi... regulations;

   D. The issuance of this amendment will not be ... ical to the common defense and security or to the health and ... of the public; and

   E. The issuance of this amendment is in accord... with 10 CFR Part 51 of the Commission's regulations and all a... able requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 84 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Paragraph 2.D.1 to Facility Operating license No. NPF-4 is hereby revised to read:

   2.D.1 Maximum Power Level

   (a) VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core levels not in excess of 2893 megawatts (thermal).

4. This license amendment is effective within 60 days from its date of issuance with the exception that the change to Table 3.3-2 requiring a response time test of the source range, neutron flux trip at least once per 18 months shall be effective prior to restart after the forthcoming sixth (6th) refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 25, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 84

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

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

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QUADRANT POWER TILT RATIO

1.23 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

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 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.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.27 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.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a solid form that meets shipping and burial ground requirements.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.31 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY where access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, 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 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

Nominal T_{avg} = 586.8°F
Nominal RCS flow = 289200 GPM

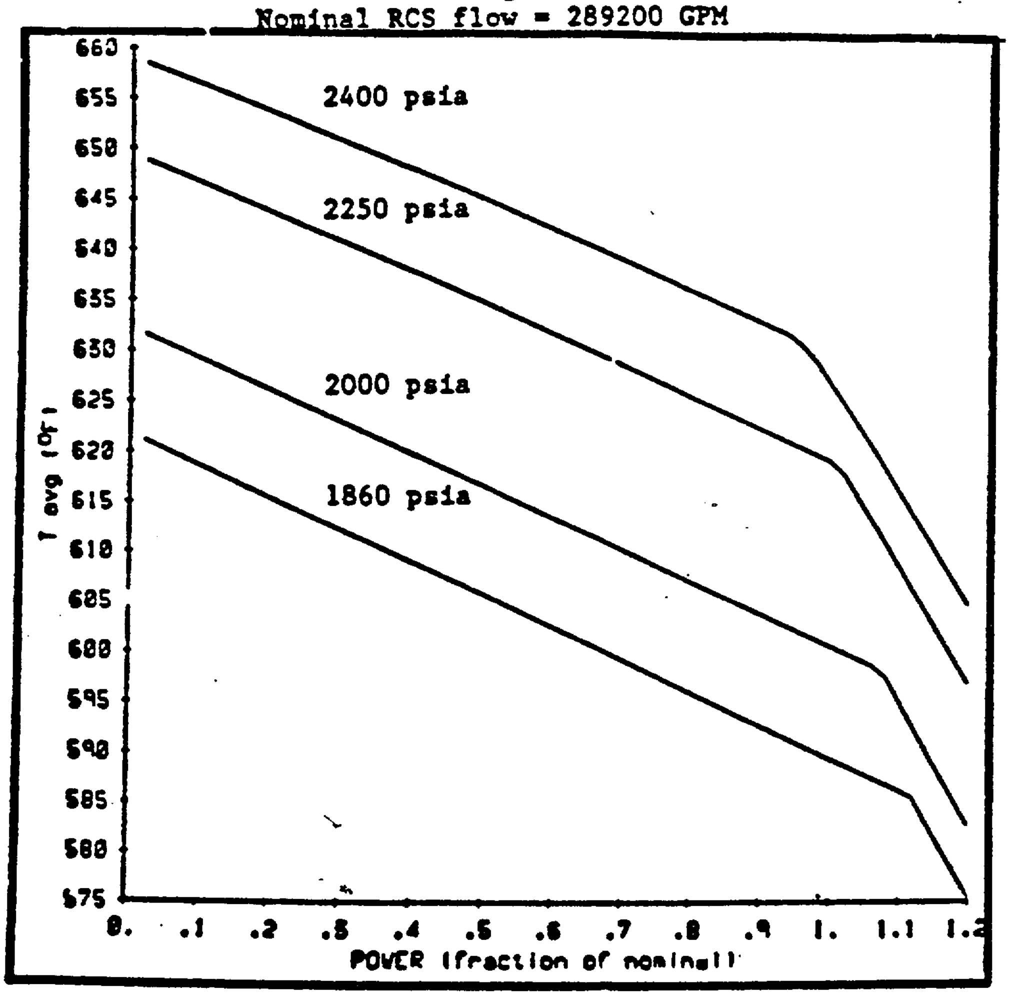

Figure 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1870 psig	$\geq$ 1860 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 96,400 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pump Busses	≥ 2905 volts-each bus	≥ 2870 volts-each bus
16. Underfrequency-Reactor Coolant Pump Busses	≥ 56.1 Hz - each bus	≥ 56.0 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 40 psig
B. Turbine Stop Valve Closure	≥ 1% open	≥ 0% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

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

where:

ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, ^{o}F
T'	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 586.8^{o}F
P	=	Pressurizer pressure, psig
P'	=	2235 psig (indicated RCS nominal operating pressure)
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1 & τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.
S	=	Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.264	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 = ( )	K_3 = ( )

and f_1 (ΔI) 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:

(i) for $q_t - q_b$ between - 44 percent and + 3 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 44 percent the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T') - f_2(\Delta I)\right]$

Where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, ^oF

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 586.8^oF

K_4 = 1.079

K_5 = 0.02/^oF for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00164 for $T > T'$; K_6 = 0 for $T \leq T'$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 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, DNBR, 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 DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figures 2.1-1, 2.1-2, and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1+ 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Source and Intermediate Range, Nuclear Flux trips provide reactor core protection during shutdown (Modes 3, 4 and 5) when the reactor trip system breakers are in the closed position. The Source and Intermediate Range trips in addition to the Power Range trips provide core protection during reactor startup (Mode 2). Reactor startup is prohibited unless the Source, Intermediate and Power Range trips are operable in accordance with Specification 3.3.1.1. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. In the accident analyses, bounding transient results are based on reactivity excursions from an initially critical condition, where the source range trip is assumed to be blocked. Accidents initiated from a subcritical condition would produce less severe results since the source range trip would provide core protection at a lower power level. No credit was taken for operation of the trip associated with the Intermediate Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transient delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

BASES

Operation with a reactor coolant loop out of service below the 3 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 2 loop operation exclusive of the Overtemperature ΔT set point. Two loop operation above the 3 loop P-8 set point is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature ΔT channels and raising the P-8 set point to its 2 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure. The low pressure trip is blocked below P-7.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. The pressurizer high water level trip is blocked automatically below the P-7 setpoint.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 31% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design limit during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 71% RATED THERMAL POWER with the loop stop valves closed in the nonoperating loop, will prevent the minimum value of the DNBR from going below the design limit during normal operational transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system. The steam generator water level low-low trip is blocked when the loop stop valves are closed. A steam generator water level high-high signal trips the turbine which in turn trips the reactor if above the P-7 setpoint.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.15}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.30] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f, below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP}\ [1 + 0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

SURVEILLANCE REQUIREMENTS (Continued)

(b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for Rated Thermal Power (F_{xy}RTP) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes, in a Core Surveillance Report per Technical Specification 6.9.1.7.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ±2%, 32.1 ±2%, 46.4±2%, 60.6±2% and 74.9±2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

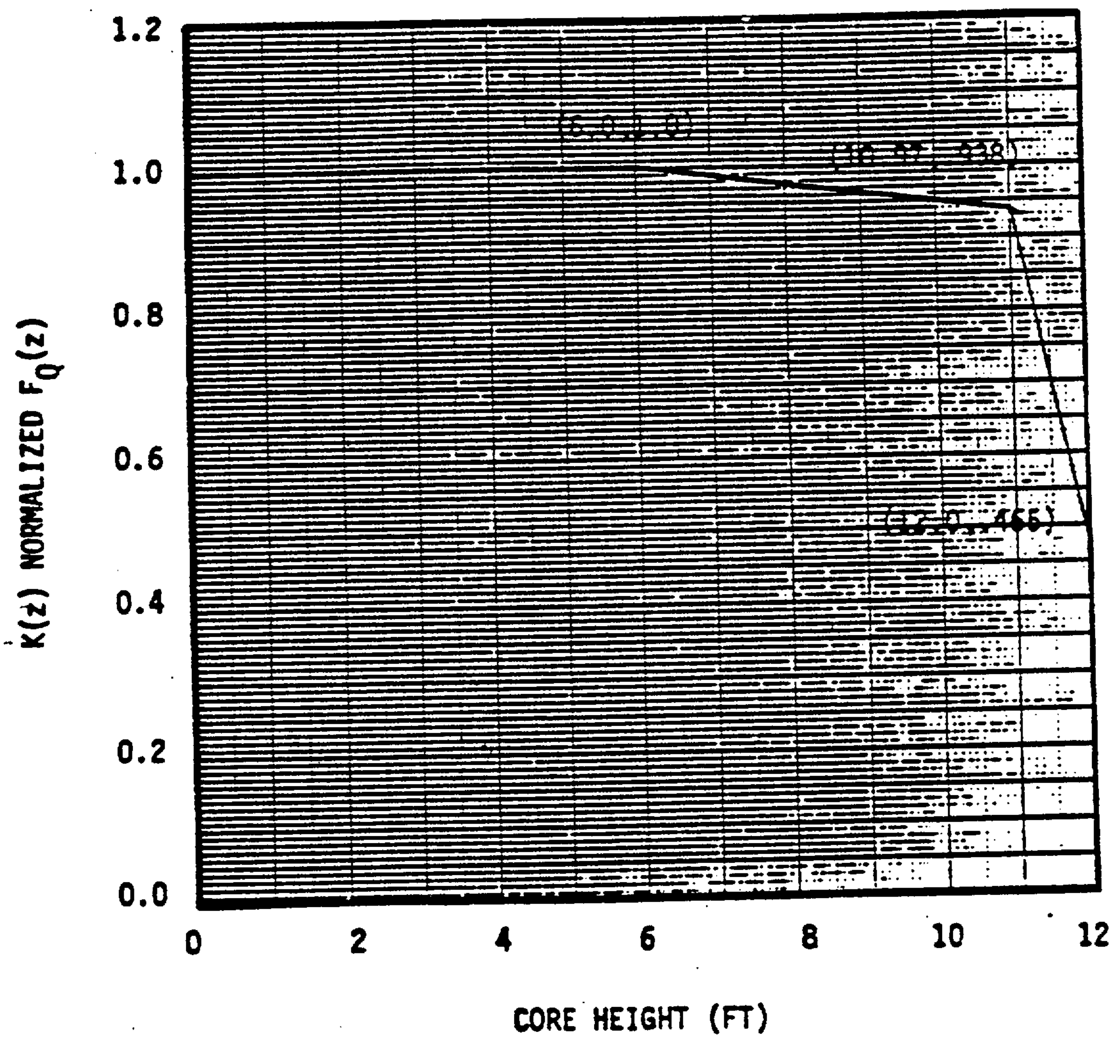

Figure 3.2-2 NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.49\ [1 + 0.3\ (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\gtrsim$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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.

c. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops in Operation	LIMITS: 2 Loops in Operation** & Loop Stop Valves Open	LIMITS: 2 Loops in Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 591^{0}F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 289,200 gpm		

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

**Values dependent on NRC approval of ECCS evaluation for these conditions.

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.15]\ [K(Z)]}{(\bar{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\bar{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m% of RATED THERMAL POWER in accordance with:

$$\bar{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceeding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3.1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##	4
B. Shutdown	2	1	2	3*,4* and 5*	15
C. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	7#
Two Loop Operation	3	1**	2	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and 2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8% 8%	>7% >7%	Prevents reactor trip on: Low flow or reactor coolant pump breakers open in more than one loop, Undervoltage (RCP busses), Underfrequency (RCP busses), Turbine Trip, Pressurizer low pressure, and Pressurizer high level.
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Permit reactor trip on low flow or reactor coolant pump breaker open in a single loop.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Blocks reactor trip on low flow or reactor coolant pump breaker open in a single loop.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	≤ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	≤ 0.5 seconds*
7. Overtemperature ΔT	≤ 4.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response of the neutron flux signal portion of the channel time shall be measured from detector output or input of first electronic component in channel.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3, 4	14^{*}
d. Pressurizer Pressure-Low-Low	3	2	2	1, 2, $3^{\#}$	14^{*}
e. Differential Pressure Between Steam Lines - High				1, 2, $3^{\#\#}$	
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		14^{*}
Two Loops Operating	3/operating steam line	$2^{\#\#\#}$/steam line twice in either operating steam line	2/operating steam line		15
f. Steam Flow in Two Steam Lines-High				1, 2, $3^{\#\#}$	

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	1, 2, 3, 4
d. Pressurizer Pressure--Low-Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M(3)	1, 2, 3
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--Intermediate High-High	S	R	M	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low-Low or Steam Line Pressure--Low	S	R	M	1, 2, 3

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core from going beyond the design limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

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

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to the average power density in the horizontal plane at core elevation Z.

3/4 2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and P_f% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than P_f% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and P_f% and 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

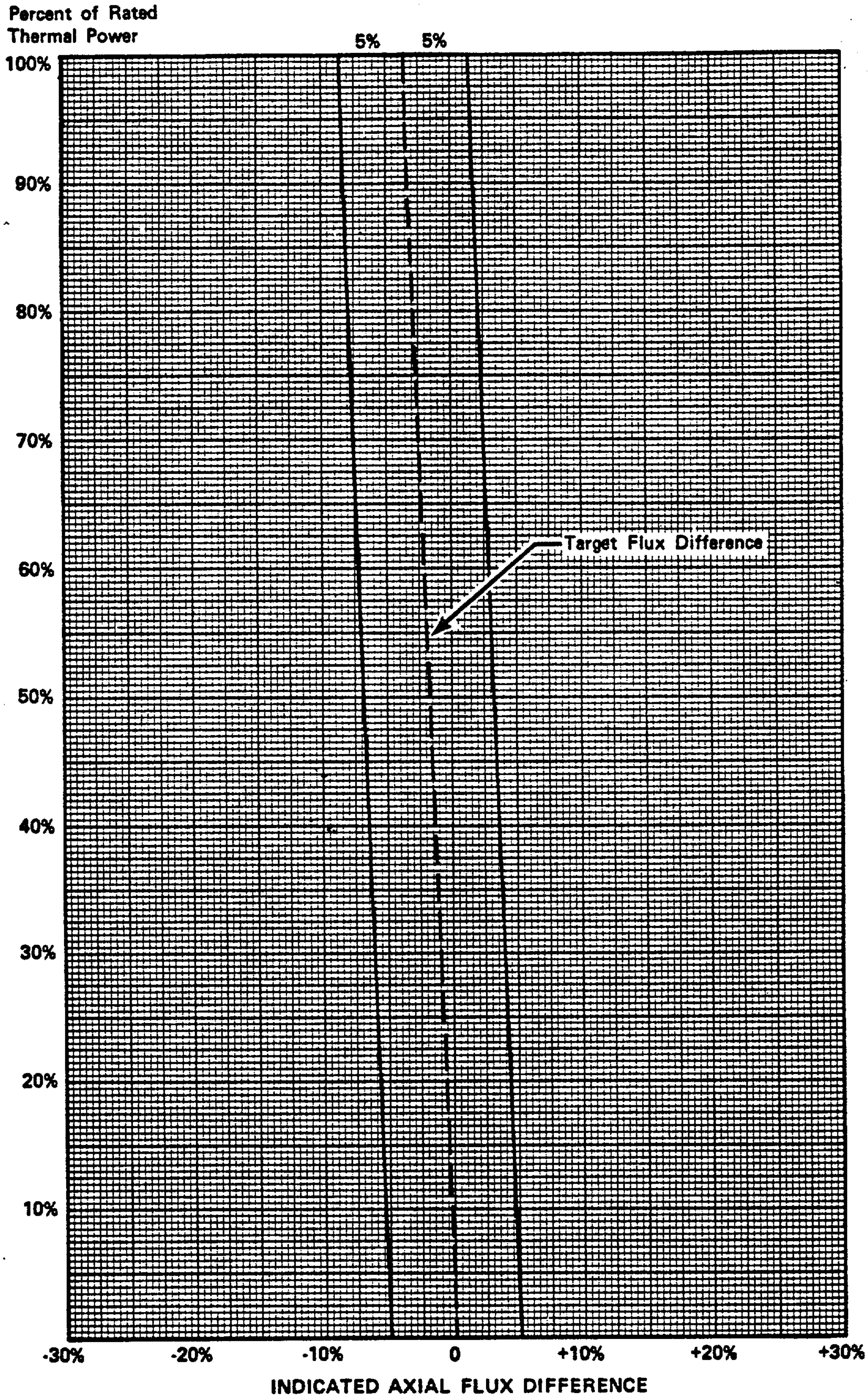

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H} \leq 1.49$. The 4% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the margin available between the safety analysis design DNBR values (1.57 and 1.59 for thimble and typical cells, respectively) and the limiting design DNBR values (1.39 for thimble cells and 1.42 for typical cells). The applicable value of rod bow penalties can be obtained from the FSAR.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. The two sets of 4 symmetric thimbles is a unique set of 8 detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties must be accounted for during the periodic surveillance.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above P_m% of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA. The value for P_m is based on the cycle dependent potential violation of the F_QxK(Z) limit, where K(Z) is the graph shown in Figure 3.2-2. The amount of potential violation is determined by subtracting 1 from the maximum ratio of the predicted F_Q(Z) analysis (flyspeck) results for a particular fuel cycle to the F_QxK(Z) limit. This amount of potential violation, in percent, is subtracted from 100% to determine the value for P_m. If P_m is equal to 100%, no axial power distribution surveillance is required. P_m will not exceed 100%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

   a) Centrifugal charging pump, and

   b) Low head safety injection pump.

f. By verifying that each of the following pumps develop the indicated discharge pressure (after subtracting suction pressure) on recirculation flow when tested pursuant to Specification 4.0.5.

   1. Centrifugal charging pump $\geq$ 2410 psig.

   2. Low head safety injection pump $\geq$ 156 psig

g. By verifying that the following manual valves requiring adjustment to prevent pump "runout" and subsequent component damage are locked and tagged in the proper position for injection:

   1. Within 4 hours following completion of any repositioning or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

1.	1-SI-188	Loop A Cold Leg
2.	1-SI-191	Loop B Cold Leg
3.	1-SI-193	Loop C Cold Leg
4.	1-SI-203	Loop A Hot Leg
5.	1-SI-204	Loop B Hot Leg
6.	1-SI-205	Loop C Hot Leg

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

   1. For high head safety injection lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is $\geq$ 384 gpm, and

      b) The total pump flow rate is $\leq$ 650 gpm.

ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of automatically transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank or from the containment sump when the suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of the system design pressure, during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.83×10^6 lbs/hr which is greater than the total secondary steam flow of 12.77×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 3 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

For 2 loop operation with stop valves closed

$$SP = \frac{(X) - (Y)(U)}{X} \times 71$$

For 2 loop operations with stop valves open

$$SP = \frac{(X) - (Y)(U)}{X} \times 66$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation

71 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation with stop valves closed.

66 = Maximum percent of RATED THERMAL POWER permissible by P-8 setpoint for 2 loop operation with stop valves open.

X = Total relieving capacity of all safety valves per steam line in lbs/hour = 4,275,420

Y = Maximum relieving capacity of any one safety valve in lbs/hour = 855,084

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 340 gpm at a pressure of 1064 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1064 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated May 2, 1985, as supplemented February 6, April 30, June 4, July 3, and August 20, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 71 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Paragraph 2.C.(1) to Facility Operating license No. NPF-7 is hereby revised to read:

   2.C.(1) Maximum Power Level

   VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2893 megawatts (thermal).

4. This license amendment is effective within 30 days from its date of issuance with the exception that the change to Table 3.3-2 requiring a response time test of the source range, neutron flux trip at least once per 18 months shall be effective prior to restart after the forthcoming fifth (5th) refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 25, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 71

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

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

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QUADRANT POWER TILT RATIO

1.23 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

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 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.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.27 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.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a solid form that meets shipping and burial ground requirements.

SOURCE CHECK

1.30 A SOURCE CHECK shall be tha qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.31 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY where access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, 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 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

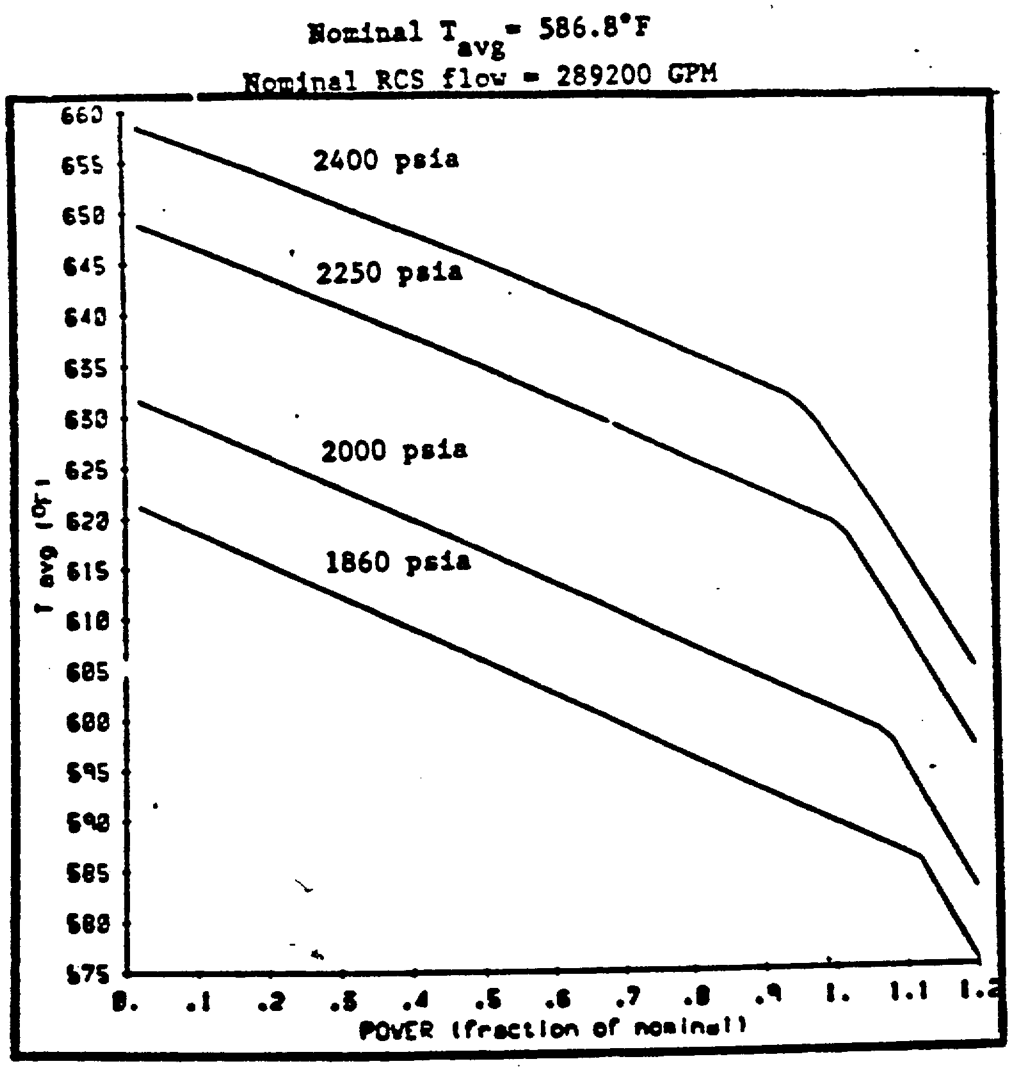

Figure 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds.
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1870 psig	$\geq$ 1860 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 96,400 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pump Busses	≥ 2905 volts-each bus	≥ 2870 volts-each bus
16. Underfrequency-Reactor Coolant Pump Busses	≥ 56.1 Hz - each bus	≥ 56.0 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 40 psig
B. Turbine Stop Valve Closure	≥ 1% open	≥ 0% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T - T') + K_3(P - P') - f_1(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 586.8°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.264	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 + ( )	K_3 = ( )

and f_1 (ΔI) 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:

(i) for $q_t - q_b$ between - 44 percent and + 3 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 44 percent, the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_0 [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S} \right) T - K_6 (T - T'') - f_2(\Delta I)]$

where: ΔT_0 = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T'' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 586.8°F.

K_4 = 1.079

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00164 for $T > T''$; $K_6 = 0$ for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 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, DNBR, 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 DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figures 2.1-1, 2.1-2, and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1+0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Source and Intermediate Range, Nuclear Flux trips provide reactor core protection during shutdown (Modes 3, 4, and 5) when the reactor trip system breakers are in the closed position. The Source and Intermediate Range trips in addition to the Power Range trips provide core protection during reactor startup (Mode 2). Reactor startup is prohibited unless the Source, Intermediate and Power Range trips are operable in accordance with Specification 3.3.1.1. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. In the accident analyses, bounding transient results are based on reactivity excursions from an initially critical condition, where the source range trip is assumed to be blocked. Accidents initiated from a subcritical condition would produce less severe results since the source range trip would provide core protection at a lower power level. No credit was taken for operation of the trip associated with the Intermediate Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that that transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 3 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 2 loop operation exclusive of the Overtemperature Delta T setpoint. Two loop operation above the 3 loop P-8 setpoint is permissible after resetting the K1, K2, and K3 inputs to the Overtemperature Delta T channels and raising the P-8 setpoint to its 2 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure. The low pressure trip is blocked below P-7.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. The pressurizer high water level trip is blocked automatically below the P-7 setpoint.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent (P-7) of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above 31% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow.

This latter trip will prevent the minimum value of the DNBR from going below the design limit during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 71% RATED THERMAL POWER with the loop stop valves closed in the nonoperating loop, will prevent the minimum value of the DNBR from going below the design limit during normal operational transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level low-low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system. The steam generator water level low-low trip is blocked when the loop stop valves are closed. A steam generator water level high-high signal trips the turbine which in turn trips the reactor if above the P-7 setpoint.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip setting and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than 1.616×10^6 lbs/hour of full steam flow at RATED THERMAL POWER. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.15}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.30] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.15}{P \times N(z)} \times K(z) \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.15}{N(z) \times 0.5} \times K(z) \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.15 is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Core Surveillance Report as per Specification 6.9.1.7.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(z)$ was last determined by 10% or more of RATED THERMAL POWER*, or

2. At least once per 31 effective full power days, whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

*During power escalation, the power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

1. $F_Q^M(z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the realtionships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by substracting one from the measurement/limit ratio and multiplying by 100:

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{2.15}{P} \times N(z)} \times K(z) \right) -1 \right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{2.15}{0.5} \times N(z)} \times K(z) \right) -1 \right\} \times 100 \quad \text{for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Power operation may continue provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 15 percent inclusive.

2. Upper core region 85 to 100 percent inclusive.

4.2.2.3 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

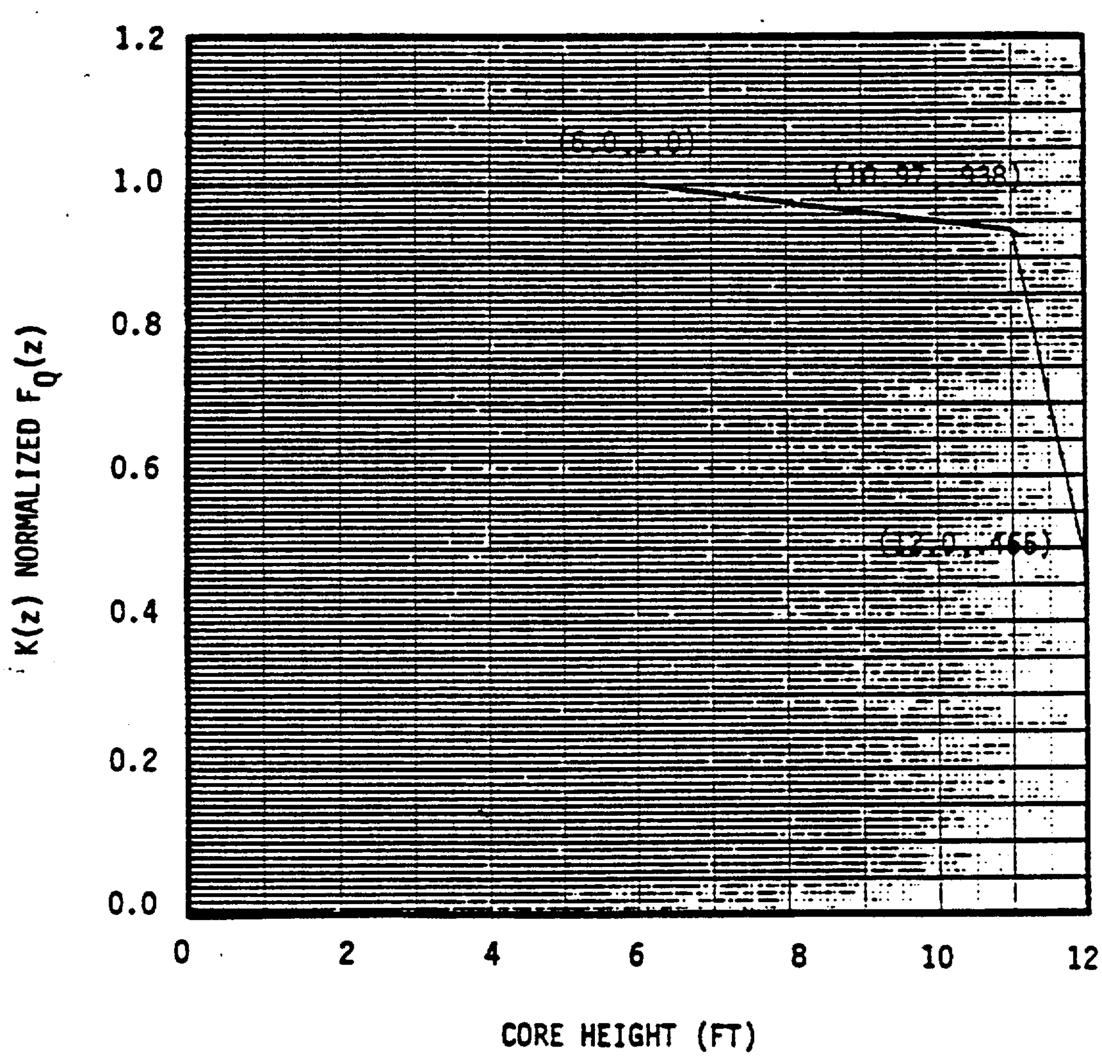

Figure 3.2-2 NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.49\ [1 + 0.3\ (1-P)]$$

where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

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

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to

ACTION Continued

exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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.

c. The provisions of Specification 4.0.4 are not applicable.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops in Operation	2 Loops in Operation** & Loop Stop Valves Open	2 Loops in Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq 591°F$		
Pressurizer Pressure	≥ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq 289,200$ gpm		

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

**Values dependent on NRC approval of ECCS evaluation for these conditions.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceeding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3.1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##	4
B. Shutdown	2	1	2	3*, 4* and 5*	15
C. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	7#
Two Loop Operation	3	1**	2	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and 2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8% 8%	>7% >7%	Prevents reactor trip on: Low flow or reactor coolant pump breakers open in more than one loop, Undervoltage (RCP busses), Underfrequency (RCP busses), Turbine Trip, Pressurizer low pressure, and Pressurizer high level.
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Permit reactor trip on low flow or reactor coolant pump breaker open in a single loop.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Blocks reactor trip on low flow or reactor coolant pump breaker open in a single loop.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	$\leq$ 0.5 seconds*
7. Overtemperature ΔT	$\leq$ 4.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response of the neutron flux signal portion of the channel time shall be measured from detector output or input of first electronic component in channel.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3, 4	14*
d. Pressurizer Pressure -- Low-Low	3	2	2	1, 2, 3#	14*
e. Differential Pressure Between Steam Lines - High				1, 2, 3##	
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		14*
Two Loops Operating	3/operating steam line	2###/steam line twice in either operating steam line	2/operating steam line		15

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M(3)	1, 2, 3

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	1, 2, 3, 4
d. Pressurizer Pressure--Low-Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3

f. By verifying that each of the following pumps develop the indicated discharge pressure (after subtracting suction pressure) on recirculation flow when tested pursuant to Specification 4.0.5.

   1. Centrifugal charging pump greater than or equal to 2410 psig.

   2. Low head safety injection pump greater than or equal to 156 psig

g. By verifying that the following manual valves requiring adjustment to prevent pump "runout" and subsequent component damage are locked and tagged in the proper position for injection:

   1. Within 4 hours following completion of any repositioning or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

1.	2-SI-89	Loop A Cold Leg
2.	2-SI-97	Loop B Cold Leg
3.	2-SI-103	Loop C Cold Leg
4.	2-SI-116	Loop A Hot Leg
5.	2-SI-111	Loop B Hot Leg
6.	2-SI-123	Loop C Hot Leg

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

   1. For high head safety injection lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is $\geq$ 384 gpm, and

      b) The total pump flow rate is $\leq$ 650 gpm.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} less than 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of automatically transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank or from the containment sump when the suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350° by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 340°F.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core from going beyond the design limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

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

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI-power operating space and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

BASES

When $F^N_{\Delta H}$ is measured, 4% is the appropriate experimental error allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H}$ less than or equal to 1.49. The 4% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the margin available between the safety analysis design DNBR values (1.57 and 1.59 for thimble and typical cells, respectively) and the limiting design DNBR values (1.39 for thimble cells and 1.42 for typical cells). The applicable value of rod bow penalties can be obtained from the FSAR.

The hot channel factor $F_Q M(Z)$ is measured periodically and increased by a cycle and height dependent power factor, N(Z), to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. N(Z) accounts for the non-equilibrium effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The N(Z) function for normal operation is provided in the Core Surveillance Report per Specification 6.9.1.7.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. The two sets of 4 symmetric thimbles is a unique set of 8 detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties must be accounted for during the periodic surveillance.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of the system design pressure, during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.83×10^6 lbs/hr which is greater than the total secondary steam flow of 12.77×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip-setpoint reductions are derived on the following bases:

For 3 loop operation

$$SP = \frac{(X) \qquad (Y)(V)}{X} \times 109$$

For 2 loop operation with stop valves closed

$$SP = \frac{(X) \qquad (Y)(U)}{X} \times 71$$

For 2 loop operations with
stop valves open

$$SP = \frac{(X) - (Y)(U)}{X} \times 66$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation

71 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation with stop valves closed.

66 = Maximum percent of RATED THERMAL POWER permissible by P-8 setpoint for 2 loop operation with stop valves open.

X = Total relieving capacity of all safety valves per steam line in lbs/hour = 4,275,420

Y = Maximum relieving capacity of any one safety valve in lbs/hour = 855,084

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 84 AND 71 TO

FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-338 AND 50-339

Introduction

By letters dated May 2, 1985, as supplemented February 6, April 30, June 4, July 3, and August 20, 1986, the Virginia Electric and Power Company (the licensee) requested amendments to Facility Operating Licenses No. NPF-4 and No. NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). Specifically, the amendments would revise the NA-1&2 Technical Specifications (TS) to increase the presently rated core power level of 2775 Megawatts thermal (MWt) to 2893 MWt in accordance with the licensee's initial application for amendments dated May 2, 1985. The proposed changes would allow NA-1&2 to operate at a Nuclear Steam Supply System (NSSS) power of 2905 MWt.

Discussion

The proposed changes would represent an approximate 4.5 percent increase over the presently licensed core power rating of 2775 MWt. The proposed changes would increase the electrical power output for each unit by 32 Megawatts-electrical (MWe). It is noted that the NA Updated Final Safety Analysis Report (UFSAR) analyses and accident evaluations were evaluated at the presently Engineered Safeguards Design Stretch Core Power Rating of 2900 MWt and a NSSS power rating of 2910 MWt. In order to maintain a consistent basis between information reported in the instant proposed amendment request and that reported by reference in the UFSAR, the licensee's evaluation of NSSS capability has also been performed at 2910 MWt. However, because of turbine generator limitations, the licensee's proposed request would allow operation at a NSSS rating of only 2905 MWt and a core rating of 2893 MWt.

The scope of the licensee's review to support the proposed core uprate encompassed all aspects of the NA-1&2 NSSS design and operation affected by the increase. NSSS designs were reviewed to verify compliance at the increased power rating with licensing criteria and standards currently specified in the

NA-1&2 operating licenses. In addition, a review was conducted by the licensee to identify any potential unreviewed safety questions that might occur as a result of the increased power rating in accordance with 10 CFR Part 50.59. The structural design of NSSS equipment was reviewed to assure that compliance had been maintained at the increased power rating with industry codes and standards that applied when the equipment was originally built. In addition, the review encompassed the verification that NSSS components and systems will continue to meet functional requirements specified in the FSAR at the increased power rating. Currently approved NRC analytical techniques were used for analyses performed at the increased power rating. Also, the definition of NSSS/Balance of Plant (BOP) safety related interfaces were reviewed for any impact at the increase in power rating. Based on the scope of review as outlined above, the licensee states that NA-1&2 are capable, in their present design configuration, of operating at the proposed core power rating of 2893 MWt and an NSSS power rating of 2905 MWt without violating any of the design criteria or safety limits specified in the NA-1&2 FSAR and as currently required in Facility Operating Licenses NPF-4 and NPF-7 for NA-1&2, respectively.

In addition to evaluating the ability of the plant to perform at the new power level under steady state conditions, the licensee reevaluated all the design basis transients and accidents which the NRC staff utilizes to determine that adequate safety margins are maintained. These analyses were performed by Westinghouse using the FACTRAN, LOFTRAN, TWINKLE and THINK computer codes which have been previously reviewed and approved by the NRC staff. Those events which might challenge the core Departure from Nucleate Boiling Ratio (DNBR) limits were evaluated using the Westinghouse Improved Thermal Design Procedure. Steady state instrument errors were considered in establishing the initial conditions, including the addition of 2 percent to the initial power to account for calorimetric error. The effect of the positive moderator reactivity coefficient was considered for those events which would produce limiting consequences at the beginning of a core cycle at less than or equal to 70 percent of power and as specified in the NA-1&2 TS. The analyses included reduction in core power peaking factors at the increased power level. These new peaking factor limits are included in the TS changes provided by the licensee as part of the power upgrade request for NA-1&2.

Excessive Heat Removal Events

The licensee evaluated excessive cooling by feedwater such as might occur from malfunction of the feedwater system controller or failure of feedwater heating. In addition, overcooling from excessive load increases was evaluated. The design margin to DNB was determined to be maintained throughout these events.

The licensee did not reanalyze the consequences from accidental steam system depressurization, including steam line breaks. The limiting conditions for these events are low power when the steam generator water inventory is greatest and end-of-core life when the moderator temperature is most negative. These conditions have previously been evaluated in the NA-1&2 FSAR.

Decrease in Heat Removal

Loss of the heat sink provided by the steam generators would cause the temperature and pressure to increase within the reactor system. If the reactivity coefficient of the moderator were positive, the power would also increase until reactor trip occurred. The licensee analyzed the loss of external electric load as a complete and instantaneous loss of steam flow to the turbine. Pressure control systems were assumed inoperable. The results were determined to be relatively insensitive to moderator temperature coefficient. Following reactor trip on high pressure, the pressurizer pressure reached 2554 pounds per square inch absolute (psia) with the most positive coefficient expected during plant operation, and 2538 psia with the most negative expected coefficient. The maximum pressure within the reactor system would be somewhat higher than the above calculated pressurizer pressures as the result of gravity and coolant pump heads. These are sufficiently small so that the maximum pressure would remain below the 110 percent design limit of 2750 pounds per square inch gage (psig). The licensee did not reanalyze loss of feedwater or feedwater line break events. These events already have been analyzed in the FSAR at the proposed power rating of 2910 MWt.

Loss of Forced Reactor Coolant Flow

An excessive reduction in the reactor coolant flow at power would cause the reactor system pressure and temperature to increase. If the moderator temperature coefficient were positive, the reactor power would also increase until the reactor tripped. The licensee calculated the consequences from single and multiple trips of the reactor coolant pumps and concluded that reactor system pressure and core DNBR margin would remain within the design limits.

In the unlikely event that a reactor coolant pump shaft were to fail, either from shear or seizure, the flow contribution from one coolant loop would be rapidly lost. This would be approximately one third of the flow for the three loop NA plants. The staff has not accepted arguments that fuel rods which experience DNB for brief periods at high pressure will not fail. Therefore, those fuel elements which fell below the DNBR design limit were assumed to experience cladding damage and to release the fission products in the gap between the fuel and the cladding. This assumption affected 13 percent of the core.

The licensee calculated the offsite dose consequences from this event assuming failure of a steam generator relief valve to close as a single failure. The steam generator relief valves would be opened following the reactor trip/turbine trip produced by a low reactor coolant flow signal. The steam dump to the condenser was not assumed to be operable since offsite power was assumed to be lost in accordance with General Design Criteria 17. All steam release was assumed to be directed to the atmosphere. It was further assumed that the operator isolated auxiliary feedwater from the affected steam generator in accordance with procedures allowing the tubes to become uncovered.

With the steam generator tubes uncovered in the affected steam generator, a direct path would exist for reactor coolant to escape from steam generator tube leakage directly to the atmosphere. The licensee calculated the offsite dose consequences from this event assuming that the plant operators manually isolated the stuck open relief valve in 15 minutes. The offsite dose consequences were approximately 10% of the 10 CFR 100 guideline. The staff concludes this evaluation is acceptable in view of the conservatisms existing in the offsite dose calculations.

Reactivity Insertion Events

The licensee evaluated the consequences of inadvertent control rod withdrawal from subcritical and from various power levels. The effect of a positive moderator coefficient was included. The reactor was calculated to trip on high differential temperature for slow reactivity insertion rates and on high nuclear power for rapid reactivity insertion rates. For initial power levels below 25 percent the reactor was assumed to trip at the maximum value of the low level power range trip at 35 percent nuclear power. The minimum DNBR in the core was calculated to remain above the design limit.

The staff questioned the use of the low range power trip for analysis of control rod withdrawal from subcritical since the TS do not require this trip to be operable when the reactor is subcritical. In addition, the analysis assumed the reactor coolant pumps to be operating, whereas the TS do not require the reactor coolant pumps to operate when the reactor is subcritical and coolant is below 350°F. The licensee responded by proposing the NA-1&2 TS require that redundant source range channels be available to trip the reactor in the event of inadvertent control rod withdrawal while subcritical. The source range channels are set to trip the reactor when the neutron level exceeds 10^5 counts/sec which would occur before significant power could be generated in the core. The source range channels are not seismically qualified. The staff concludes that this is acceptable since the simultaneous occurrence of an earthquake and a control rod withdrawal event, which would last only a few seconds, would be unlikely.

The licensee reanalyzed the consequences of postulated control rod ejection accidents at the beginning and end-of-life. The results were not significantly different from those of the FSAR and remain below the staff's acceptance criteria for maximum fuel sensible heat and percent fuel melting. Boron dilution events were not reanalyzed since the course of these events would not be affected by the power upgrade.

Loss of Coolant Accidents

The licensee reanalyzed the consequences from postulated Loss of Coolant Accidents (LOCA) at the upgraded power level. Large break LOCAs were analyzed using the NRC-approved 1981 Westinghouse ECCS Evaluation Model with the BART computer code and the corrections and additions set forth in WCAP-9561, Addendum 3, Revision 1.

Following a postulated double ended rupture of a reactor cold leg, the fuel cladding temperature was determined to reach a peak temperature of 2152°F before being quenched by incoming water from the Emergency Core Cooling System (ECCS). The staff finds this peak temperature of 2152°F to be acceptable.

The consequences from small break LOCA were reanalyzed by the licensee at the higher power level using an approved Westinghouse small break evaluation model. This model utilizes the WFLASH code to calculate the reactor system thermal/hydraulic response and the LOCTA-IV code to calculate the cladding heatup. The licensee determined that the maximum fuel cladding temperature would occur for a break in a cold leg with an equivalent diameter of 3 inches. The maximum cladding temperature was determined to be 1749°F, which is acceptable.

The staff raised questions regarding the consequences from loss of coolant accidents during hot standby and shutdown. During these modes of operation the heat source within the reactor core would be less than immediately following reactor trip, but a portion of the ECCS pumps and actuation circuits are bypassed. The licensee responded by proposing that the NA-1&2 TS requirements be augmented for automatic ECCS actuation on high containment pressure to Mode 4 (hot shutdown) and by requiring automatic alignment of the ECCS pump suction to the refueling water storage tank for ECCS actuation while in Mode 4. The licensee evaluated the consequences from large break LOCA during hot standby and hot shutdown with the accumulators isolated in accordance with procedures. The licensee concluded that flow from the ECCS pumps would be adequate to recover the core and the peak cladding temperature would be less than 2200°F, which is acceptable.

For small break LOCA the ECCS might not be automatically actuated on high containment pressure and operator action would be required to initiate the emergency cooling systems. The licensee provided the results from a small break LOCA analysis which indicated that the operator would have more than 22 minutes to manually actuate safety injection. The operator would have adequate indication of small break LOCA from high radiation alarms in the containment dome and sump. In addition, indication of core uncovery is provided by the core exist thermocouples and the Reactor Vessel Level Indicating System. The staff finds these measures to be acceptable.

Technical Specification Changes

To support the power upgrade request, the licensee provided proposed TS changes. These changes increase the maximum allowable power output, revise safety limits, trip setpoints, and reduce the allowable peaking factors to reflect the revised safety analyses. The appropriate basis sections would also be revised. The staff concludes the changes which are discussed below are acceptable. The changes are discussed by page number(s) for Unit 1, with the corresponding Unit 2 page number(s) in parentheses.

a. Page 1-5, (1-5) of Facility Operating Licenses NPF-4 and NPF-7 for NA-1&2, respectively. The changes on these pages only replace the existing maximum reactor power level with the uprated maximum reactor core power level of 2893 MWt.

b. Page 2-2, (2-2). This change appropriately adjusts the figure defining the reactor core safety limit (maximum allowable temperature as a function of pressure and power) for uprated power, and nominal design flow of 289,200 gallons per minute (gpm).

c. Page 2-6, (2-6). The change on this page corrects the footnote for the new loop design flow.

d. Page 2-8, (2-8).. The change on this page corrects the indicated T_{avg} to 586.8°F for uprated power conditions.

e. Page 2-9, (2-9). The changes on this page adjust K_1, K_2 and K_3 and the wings of the $F_{\Delta I}$ function for the uprated power level.

f. Page 2-10, (2-10). The changes on this page adjust the indicated T_{avg} to 586.8°F and K_4 and K_6 for the uprated power level.

g. Pages B2-1, B2-2, B2-4, B2-6, (B2-1, B2-2, B2-3, B2-4, B2-6). The changes on these pages appropriately correct the Technical Specification Bases to reflect the power uprate, use of the WRB-1 DNB correlation and the other changes discussed above.

h. Page 3/4 2-5 (3/4 2-5). The change on this page adjusts the maximum allowable F_Q to 2.15 in the 50% to 100% power range and to 4.30 at lower power levels, as assumed as an initial condition in the LOCA analysis for uprated power.

i. Pages (3/4 2-6, 3/4 2-7). The changes on these pages adjust the maximum allowable $F_Q(z)$ value to 2.15.

j. Page 3/4 2-8, (3/4 2-8). This change provides a new normalized K(z) curve reflecting the change in $F_Q(z)$ in the previous two items.

k. Page 3/4 2-9, (3/4 2-9). The changes on this page adjust the maximum allowable full power $F^N_{\Delta H}$ to 1.49 in keeping with the use of the improved thermal hydraulic correlation of the uprated core analysis, and removes language concerning $F^N_{\Delta H}$ and flow measurement uncertainty which is no longer needed for the reanalyzed core.

l. Page 3/4 2-10, (3/4 2-10). This change removes a note concerning measurement uncertainty which is no longer appropriate in view of the removal of the uncertainties in the previous item.

m. Page 3/4 2-15 (3/4 2-16). This page changes the DNB parameters of T_{avg} to $\leq$ 591°F and Total Flow Rate to $\geq$ 289,200 gpm as used in the core uprate analysis.

n. Page 3/4 3-2, (3/4 3-2). Table 3.3.1, "Reactor Trip Instrumentation," has been revised to provide a greater level of redundancy for the source range neutron flux trip during shutdown Modes 3, 4, and 5. This change is more restrictive.

o. Page 3/4 3-10 (3/4 3-10). Table 3.3.2, "Reactor Trip System Instrumentation Response Times," has been revised to require response time testing for the source range neutron flux trip. This change is more restrictive.

p. Pages 3/4 3-16, 3/4 3-31 (3/4 3-16, 3/4 3.33). Tables 3.3.3 and 4.3.2 have been modified to require Safety Injection from a containment pressure-high signal during Mode 4 operation. This change is more restrictive.

q. Page 3/4 5-6 (3/4 5-6). T.S. 3.5.3, ECCS Subsystems-T_{avg} < 300 Degrees F has been revised to require an operable flow path capable of automatically transferring fluid to the RCS when taking suction from the RWST. This change is more restrictive.

r. Pages B 3/4 2-1, B 3/4 2-4, B 3/4 2-5, B 3/4 2-6. (B 3/4 2-1, B 3/4 2-5, B 3/4 2-6). The changes to these pages of the Bases appropriately reflect the changes to the $F^N_{\Delta H}$ and DNB Specifications discussed above.

s. Page B 3/4 7-1, (B 3/4 7-1). The change on this page of the Bases corrects the steam flow and safety valve lift settings to reflect the uprated power.

Evaluation

The staff has reviewed the licensee's proposed increase in the licensed core power rating for NA-1&2 with respect to the impact on the NA-1&2 NSSS design and operations as follows:

(1) The consequences of accidents postulated in the NA-1&2 FSAR,

(2) The capability of systems and equipment to meet design bases and criteria specified in the NA-1&2 FSAR,

(3) The definition of NSSS/BOP safety related interfaces, and

(4) Operating limits and conditions specified in the NA-1&2 TS that are affected by the instant proposed power increase.

Based on the staff's review, it has been determined that the licensee's reanalysis of the ECCS performance for postulated small and large break LOCA's for the proposed NA-1&2 core power uprate is in compliance with Appendix K to 10 CFR Part 50. In addition, the staff has verified that the analytical techniques used in the licensee's reanalysis are in full compliance with Appendix K, 10 CFR part 50.

Further, the staff has determined that NA-1&2 are capable, in their present design configuration to operate at the proposed rated core power level at 2893 MWt and a NSSS power rating of 2905 MWt without violating any of the design criteria or safety limits specified in the NA-1&2 FSAR. NA-1&2 shall be operated in accordance with the TS changes proposed for the core uprate as approved by the staff and as specified above.

Based on all of the above, the staff finds that:

1. The probability of a malfunction of NSSS/BOP equipment important to safety previously evaluated in the FSAR will not be increased at the proposed power rating.

2. The consequences of a malfunction of NSSS/BOP equipment important to safety previously evaluated in the FSAR will not be increased at the proposed power rating.

3. The possibility of a malfunction of NSSS/BOP equipment important to safety different from any already evaluated in the FSAR is not created by operation at the proposed power rating.

4. The margin of safety as defined in the bases to any TS will not be reduced by operation at the proposed power rating.

Therefore, we find the proposed 4.5 percent increase in the reactor core power level from 2775 MWt to 2893 MWt to be acceptable. Also, the proposed changes in the NA-1&2 TS to support a reactor core power level of 2893 MWt are acceptable.

Environmental Considerations

Pursuant to 10 CFR 51.32, the Commission has determined that issuance of the amendments will have no significant impact on the environment (August 11, 1986, 51 FR 28784).

Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 25, 1986

Principal Contributors:
W. Jensen, J. Minns, M. Dunenfeld, H. Gilpun, N. Trehan, T. Quay, and L. Engle

7590-01

U.S. NUCLEAR REGULATORY COMMISSION

VIRGINIA ELECTRIC AND POWER COMPANY, ET AL

DOCKET NOS. 50-338 AND 50-339

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 84 and 71 to Facility Operating License Nos. NPF-4 and NPF-7, issued to Virginia Electric and Power Company and Old Dominion Electric Cooperative (the licensee), which amended the Licenses for operation of the North Anna Power Station, Units 1 and 2 (the facilities), located in Louisa County, Virginia. Amendment No. 84 to Facility Operating License No. NPF-4 for the North Anna Power Station, Unit No. 1 is effective within 60 days from the date of issuance. Amendment No. 71 to Facility Operating License No. NPF-7 for the North Anna Power Station, Unit No. 2 is effective within 30 days from the date of issuance.

The amendments revise the North Anna Power Station, Units No. 1 and 2 Facility Operating Licenses No. NPF-4 and No. NPF-7, and the Technical Specifications, respectively, to increase the presently rated core power level of 2775 Megawatts-thermal to 2893 Megawatts-thermal. The amendments allow the two units to operate at a Nuclear Steam Supply thermal power of 2905 Megawatts-thermal. The amendments represent an increase of approximately 4.5 percent over the currently licensed core power rating and nuclear steam thermal power rating. The amendments will increase the electrical power output for each unit by 32 Megawatts-electrical.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments and Opportunity for Prior Hearing in connection with this action were published in the FEDERAL REGISTER on July 26, 1985 (50 FR 30550). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission prepared an Environmental Assessment and Finding of No Significant Impact (August 11, 1986, 51 FR 28784) related to the action and concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action significantly beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated April 1973, as amended.

For further details with respect to this action, see (1) the application for amendments dated May 2, 1985, as supplemented by letters dated February 6, April 30, June 4, July 3, and August 20, 1986, (2) Amendment Nos. 84 and 71 to Facility Operating License Nos. NPF-4 and NPF-7, (3) and the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093 and the Alderman Library, Louisa, Virginia 23093 and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of PWR Licensing-A.

Dated at Bethesda, Maryland, this 25th day of August, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Lester S. Rubenstein, Director
PWR Project Directorate No. 2
Division of PWR Licensing-A