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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-72

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

   A. The applications for amendment by Florida Power Corporation, et al (the licensees) dated November 29, 1978, February 28, 1979, November 20, 1979, and July 9, 1981, and supplemental filings comply 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 applications, 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 paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. DPR-72 are hereby amended to read as follows:

   (1) Maximum Power Level

   Florida Power Corporation is authorized to operate the facility at a steady state reactor core power level not in excess of 2544 Megawatts (100 percent of rated core power level).

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 21, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

Remove Pages	Insert Pages
1-1	1-1
2-2	2-2
2-3	2-3
2-5	2-5
2-6	2-6
2-7	2-7
B2-3	B2-3
B2-4	B2-4
B2-5	B2-5
B2-6	B2-6
B2-7	B2-7
B2-8	B2-8
3/4 2-3	3/4 2-3
3/4 3-2	3/4 3-2
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2544 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

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

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified as a reportable occurrence in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

b. All equipment hatches are closed and sealed,

c. Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

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

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the safety limit shown in Figure 2.1-2 for the various combinations of three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate safety limit, be in HOT STANDBY within one hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

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

ACTION:

MODES 1 and 2 Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3, 4 and 5 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

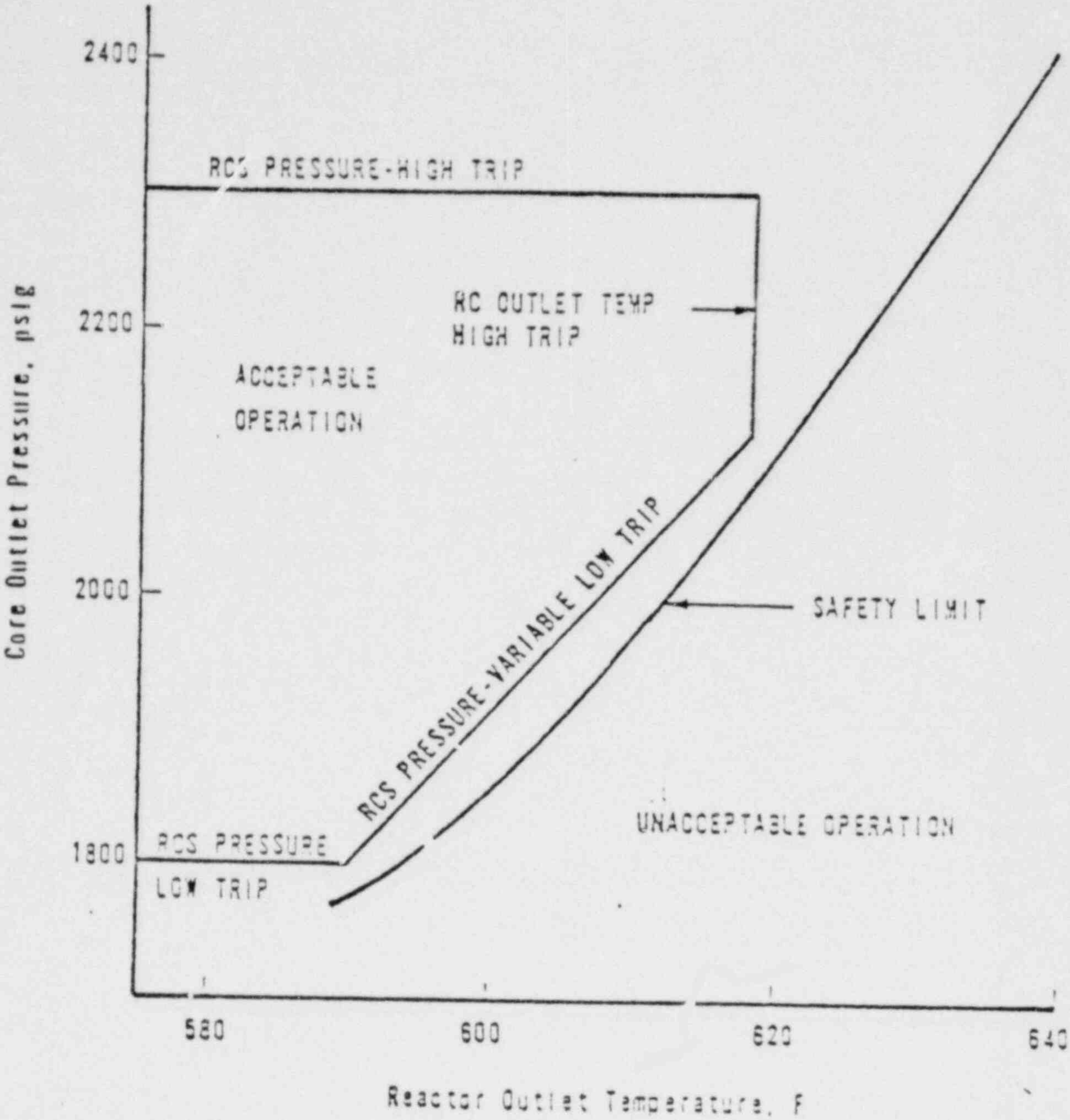


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

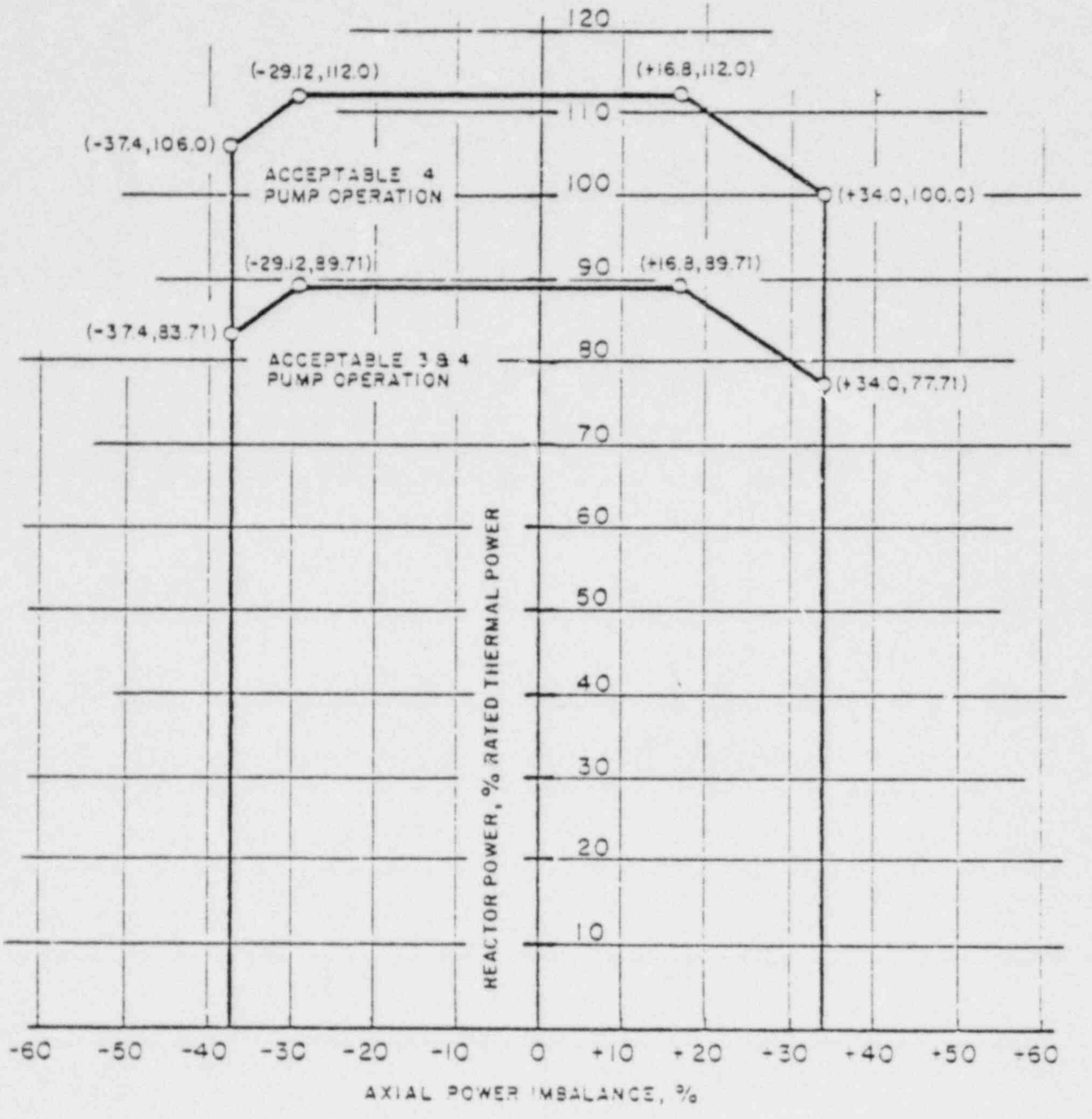


FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection 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 Protection 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 PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq$ 104.88% of RATED THERMAL POWER with four pumps operating $\leq$ 79.92% of RATED THERMAL POWER with three pumps operating	$\leq$ 104.88% of RATED THERMAL POWER with four pumps operating $\leq$ 79.92% of RATED THERMAL POWER with three pumps operating
3. RCS Outlet Temperature-High	$\leq$ 618°F	$\leq$ 618°F
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE(1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1	Allowable Values not to exceed the limit line of Figure 2.2-1
5. RCS Pressure-Low(1)	$\geq$ 1800 psig	$\geq$ 1800 psig
6. RCS Pressure-High	$\leq$ 2300 psig	$\leq$ 2300 psig
7. RCS Pressure-Variable Low(1)	$\geq$ (11.59 T_{out} °F - 5037.8) psig	$\geq$ (11.59 T_{out} °F - 5037.8) psig

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Nuclear Overpower Based on Reactor Coolant Pump Power Monitors[i]	More than one pump not operating	More than one pump not operating
9. Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig

[i] Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is < 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of < 1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

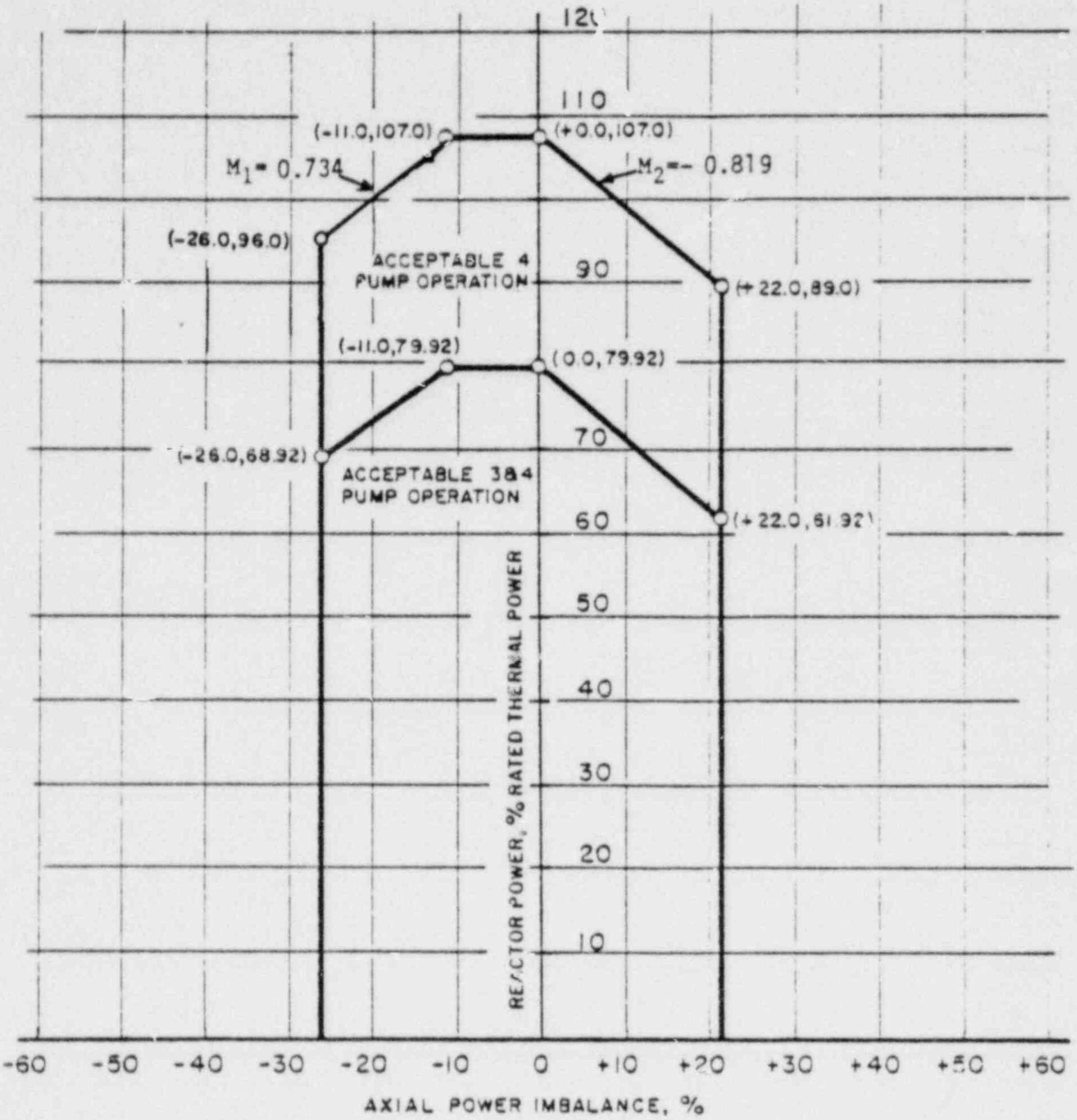


FIGURE 2.2-1

TRIP SETPOINT FOR NUCLEAR OVERPOWER BASED ON RCS FLOW AND AXIAL POWER IMBALANCE

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for three pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the three pump curve will be above and to the left of the other curves.

2.1.3 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 Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to USAS B 31.7, February, 1968 Draft Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Trip Setpoint 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 setpoint less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Shutdown Bypass provides for bypassing certain functions of the Reactor Protection System in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the Shutdown Bypass RCS Pressure-High trip is to prevent normal operation with Shutdown Bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The Nuclear Overpower Trip Setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

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

Nuclear Overpower

A Nuclear Overpower trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 104.88% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip $\leq$ 618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has bee established to accommodate flow decreasing transients from high power.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is $\geq$ 107% and reactor flow rate is 100%, or flow rate is $\leq$ 93.45% and power level is 100%.

2. Trip would occur when three reactor coolant pumps a e operating if power is $\geq$ 79.92% and reactor flow rate is .+.7%, or flow rate is $\leq$ 70.09% and power is 75%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

LIMITING SAFETY SYSTEM SETTINGS

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.07% for a 1% flow reduction.

RCS Pressure - Low, High, and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection and is, therefore, set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, (11.59 T_{out} °F - 5037.8) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of (11.59 T_{out} °F - 5037.8) psig.

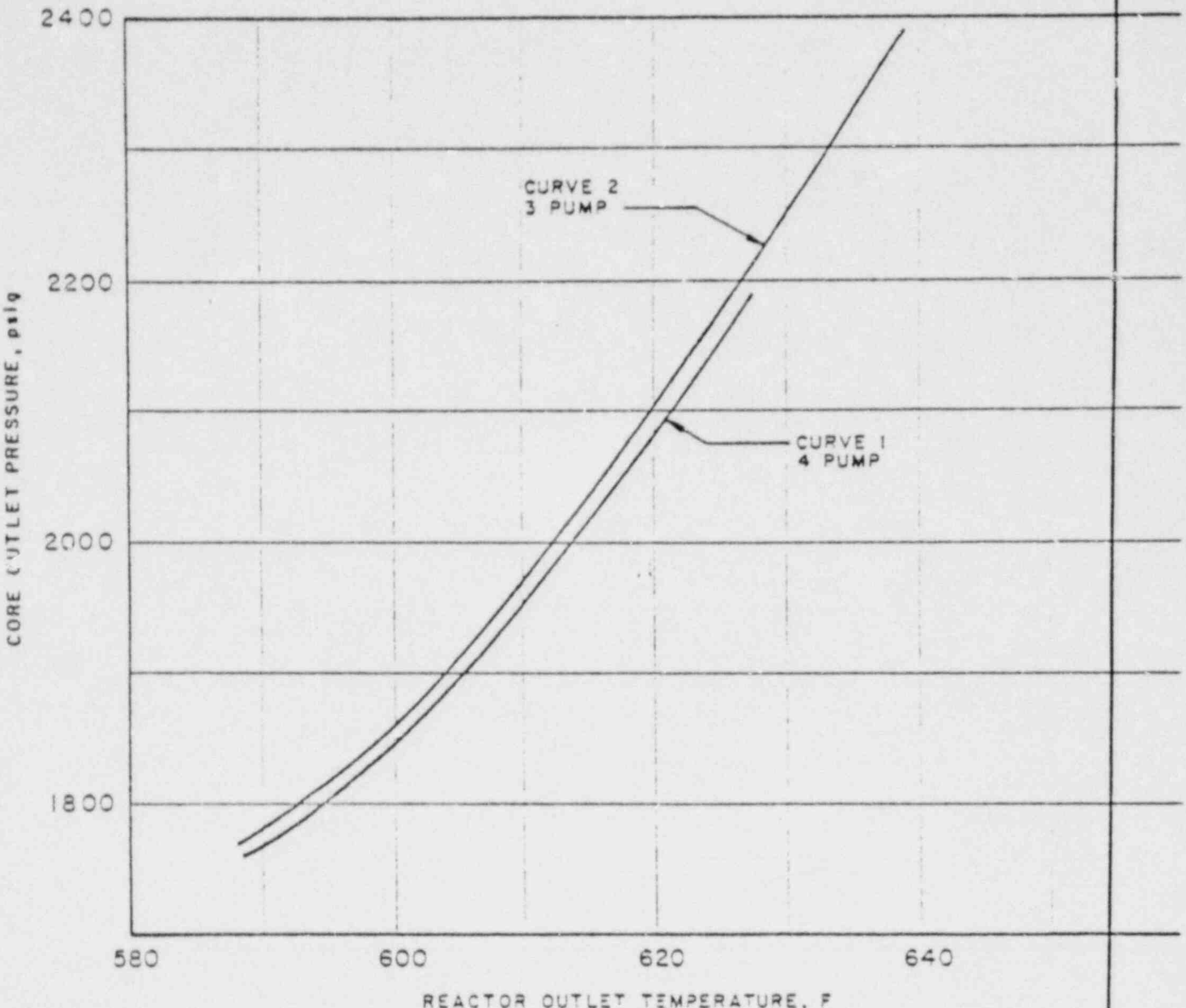
Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint ≤ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure - Low trip.

Reactor Coolant Pump Power Monitors

In conjunction with the power/imbalance/flow trips, the Reactor Coolant Pump Power Monitors trip prevents the minimum core DWBR from decreasing below 1.30 by tripping the reactor due to more than one reactor coolant pump/not operating.

A reactor coolant pump is considered to be not operating when the power required by the pump is >120% or is <70% of the nominal operating power. The nominal operating power decreases from when a pump is first started during heatup and is pumping dense fluid (typically 7500KW) to when a pump is operating at full reactor power and is pumping less dense fluid (typically 5500KW). In order to avoid spurious trips during normal operation, the 120% trip setpoint (9000KW) is based on the nominal operating power for a pump during heatup and the 70% trip setpoint (3900KW) is based on the nominal operating power for a pump operating at full reactor power.



REACTOR COOLANT FLOW

CURVE	FLOW (% DESIGN)	POWER (RTP)	PUMPS OPERATING (TYPE OF LIMIT)
1	139.7×10^6 (106.5%)	113.05%	4 PUMPS (DNBR)
2	104.4×10^6 (79.6%)	90.84%	3 PUMPS (DNBR)

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM
ALLOWABLE POWER FOR MINIMUM DNBR

BASES FIGURE 2.1

THIS FIGURE IS DELETED
FOR THE REMAINDER OF
CYCLE 3

FIGURE 3.2-2

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

$$F_Q \leq \frac{3.08}{P}$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ and $P \leq 1.0$.

APPLICABILITY: MODE 1.

ACTION:

With F_Q exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the Nuclear Overpower Trip Setpoint and Nuclear Overpower based on RCS Flow and AXIAL POWER IMBALANCE Trip Setpoint within 4 hours.

b. Demonstrate through in-core mapping that F_Q 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.

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_Q 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.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_Q shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System instrumentation channels and bypasses 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 Protection System instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR PROTECTION 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 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 PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	1	1	1	1, 2 and *	8
2. Nuclear Overpower	4	2	3	1, 2	2#
3. RCS Outlet Temperature--High	4	2	3	1, 2	3#
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	4	2(a)	3	1, 2	2#
5. RCS Pressure--Low	4	2(a)	3	1, 2	3#
6. RCS Pressure--High	4	2	3	1, 2	3#
7. Variable Low RCS Pressure	4	2(a)	3	1, 2	3#
8. Reactor Containment Pressure--High	4	2	3	1, 2	3#
9. Intermediate Range, Neutron Flux and Rate	2	0	2	1, 2 and *	4
10. Source Range, Neutron Flux and Rate					
A. Startup	2	0	2	2## and *	5
B. Shutdown	2	0	1	3, 4 and 5	6
11. Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
12. Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
13. Shutdown Bypass RCS Pressure-High	4	2	3	2**, 3**, 4**, 5**	6#
14. Reactor Coolant Pump Power Monitors	4	2(a)	3	1, 2	3#

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

a. $\leq$ 5% of RATED THERMAL POWER restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. > 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. $\leq 10^{-10}$ amps on the Intermediate Range (IR) instrumentation, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10^{-10} amps on the IR instrumentation.

b. $> 10^{-10}$ amps on the IR instrumentation, operation may continue.

ACTION 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 within one hour and at least once per 12 hours thereafter.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. Within 1 hour:

1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperative channel.

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1. The inoperable channel above may not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

ACTION 8 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIMES
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower*	$\leq$ 0.3 seconds
3. RCS Outlet Temperature-High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	$\leq$ 1.4 seconds
5. RCS Pressure -- Low	$\leq$ 0.5 seconds
6. RCS Pressure -- High	$\leq$ 0.5 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Reactor Containment Pressure -- High	Not Applicable
9. Reactor Coolant Pump Power Monitors	$\leq$ 0.62 seconds

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

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Nuclear Overpower	S	D(2) and Q(7)	M	1, 2
3. RCS Outlet Temperature--High	S	R	M	1, 2
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	S(4)	M(3) and Q(7,8)	M	1, 2
5. RCS Pressure--Low	S	R	M	1, 2
6. RCS Pressure--High	S	R	M	1, 2
7. Variable Low RCS Pressure	S	R	M	1, 2
8. Reactor Containment Pressure--High	S	R	M	1, 2
9. Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	1, 2 and *
10. Source Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	2, 3, 4 and 5
11. Control Rod Drive Trip Breaker	N.A.	N.A.	M and S/U(1)	1, 2 and *
12. Reactor Trip Module	N.A.	N.A.	M	1, 2, and *
13. Shutdown Bypass RCS Pressure-High	S	R	M	2**, 3**, 4**, 5**
14. Reactor Coolant Pump Power Monitors	S	R	M	1, 2

TABLE 4.3-1 (Continued)

NOTATION

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 30% of RATED THERMAL POWER [RTP], compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I]. as follows:

$$\frac{RTP}{TP} \left| (API_O - API_I) \right| = \text{Imbalance Error}$$

Recalibrate if the absolute value of the Imbalance Error is equal to or greater than 3.5%.

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - Verify at least one decade overlap if not verified in previous 7 days.

(6) - Each train tested every other month.

(7) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(8) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once per 18 months.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

Enclosure 2

May 13, 1981

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations

FROM: Raymond F. Fraley, Executive Director, ACRS

Raymond F. Fraley

SUBJECT: POWER LEVEL INCREASE AT CRYSTAL RIVER NUCLEAR PLANT UNIT 3

During its 253rd meeting, the ACRS heard a report from its Subcommittee on Babcock and Wilcox reactors regarding the request from the Florida Power Corporation to increase the power level of the Crystal River Nuclear Plant from 2452 MWt to 2544 MWt. This request was discussed by the Subcommittee during a meeting in Washington, DC on May 6, 1981.

Based on this report the Committee concluded that it need not review further the proposed power level increase and has no objection to the NRC Staff licensing the Crystal River Nuclear Plant Unit 3 tc operate at power levels up to 2544 MWt.

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