NRC FORM 195
(2-76)

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

DOCKET NUMBER	50-269/270/287	
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TO: Mr. Benard C. Rusche	FROM: Duke Power Company, Charlotte, North Carolina, Mr. William O. Parker, Jr.	DATE OF DOCUMENT: 6/11/76
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DESCRIPTION

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PLANT NAME: ACKNOWLEDGED (2-P)

Oconee 1-2-3

ENCLOSURE

Amdt. to ol/change to tech spec consisting of request to revise the flux/flow trip setpoint for unit 1 as well as incorporating a surveillance testing requirement for the internals vent valves.

(9-P)

FOR ACTION/INFORMATION — ENVIRO 6/18/76 RJL

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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

June 11, 1976

Regulatory Docket File

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Mr. Benard C. Rusche, Director
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch No. 1

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, and -287

Dear Mr. Rusche:

Pursuant to 10CFR50, §50.90, Duke Power Company hereby requests a change in the Oconee Nuclear Station Technical Specifications. The proposed change consists of revising the flux/flow trip setpoint for Oconee Unit 1 as well as incorporating a surveillance testing requirement for the internals vent valves.

In the past the core thermal-hydraulic analysis, including the flux/flow trip setpoint analyses, for Oconee Units 1, 2 and 3 included a core flow penalty for an assumed stuck open vent valve. By letter of January 30, 1976, the Commission provided for elimination of the core flow penalty for an assumed stuck open vent valve by instituting surveillance testing on the vent valves during each refueling outage. Accordingly, to permit elimination of the vent valve flow penalty, a technical specification requiring surveillance testing of all internal vent valves during each refueling outage is proposed. This surveillance testing will confirm that no vent valve is stuck in an open position and that each vent valve exhibits complete freedom of movement.

In the case of Oconee Unit 1, which has commenced Cycle 3 operation, the surveillance testing of all vent valves has been satisfactorily performed during the end-of-Cycle 2 refueling outage. The existing flux/flow trip setpoint is, however, based on an analysis that included a flow penalty for an assumed stuck open vent valve. A re-analysis of the flux/flow trip setpoint has been performed without including the vent valve flow penalty but including the other conservative assumptions and allowances:

6150

(1) a steady-state power level of 108 percent (indicated power level of 102 percent plus 6 percent uncertainty in power level measurement),

(2) design power peaking factors and hot channel factors,

(3) a conservative value for the trip delay time,

(4) maximum effect of fuel densification on DNBR,

(5) an allowance for errors in the coolant inlet temperature and system pressure,

(6) a 5 percent reduction in hot assembly flow to account for flow maldistribution,

(7) a reactor coolant system flow rate of 107.6 percent of the original design flow rate (as compared to the measured flow values of 108.6 percent for Oconee 1), and

(8) a conservative allowance for core bypass flow through control rod and instrument guide tube, core shroud, etc.

The flux/flow ratio resulting from this calculation was further reduced by 1.5 percent for flow signal noise and by 1.2 percent to account for the precision of the various components in the RPS flow instrument string (Δ transmitter, summer amplifier, function generator, and bistable comparator) and yielded a flux/flow trip setpoint of 1.08. However, a flux/flow trip setpoint of 1.07 is proposed to provide additional safety margin.

In the case of Oconee Units 2 and 3, the existing flux/flow trip setpoints are based on analyses that included the vent valve flow penalty; however, future analyses will exclude this penalty.

Enclosed are replacement pages for the Oconee Nuclear Station Technical Specifications incorporating these proposed changes. The proposed changes are identified by vertical lines in the margins of the replacement pages.

Forty (40) copies of this request, including three signed originals, are enclosed.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

PMA:vr
Enclosures

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47 and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr.

William O. Parker, Jr., Vice President

ATTEST:

John C. Goodman, Jr

John C. Goodman, Jr.
Assistant Secretary

Subscribed and sworn to before me this 11th day of June, 1976.

Harriet E. Hull

Notary Public

My Commission Expires:

June 13 - 1976

can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setponts to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 107.6 percent of 131.3 x 10^6 lbs/hr.). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and rod bowing:

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 1.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3 and 4 of Figure 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

The maximum thermal power for three-pump operation is 86.4 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.9 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

4.2.10 For Unit 1, Cycle 3 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.9 will be revised prior to Cycle 4 operation. For Unit 2, Cycle 2 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.9 will be revised prior to Cycle 3 operation. For Unit 3, Cycle 1 operation, the surveillance capsules will be removed from the reactor vessel for a portion of the cycle and the provisions of Specification 4.2.9 will be revised prior to Cycle 2 operation.

4.2.11 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

4.2.12 To assure that reactor internals vent valves are not opening during operation, all vent valves will be inspected during each refueling outage to confirm that no vent valve is stuck open and that each valve operates freely.

Bases

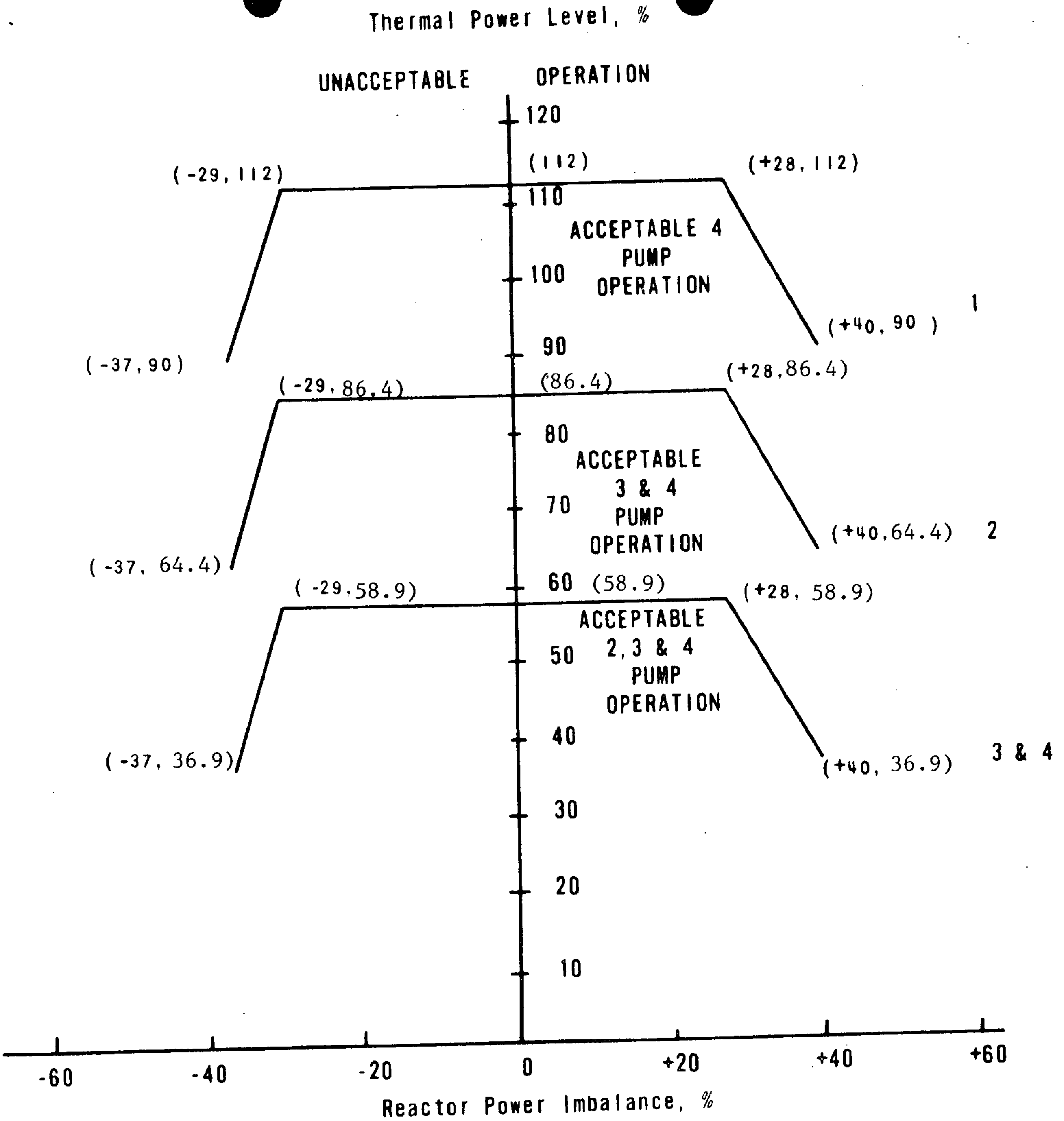
The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The reactor vessel specimen surveillance program for Unit 1 and Unit 2 is based on equivalent exposure times of 1.8, 19.8, 30.6 and 39.6 years. The contents of the different type of capsules are defined below.

A Type	B Type
Weld Material	HAZ Material
HAZ Material	Baseline Material
Baseline Material	

For Unit 3, the Reactor Vessel Surveillance Program is based on equivalent exposure times of 1.8, 13.3, 26.7, and 30.0 years. The specimens have been selected and fabricated as specified in ASTM-E-185-72.

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6
4	64.7×10^6

UNIT 1
CORE PROTECTION SAFETY LIMIT
OCONEE NUCLEAR STATION
FIGURE 2.1-2A

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

<u>Overpower Trip Based on Flow and Imbalance</u>

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip set point 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 trip set point 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 situtations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.5% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.9% and reactor flow rate is 74.7% or flow rate is 70.1% and power level is 75%.

3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 52.4% and the operating loop flow rate is 54.5% or flow rate is 47.9% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.4% and reactor flow rate is 49.0% or flow rate is 45.8% and the power level is 49%.

The flux-to-flow ratios for Units 1 and 2 account for the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

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

The 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 reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.07% for a 1% flow reduction.

The power-to-flow reduction ratio is 0.961 during single loop operation.

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}-4706$) trip
(1800) psig ($10.79\ T_{out}-4539$)
(1800) psig ($16.25\ T_{out}-7756$)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}\ -4746$)
($10.79\ T_{out}\ -4579$)
($16.25\ T_{out}\ -7796$)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620^{o}F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1A. Two conditions are imposed when
2.3-1B
2.3-1C
the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.
2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Two Pump Operation

A. Two Loop Operation

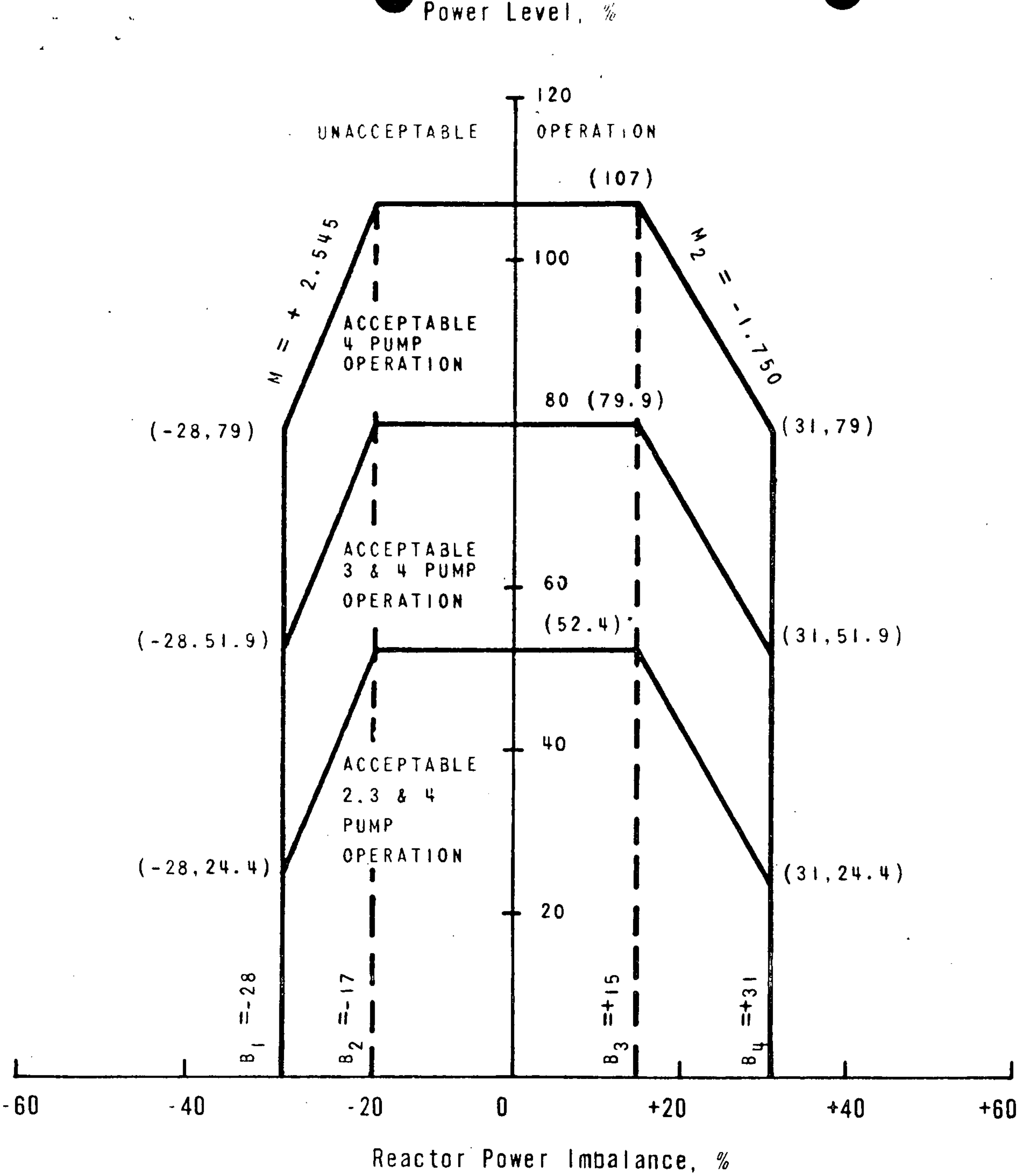
Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, reset the pump contact monitor power level trip setpoint to 55.0%.

B. Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred, the following actions will permit single loop operation:
1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. Trip one of the two protective channels receiving outlet temperature information from sensors in the Idle Loop.
3. Reset flux-flow setpoint to 0.961.

REFERENCES

(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6



UNIT 1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
OCONEE NUCLEAR STATION

FIGURE 2.3-2A

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	0.961 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55% (5)	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (^{o}F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

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TO: Mr. Benard C. Rusche

FROM: Duke Power Company, Charlotte, North Carolina, Mr. William O. Parker, Jr.

DATE OF DOCUMENT: 6/11/76

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CONTROL NUMBER: 6150

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

June 11, 1976

Mr. Benard C. Rusche, Director
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch No. 1

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, and -287

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William O. Parker

William O. Parker, Jr.

PMA:vr
Enclosures

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William O. Parker, Jr.

William O. Parker, Jr., Vice President

ATTEST:

John C. Goodman, Jr.

John C. Goodman, Jr.
Assistant Secretary

Subscribed and sworn to before me this 11th day of June, 1976.

Harriet E. Hull

Notary Public

My Commission Expires:

June 13 - 1976

can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setponts to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 107.6 percent of 131.3 x 10^6 lbs/hr.). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and rod bowing:

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

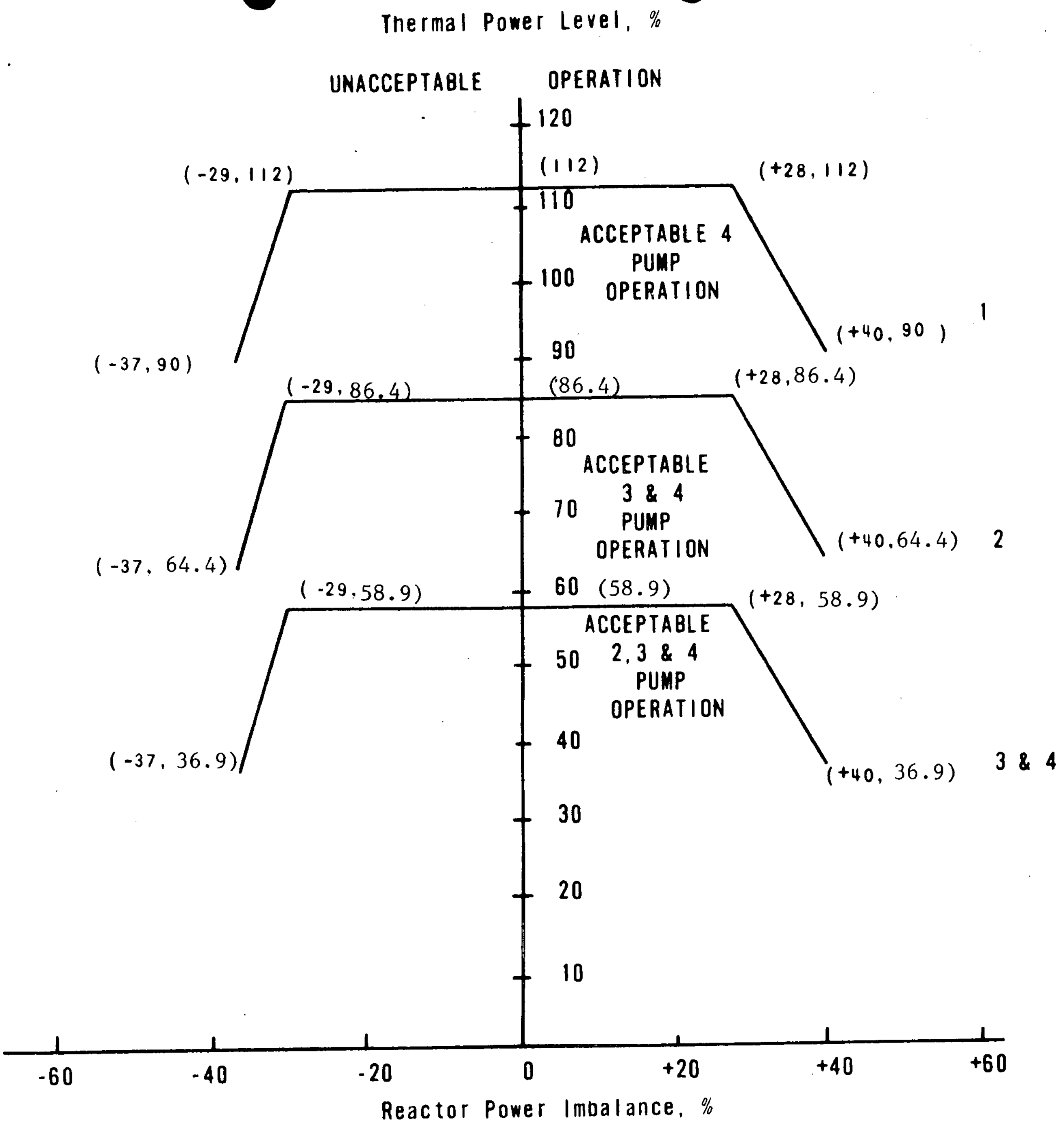
2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 1.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3 and 4 of Figure 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

The maximum thermal power for three-pump operation is 86.4 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.9 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6
4	64.7×10^6

UNIT 1
CORE PROTECTION SAFETY LIMIT
OCONEE NUCLEAR STATION
FIGURE 2.1-2A

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip set point 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 trip set point 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 situtations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.5% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.9% and reactor flow rate is 74.7% or flow rate is 70.1% and power level is 75%.

3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 52.4% and the operating loop flow rate is 54.5% or flow rate is 47.9% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.4% and reactor flow rate is 49.0% or flow rate is 45.8% and the power level is 49%.

The flux-to-flow ratios for Units 1 and 2 account for the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

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

The 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 reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.07% for a 1% flow reduction.

The power-to-flow reduction ratio is 0.961 during single loop operation.

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}-4706$) trip
(1800) psig ($10.79\ T_{out}-4539$)
(1800) psig ($16.25\ T_{out}-7756$)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}-4746$)
($10.79\ T_{out}-4579$)
($16.25\ T_{out}-7796$)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620°F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1A. Two conditions are imposed when
2.3-1B
2.3-1C
the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.
2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Two Pump Operation

A. Two Loop Operation

Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, reset the pump contact monitor power level trip setpoint to 55.0%.

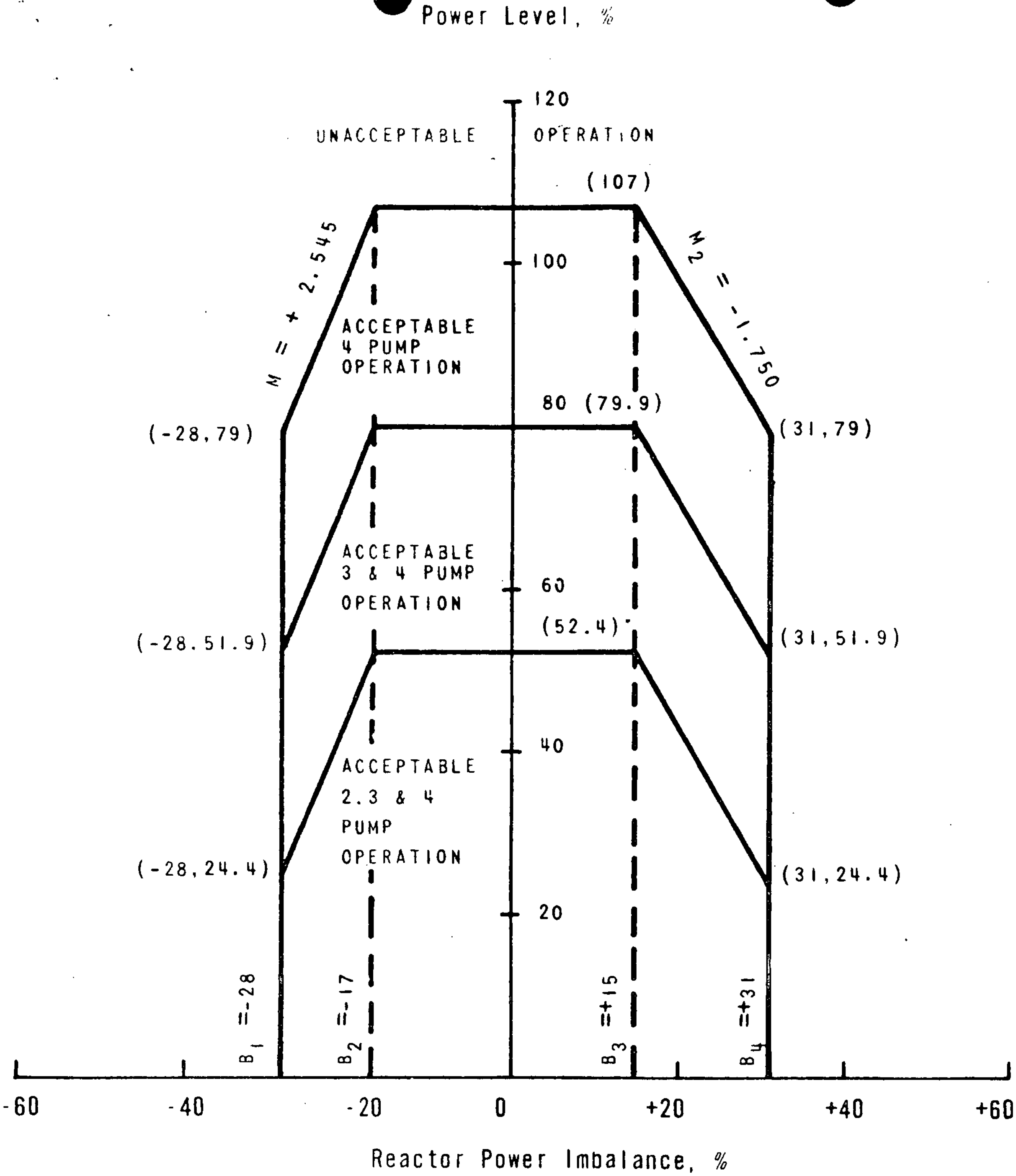
B. Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred, the following actions will permit single loop operation:

1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. Trip one of the two protective channels receiving outlet temperature information from sensors in the Idle Loop.
3. Reset flux-flow setpoint to 0.961.

REFERENCES

(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6



UNIT 1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
OCONEE NUCLEAR STATION

FIGURE 2.3-2A

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0 (3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	0.961 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55% (5)	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720 (4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)$ (1)	$(11.14\ T_{out} - 4706)$ (1)	$(11.14\ T_{out} - 4706)$ (1)	$(11.14\ T_{out} - 4706)$ (1)	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (^{o}F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

4.2.10 For Unit 1, Cycle 3 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.9 will be revised prior to Cycle 4 operation. For Unit 2, Cycle 2 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.9 will be revised prior to Cycle 3 operation. For Unit 3, Cycle 1 operation, the surveillance capsules will be removed from the reactor vessel for a portion of the cycle and the provisions of Specification 4.2.9 will be revised prior to Cycle 2 operation.

4.2.11 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

4.2.12 To assure that reactor internals vent valves are not opening during operation, all vent valves will be inspected during each refueling outage to confirm that no vent valve is stuck open and that each valve operates freely.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The reactor vessel specimen surveillance program for Unit 1 and Unit 2 is based on equivalent exposure times of 1.8, 19.8, 30.6 and 39.6 years. The contents of the different type of capsules are defined below.

A Type	B Type
Weld Material	HAZ Material
HAZ Material	Baseline Material
Baseline Material	

For Unit 3, the Reactor Vessel Surveillance Program is based on equivalent exposure times of 1.8, 13.3, 26.7, and 30.0 years. The specimens have been selected and fabricated as specified in ASTM-E-185-72.

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.