JUN 16 1978

Docket No. 50-346

Toledo Edison Company
ATTN: Mr. Lowell E. Roe
Vice President, Facilities
Development
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NO. NPF-3 FOR DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 11 to Facility Operating License No. NPF-3 for the Davis Besse Nuclear Power Station, Unit 1. The amendment is effective as of the date of issuance.

Amendment No. 11 consists of changes to the Technical Specifications in response to your applications dated April 10, 1978, and May 18, 1978, as supplemented.

Amendment No. 11 revises the Technical Specification on the DNBR Margin to reflect an increase in the observed reactor coolant system flow rate. We have determined that the actual reactor coolant system flow rate exceeds the design flow rate by an amount sufficient to compensate for the additional flow margin required to address the DNBR penalties as specified in License Condition 2.C.(3)(i).

Therefore, we find that License Condition 2.C.(3)(i) is no longer necessary and License Condition 2.C.(3)(i) is hereby deleted from Facility Operating License NPF-3 effective as of the date of issuance of Amendment No. 11.

In addition, we have determined that the actual excess reactor coolant flow rate is adequate to compensate for the increased bypass flow brought about by the removal of all burnable poison rod assemblies and all but two of the orifice rod assemblies.

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Also, we find that the modifications which you have proposed for the hold down mechanisms and the two orifice rod assemblies remaining in the core as primary neutron sources provide reasonable assurance that these two modified orifice rod assemblies will pose no significant safety concern for use at Davis Besse, Unit 1 for the duration of the first fuel cycle.

As discussed previously with you, we determined that your proposed power ascension testing program for the removal of the burnable poison rod assemblies and the orifice rod assemblies should be augmented with certain additional tests during the initial startup and power ascension test program for Davis Besse, Unit 1. We find your commitments as specified in your letters of June 8, 1978, and June 13, 1978, for performing the additional tests which we have requested to be acceptable.

Based on our review and evaluation of your application with supporting analyses, we find that the Davis Besse Nuclear Power Station, Unit 1, can be operated safely during the remaining portion of Cycle 1 without burnable poison rods and with two modified orifice rod assemblies at the rated power level of 2772 Megawatts-thermal. The revised Technical Specifications necessary for safe operation are provided in Amendment No. 11.

Finally, your requested Technical Specification change, unrelated to the core modifications, regarding a change in the alarm setpoints on quadrant tilt has been found to be acceptable. This revised Technical Specification is also provided in Amendment No. 11.

We have determined that Amendment No. 11 does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connnection with the issuance of this amendment.

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Copies of the Federal Register Notice of Issuance of Amendment No. 11 and the Safety Evaluation Supporting Amendment No. 11 to License NPF-3 are also enclosed.

Sincerely,
Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactor Branch No. 1
Division of Project Management

Enclosures:
1. Amendment No. 11 to NPF-3
2. FEDERAL REGISTER Notice
3. Safety Evaluation Supporting Amendment 11 to NPF-3

ccs w/enclosures: See page 3

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| OFFICE➤ | LWR 1 | LWR 1 | OELD | LWR 1 | | |
| SURNAME➤ | EHylton/red | LEngle | | JStolz | | |
| DATE➤ | 6/14/78 | 6/14/78 | 6/16/78 | 6/16/78 | | |

Toledo Edison Company

cc: Donald H. Hauser, Esq.
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Attorney General
Department of Attorney General
30 East Broad Street
Columbus, Ohio 43215

Ohio Department of Health
ATTN: Director of Health
450 East Town Street
Columbus, Ohio 43216

Harold Kahn, Staff Scientist
Power Siting Commission
361 East Broad Street
Columbus, Ohio 43216

Mr. Harry R. Johnson
Ottawa County Courthouse
Port Clinton, Ohio 43452

U. S. Environmental Protection Agency
Federal Activities Branch
ATTN: EIS Coordinator
Region V Office
230 South Dearborn Street
Chicago, Illinois 60604

Mr. Jack E. Hemphill
U. S. Fish & Wildlife Service
Federal Building
Fort Snelling
Twin Cities, Minnesota 55111

Mr. Frederick O. Rouse, Chairman (2 copies)
Great Lakes Basin Commission
P. O. Box 999
Ann Arbor, Michigan 49106

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

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. NPF-3

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

   A. The issuance of this license amendment 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 license, as amended, 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 amended Facility Operating License No. NPF-3 is hereby amended by changing the Technical Specifications as indicated in the attachment to this license amendment. Also, the license is amended by deleting License Condition 2.C.(3)(i) to Facility Operating License NPF-3.

2.C.(3) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 11 are hereby incorporated in the license. Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUN 16 1978

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
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   B. The facility will operate in conformity with the license, as amended, 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 amended Facility Operating License No. NPF-3 is hereby amended by changing the Technical Specifications as indicated in the attachment to this license amendment. Also, the license is amended by deleting License Condition 2.C.(3)(i) to Facility Operating License NPF-3.

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2.C.(3) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 11 are hereby incorporated in the license. Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUN 16 1978

SEE PREVIOUS YELLOW FOR CONCURRENCES*

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| OFFICE | LWR 1 * | LWR 1 * | OELD | LWR 1 | | |
| SURNAME | EHylton/red | LEngle | GFess* | JStolz | | |
| DATE | 6/ /78 | 6/ /78 | 6/ /78 | 6/16/78 | | |

# ATTACHMENT TO LICENSE AMENDMENT NO. 11

## FACILITY OPERATING LICENSE NO. NPF-3

## DOCKET NO. 50-346

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.

Pages

2-2
2-3
2-5
2-7
2-8
B 2-1
B 2-2
B 2-3
B 2-8
3/4 1-16
3/4 1-26
3/4 1-28
3/4 1-28 a (added)
3/4 1-29
3/4 1-29 a thru c (added)
3/4 1-30
3/4 1-31
3/4 1-32
3/4 2-2
3/4 2-2 a (added)
3/4 2-3
3/4 2-3 a (added)
3/4 2-4
3/4 2-4 a (added)
3/4 2-12
3/4 2-14
B 3/4 1-2
B 3/4 2-1
B 3/4 2-2
5-4

## 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 two, 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.

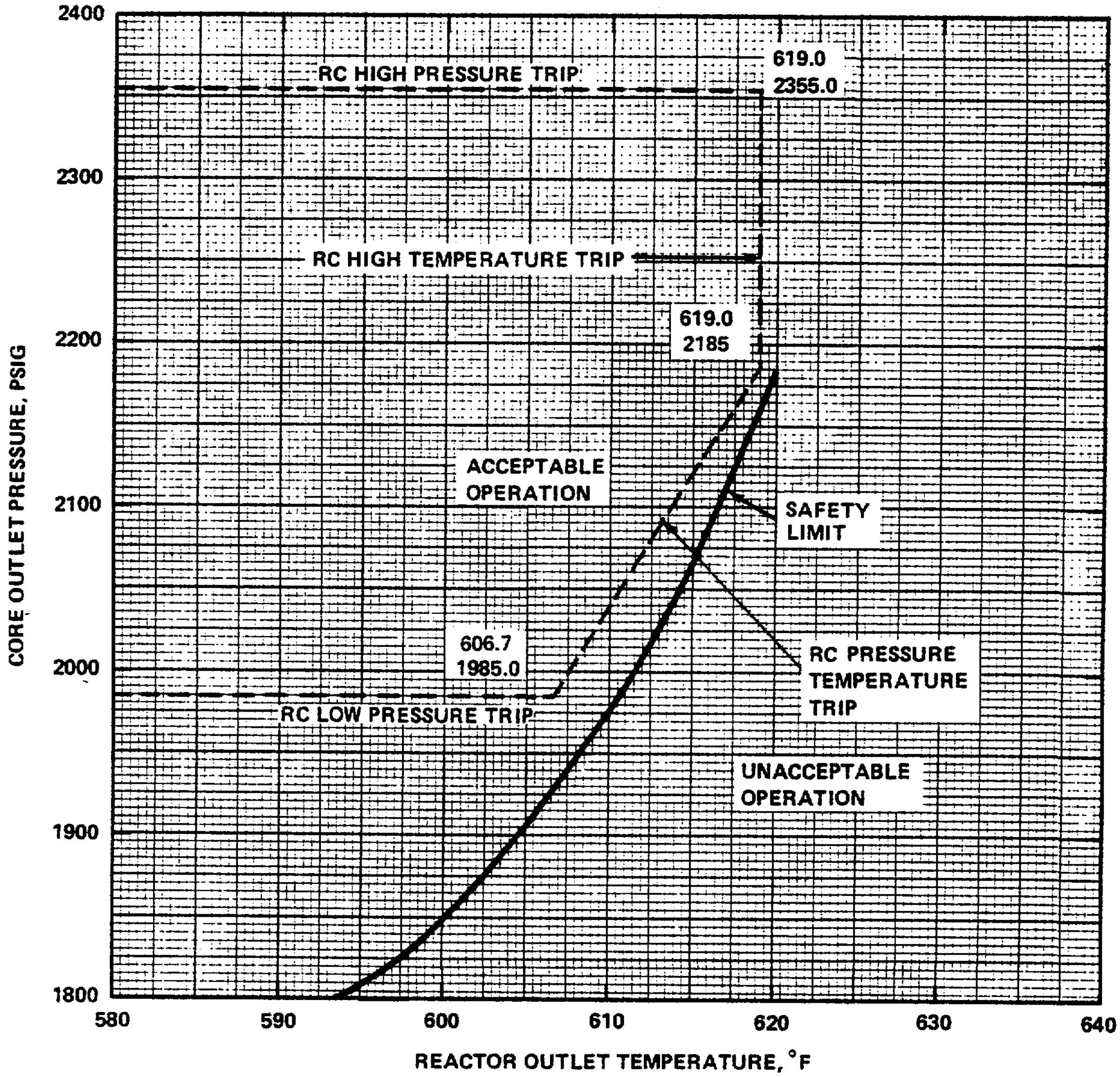
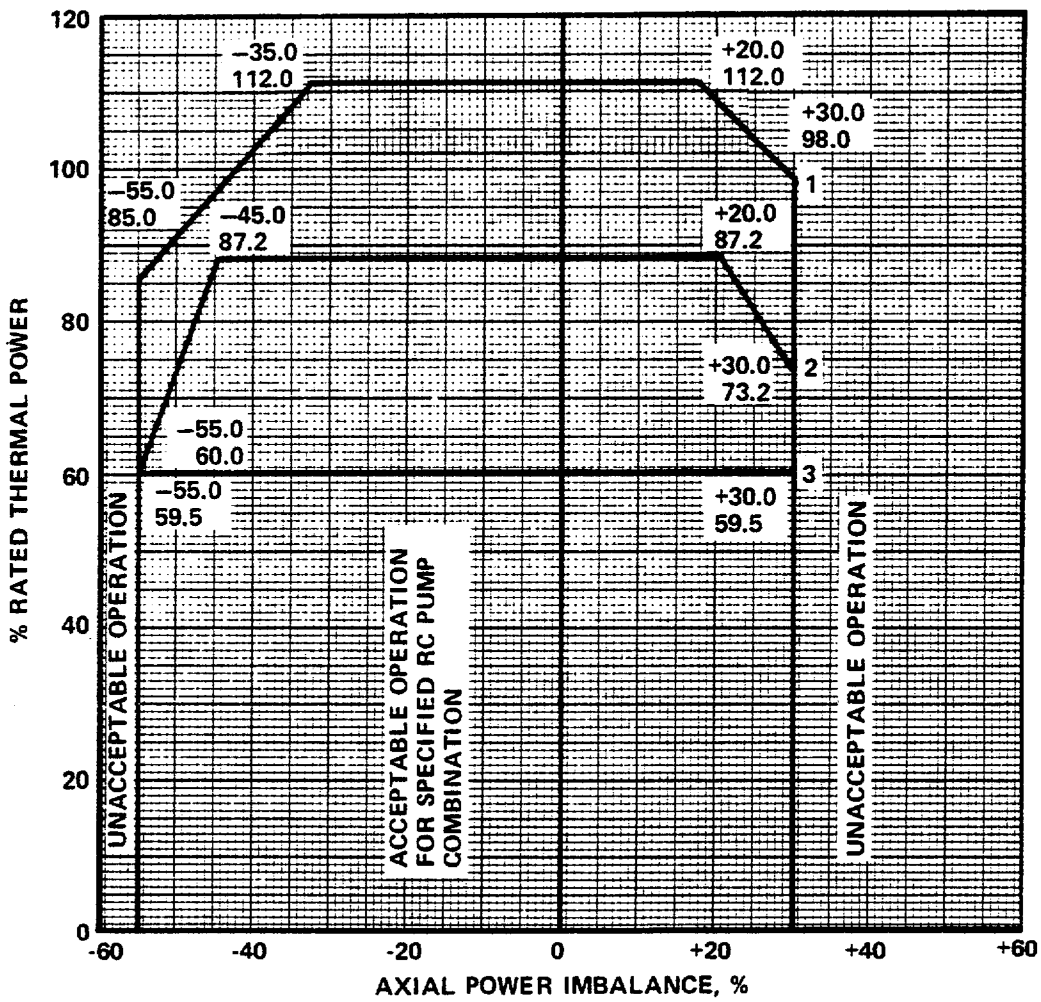



Figure 2.1-1 Reactor Core Safety Limit




| CURVE | REACTOR COOLANT FLOW (GPM) |
|---|---|
| 1 | 387,200 |
| 2 | 290,100 |
| 3 | 191,000 |

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. High Flux | $\leq$ 105.5% of RATED THERMAL POWER with four pumps operating | $\leq$ 105.6% of RATED THERMAL POWER with four pumps operating# | |
| | $\leq$ 80.7% of RATED THERMAL POWER with three pumps operating | $\leq$ 80.8% of RATED THERMAL POWER with three pumps operating# | |
| | $\leq$ 53.0% of RATED THERMAL POWER with one pump operating in each loop | $\leq$ 53.1% of RATED THERMAL POWER with one pump operating in each loop# | |
| 3. RC High Temperature | $\leq$ 619°F | $\leq$ 619.08°F# | |
| 4. Flux - $\Delta$ Flux-Flow(1) | Trip Setpoint not to exceed the limit line of Figure 2.2-1. | Allowable Values not to exceed the limit line of Figure 2.2-2.# | |
| 5. RC Low Pressure(1) | $\geq$ 1985 psig | $\geq$ 1984.0 psig* | $\geq$ 1976.5 psig** |
| 6. RC High Pressure | $\leq$ 2355 psig | $\leq$ 2356.0 psig* | $\leq$ 2363.5 psig** |
| 7. RC Pressure-Temperature(1) | $\geq$ (16.25 $T_{out}$ °F - 7873) psig | $\geq$ (16.25 $T_{out}$ °F - 7873.64) psig# | |

## TABLE 2.2-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 8. High Flux/Number of Reactor Coolant Pumps On(1) | ≤ 55.0% of RATED THERMAL POWER with one pump operating in each loop | ≤ 55.28% of RATED THERMAL POWER with one pump operating in each loop# |
| | ≤ 0.0% of RATED THERMAL POWER with two pump operating in one loop and no pumps operating in the other loop | ≤ 0.28% of RATED THERMAL POWER with two pumps operating in one loop and no pump operating in the other loop# |
| | ≤ 0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating | ≤ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating# |
| 9. Containment Pressure High | ≤ 4 psig | ≤ 4 psig# |

(1)Trip may be manually bypassed when RCS pressure ≤ 1820 psig by actuating Shutdown Bypass provided that:

a. The High Flux Trip Setpoint is ≤ 5% of RATED THERMAL POWER
b. The Shutdown Bypass High Pressure Trip Setpoint of ≤ 1820 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1820 psig.

*Allowable Value for CHANNEL FUNCTIONAL TEST

**Allowable Value for CHANNEL CALIBRATION

#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION

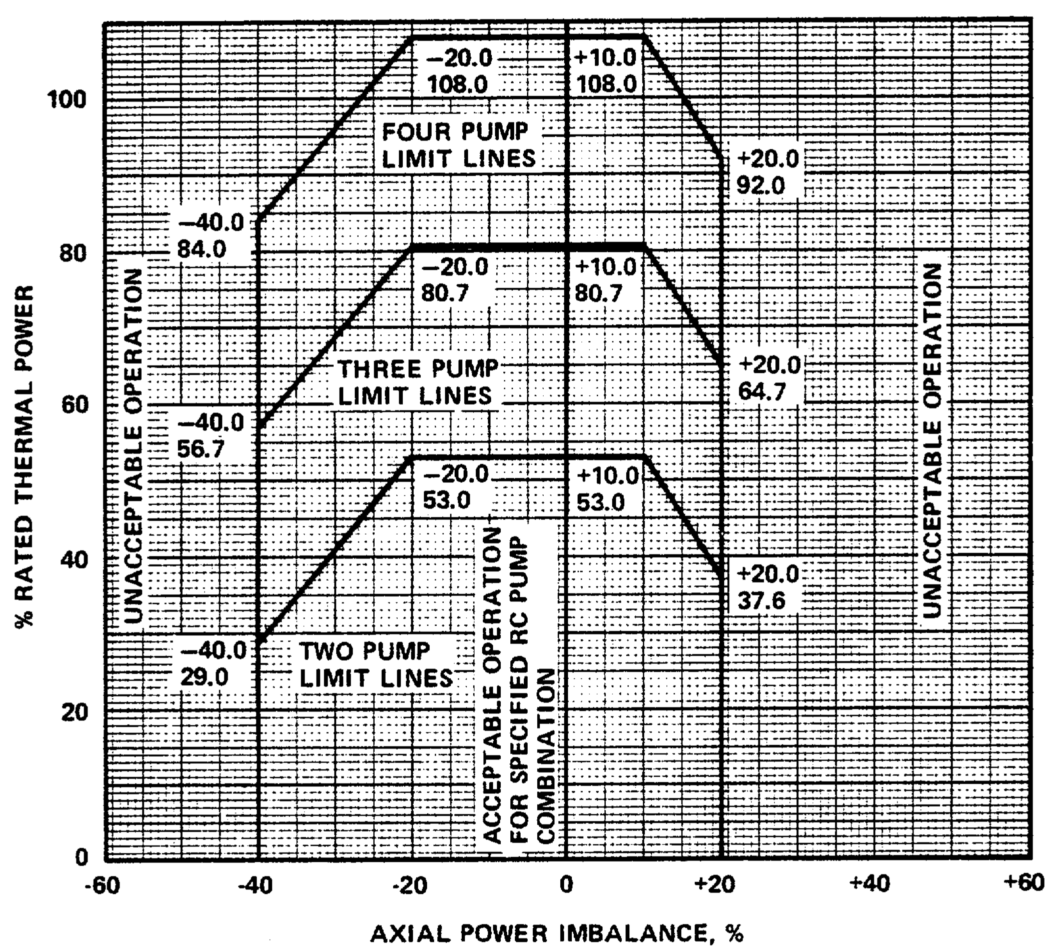



**Figure 2.2-1 Trip Setpoint for Flux—ΔFlux—Flow**

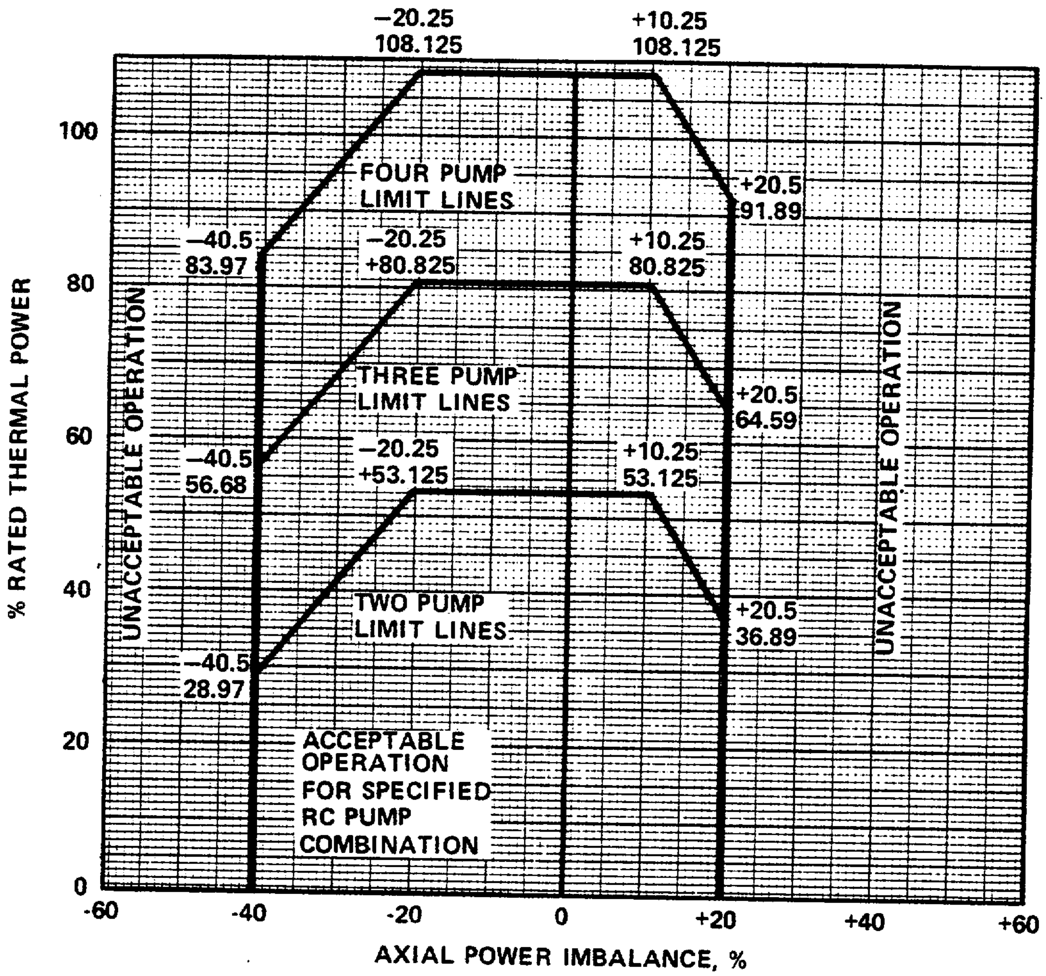



Figure 2.2-2 Allowable Value for Flux—ΔFlux—Flow

## 2.1 SAFETY LIMITS

## BASES

### 2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding 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 would 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 B&W-2 DNB correlation. The 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 minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.32. This value corresponds to a 95 percent probability at a 99 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.32 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 387, 200 GPM, which is 110% of design flow rate for four operating reactor coolant pumps. This curve is based on the following hot channel factors with potential fuel densification and fuel rod bowing effects:

$$F_Q = 2.56; \quad F^N_{\Delta H} = 1.71; \quad F^N_Z = 1.50$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

## BASES

The reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The 1.32 DNBR limit produced by a nuclear power peaking factor of $F_Q$ = 2.56 or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.32 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.4 kw/ft.

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

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

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curve of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.32 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to +22%, whichever condition is more restrictive. This curve includes the potential effects of fuel rod bow and fuel densification.

The DNBR as calculated by the B&W-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of +22% is justified on the basis of experimental data.

BASES

For the 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 1.32 or a local quality at the point of minimum DNBR less than +22% for that particular reactor coolant pump situation. The 1.32 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the three pump and two pump 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 ANSI B 31.7, 1968 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 each 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 High Pressure 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 High Flux 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.

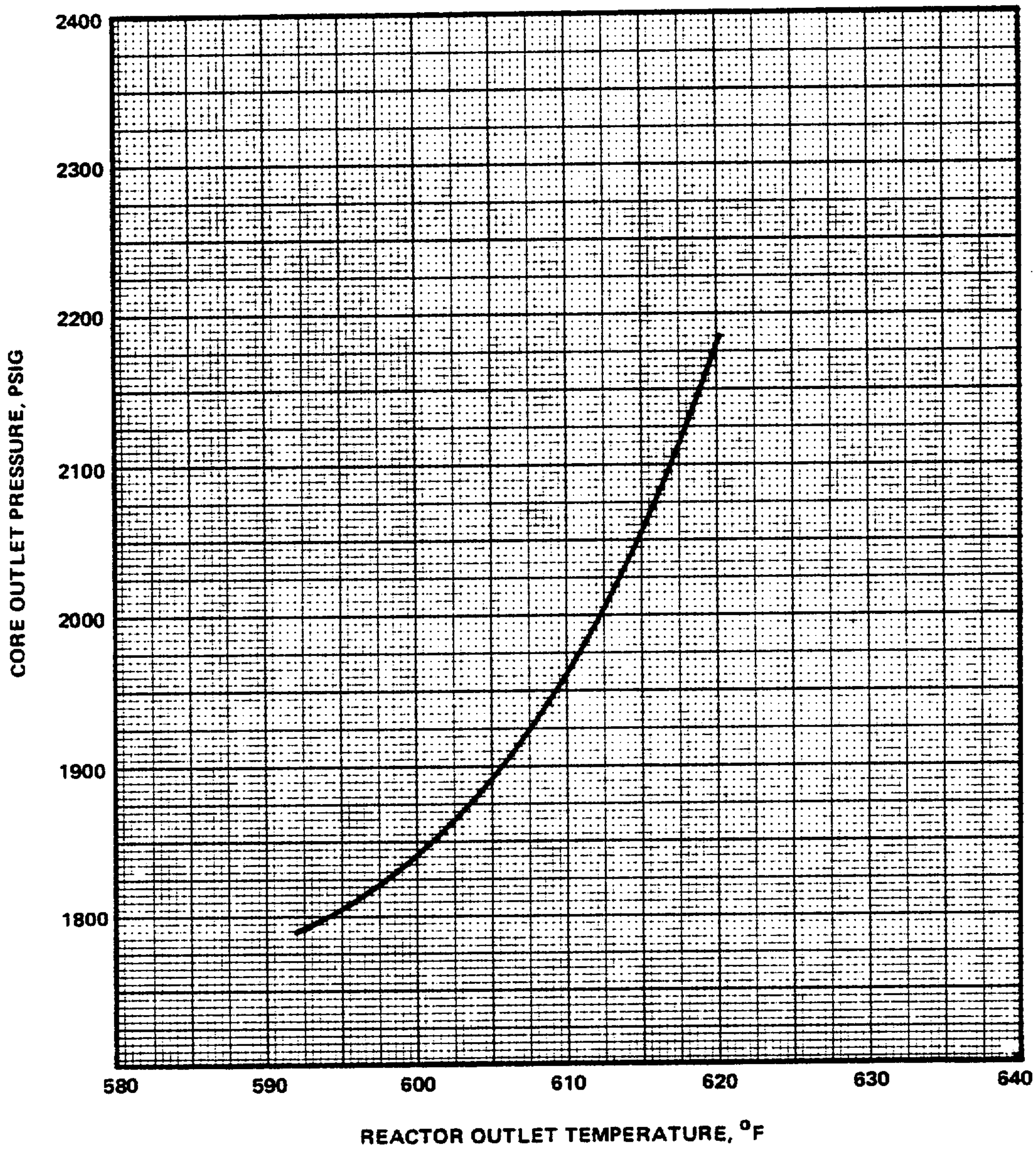
#### Manual Reactor Trip

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

#### High Flux

A High Flux 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 105.5% 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.




| RC Flow | Power | Pumps Operating |
|---|---|---|
| 387,200 GPM | 112% | Four Pumps |

Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR

BASES Figure 2.1

Containment High Pressure

The Containment High Pressure Trip Setpoint $\leq$ 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 RC Low Pressure trip.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the boric acid addition system solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the BWST temperature when it is the source of borated water and the outside air temperature is < 35°F.

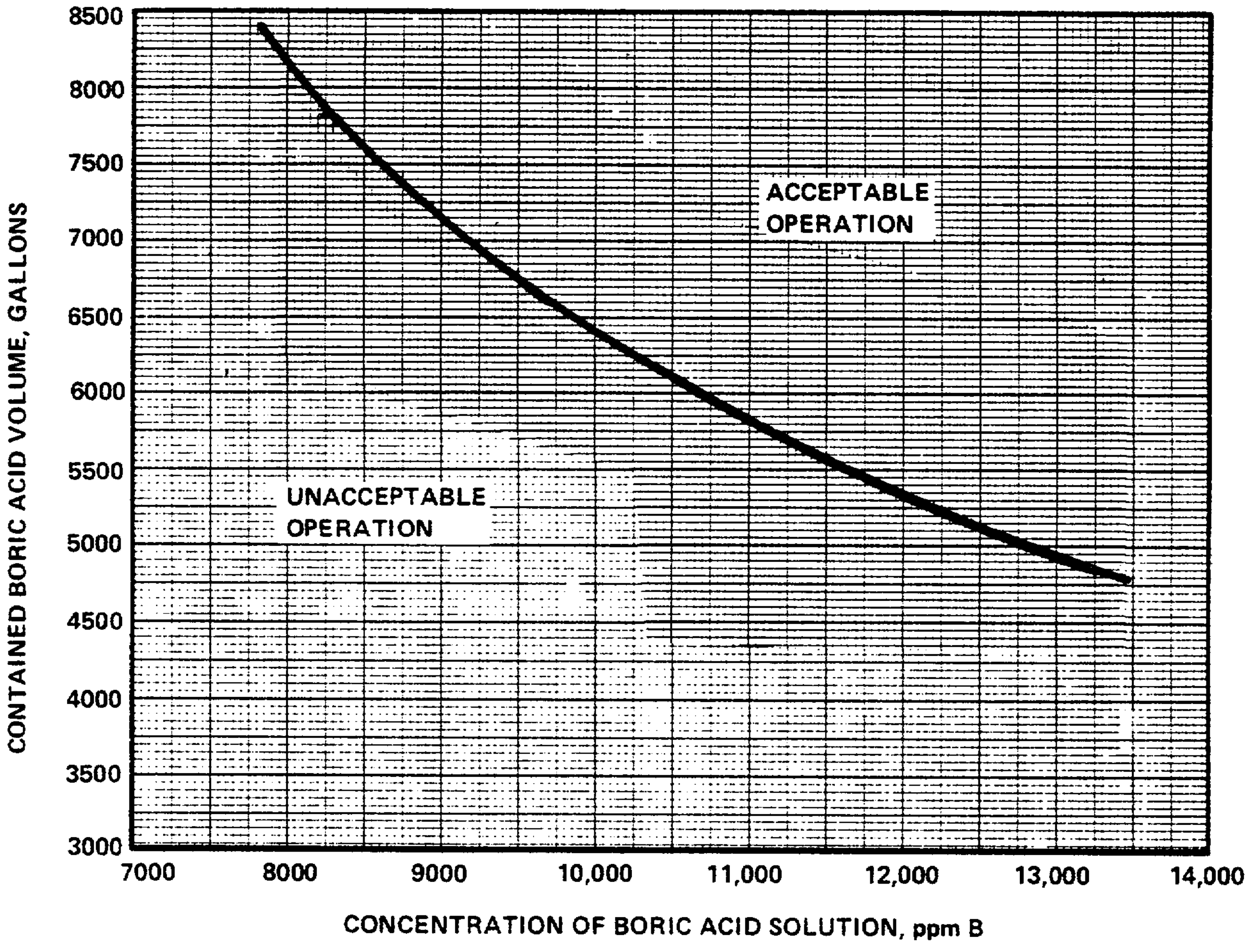



**Figure 3.1-1** Minimum Boric Acid Tank Contained Volume as Function of Stored Boric Acid Concentration—Davis-Besse 1, Cycle 1

## SAFETY ROD INSERTION LIMIT

### LIMITING CONDITION FOR OPERATION

3.1.3.5 All safety rods shall be fully withdrawn.

APPLICABILITY: 1* and 2*#.

ACTION:

With a maximum of one safety rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

### SURVEILLANCE REQUIREMENTS

4.1.3.5 Each safety rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any regulating rod during an approach to reactor criticality.

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.1 and 3.10.2.
#With $K_{eff} \geq 1.0$.

## REGULATING ROD INSERTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-2a and -2b and 3.1-3a, -3b, -3c, and -3d with a rod group overlap of 25 ± 5% between sequential withdrawn groups 5 and 6 for operation up to 145 ± 5 EFPD, and between sequential withdrawn groups 5, 6 and 7 after 145 ± 5 EFPD.**

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits, or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figures within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.1 and 3.10.2.

#With $K_{eff} \geq 1.0$.

**For operation between restart after BPRA removal and 145 ± 5 EFPD, regulating rod group 7 shall be fully inserted in the core and shall not have an overlap with group 6.

## REGULATING ROD INSERTION LIMITS

### SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating group shall be determined to be within the insertion, sequence and overlap limits at least once every 12 hours except when:

a. The regulating rod insertion limit alarm is inoperable, then verify the groups to be within the insertion limits at least once per 4 hours;

b. The control rod drive sequence alarm is inoperable, then verify the groups to be within the sequence and overlap limits at least once per 4 hours.

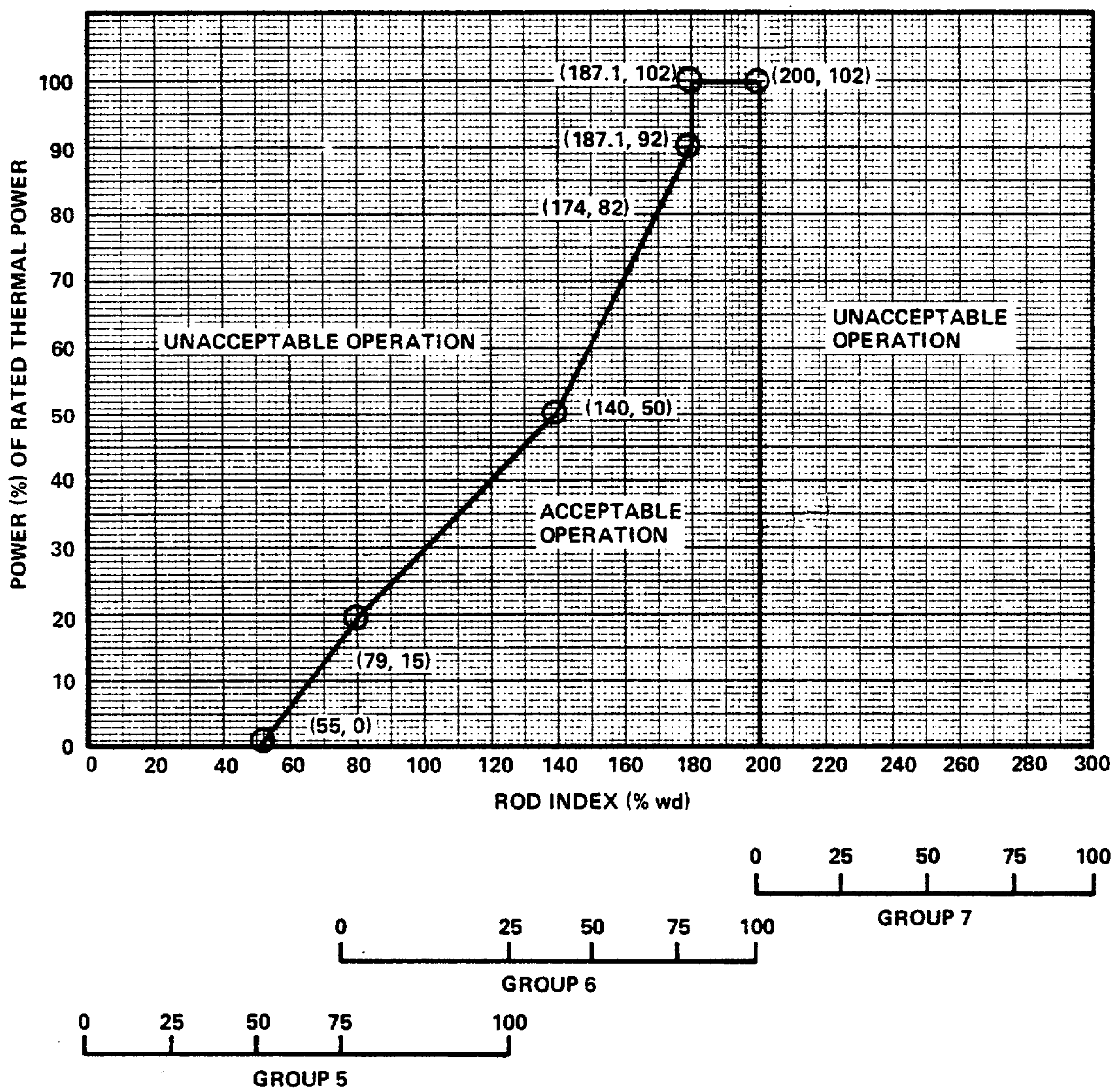



Figure 3.1-2a Regulating Rod Group Insertion Limits for Operation to 145± 5 EFPD (Four Pumps)

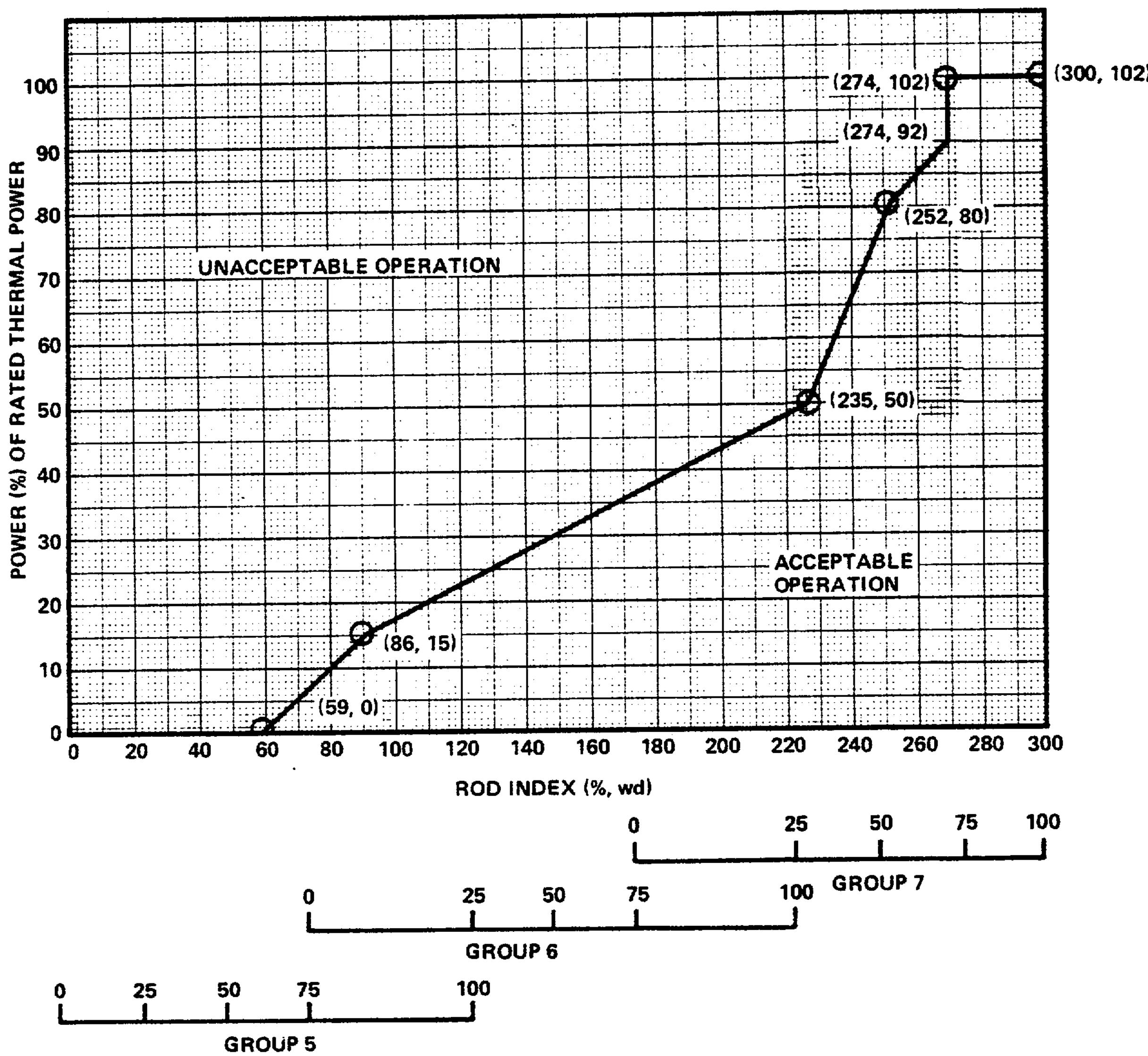



Figure 3.1-2b Regulating Rod Group Insertion Limits for Operation After 145± 5 EFPD (Four Pumps)

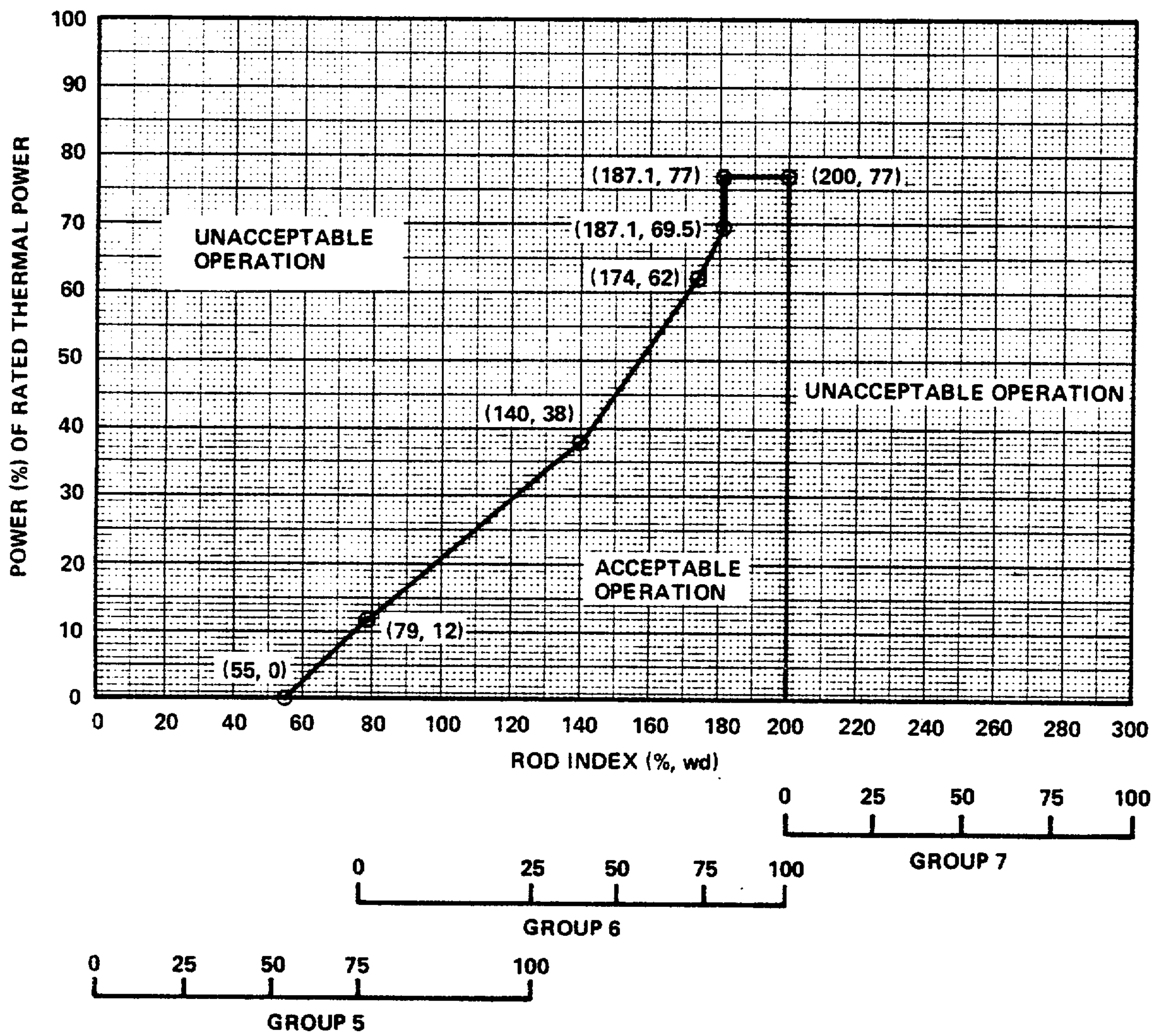



Figure 3.1-3a Regulating Rod Group Insertion Limit for Operation to 145$\pm$ 5 EFPD (Three Pumps)

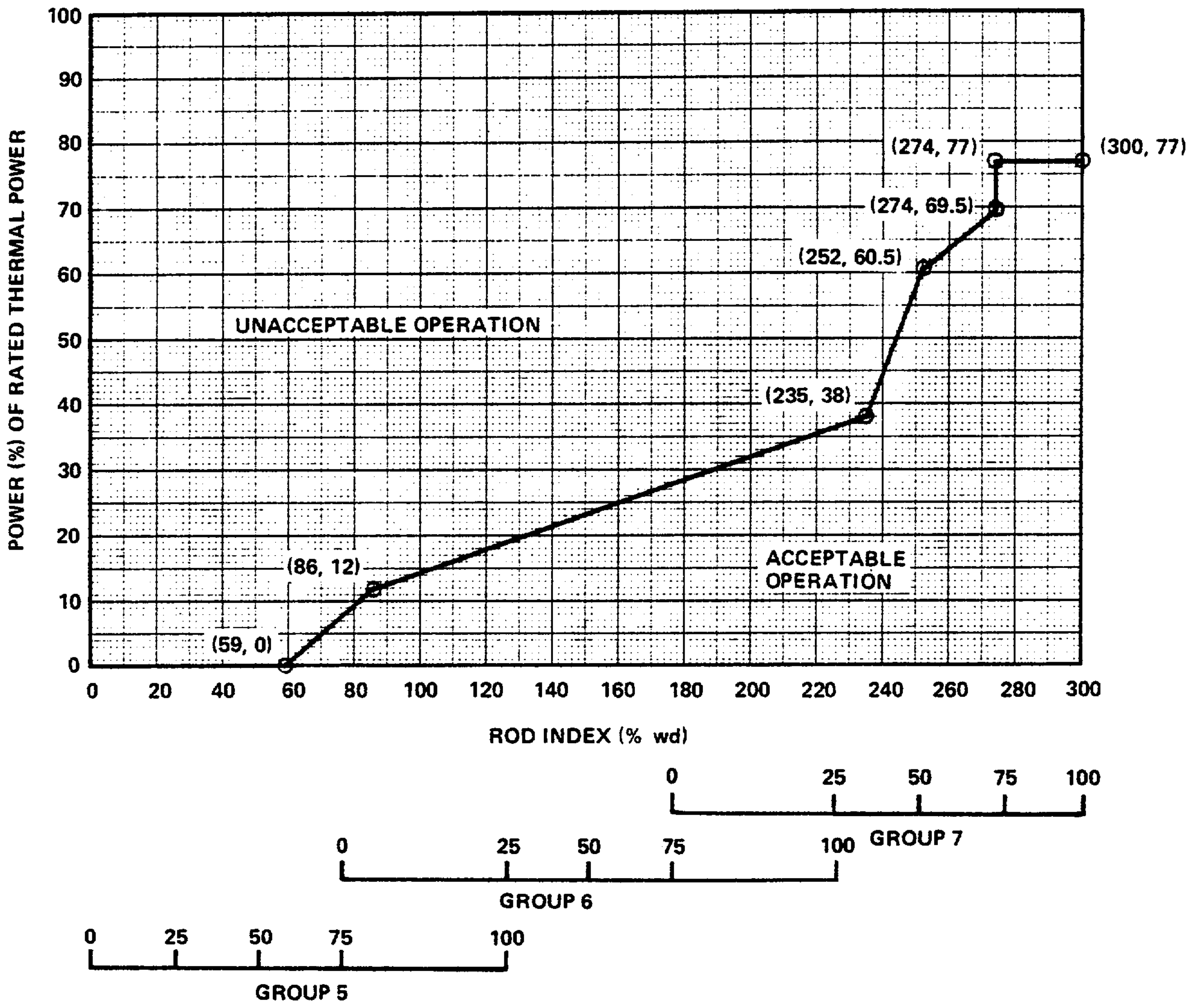



Figure 3.1-3b Regulating Rod Group Insertion Limits for Operation After 145± 5 EFPD (Three Pumps)

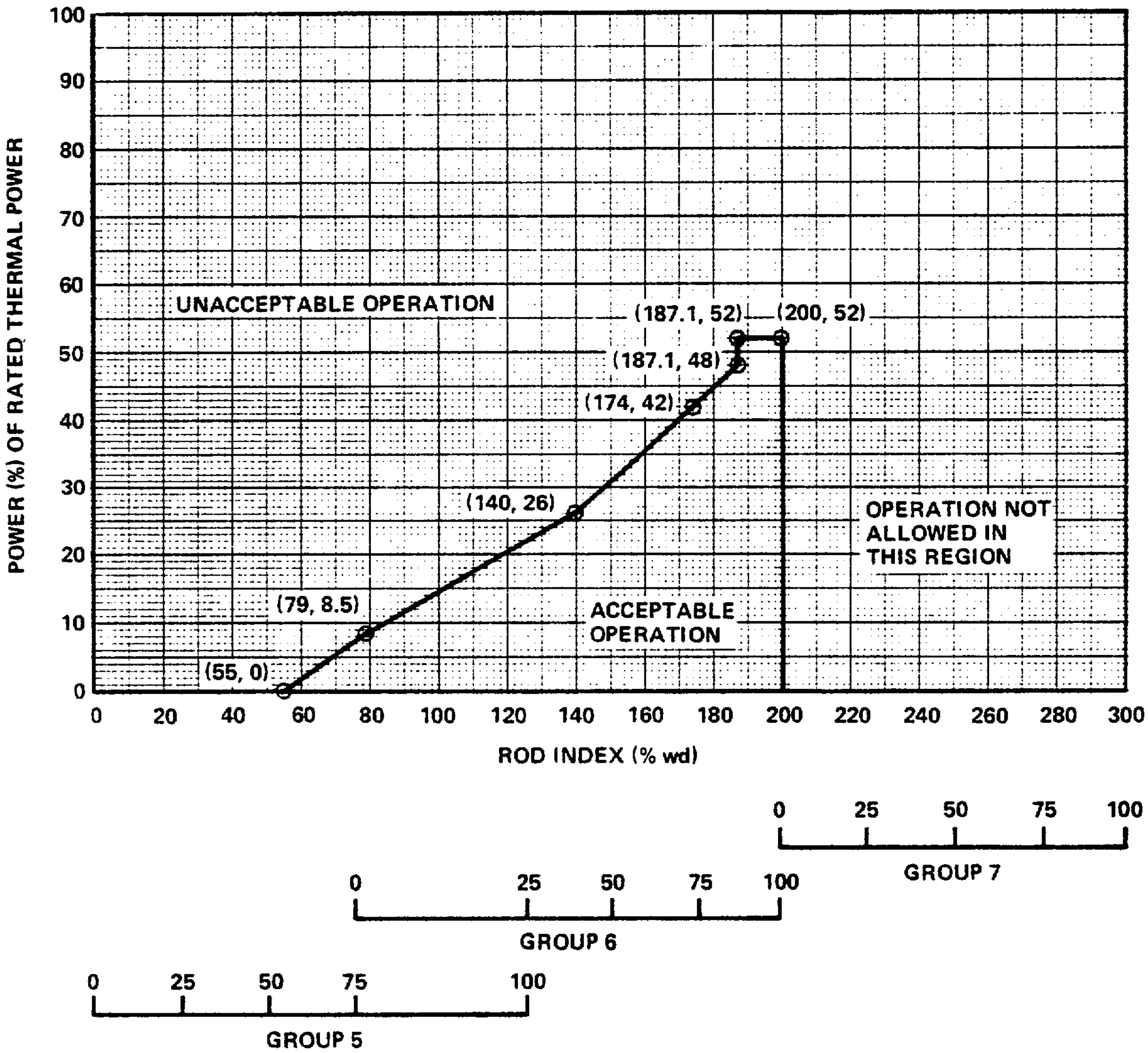



Figure 3.1-3c Regulating Rod Group Insertion Limits for Operation to 145± 5 EFPD (Two Pumps)

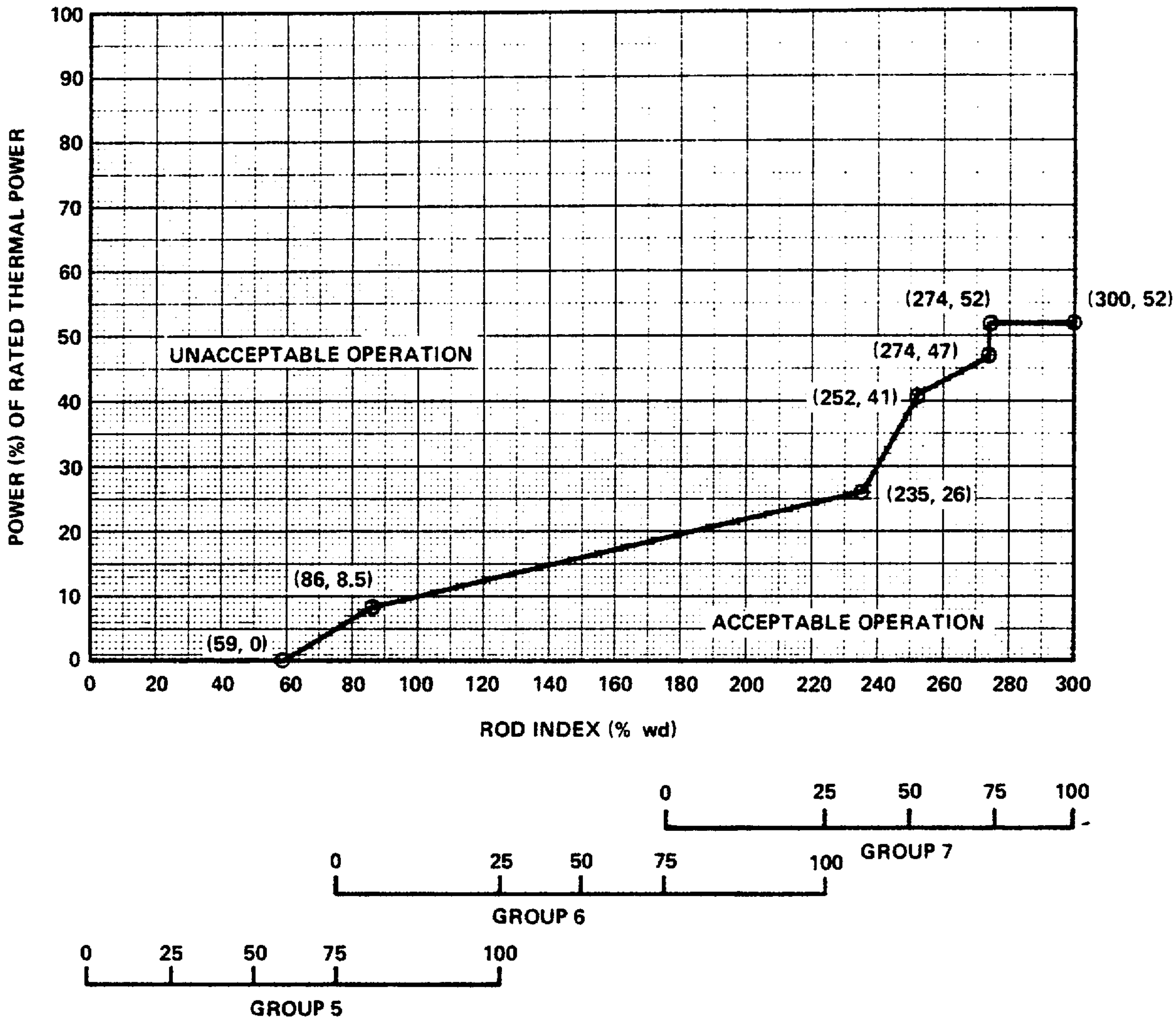



Figure 3.1-3d Regulating Rod Group Insertion Limits for Operation After 145± 5 EFPD (Two Pumps)

## ROD PROGRAM

### LIMITING CONDITION FOR OPERATION

3.1.3.7 Each control rod (safety, regulating and APSR) shall be programmed to operate in the core position and rod group specified in Figure 3.1-4.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With any control rod not programmed to operate as specified above, be in HOT STANDBY within 1 hour.

### SURVEILLANCE REQUIREMENTS

4.1.3.7

a. Each control rod shall be demonstrated to be programmed to operate in the specified core position and rod group by:

  1. Selection and actuation from the control room and verification of movement of the proper rod as indicated by both the absolute and relative position indicators:

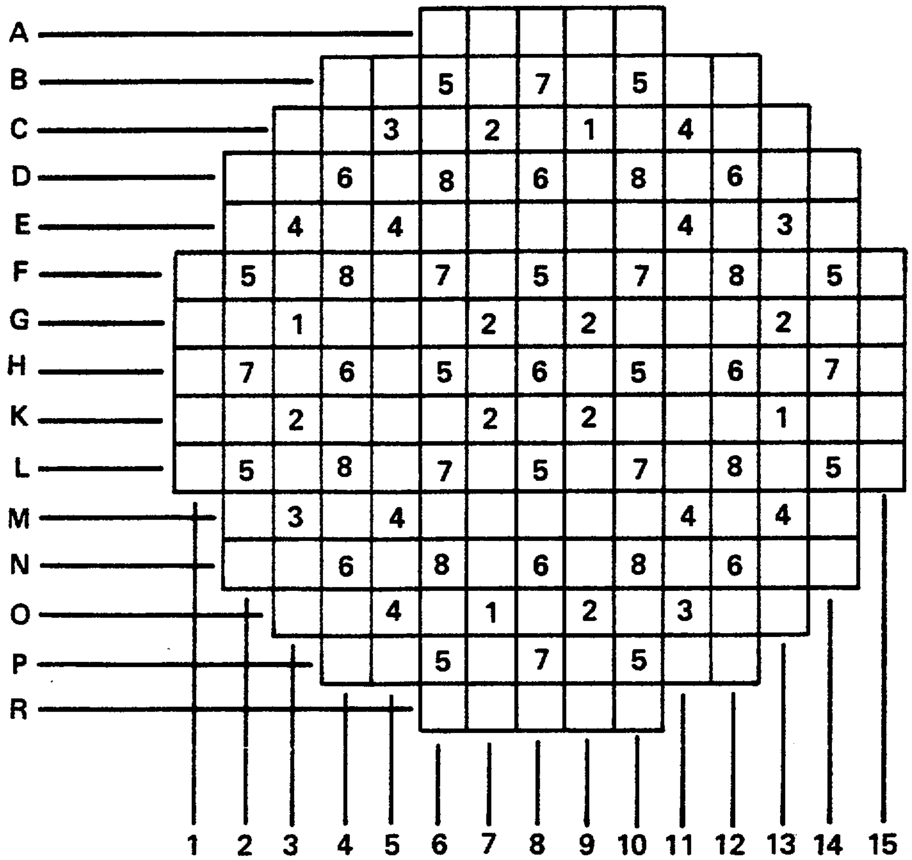
     a) For all control rods, after the control rod drive patches are locked subsequent to test, reprogramming or maintenance within the panels.

     b) For specifically affected individual rods, following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

  2. Verifying that each cable that has been disconnected has been properly matched and reconnected to the specified control rod drive.

b. At least once each 7 days, verify that the control rod drive patch panels are locked.

*See Special Test Exceptions 3.10.1 and 3.10.2.




| Bank | No. Rods | Purpose |
|---|---|---|
| 1 | 4 | Safety |
| 2 | 8 | Safety |
| 3 | 4 | Safety |
| 4 | 8 | Safety |
| 5 | 12 | Regulating |
| 6 | 9 | Regulating |
| 7 | 8 | Regulating |
| 8 | 8 | APSR |

Figure 3.1-4

Control Rod Core Location and
Group Assignments for Modified Cycle 1

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## AXIAL POWER IMBALANCE

### LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1, 3.2-2 and 3.2-3.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION:

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a. Restore the AXIAL POWER IMBALANCE to within its limits within 15 minutes, or

b. Be in at least HOT STANDBY within 2 hours.

### SURVEILLANCE REQUIREMENTS

4.2.1 The AXIAL POWER IMBALANCE shall be determined to be within limits at least once every 12 hours when above 40% of RATED THERMAL POWER except when the AXIAL POWER IMBALANCE alarm is inoperable, then calculate the AXIAL POWER IMBALANCE at least once per hour.

* See Special Test Exception 3.10.1

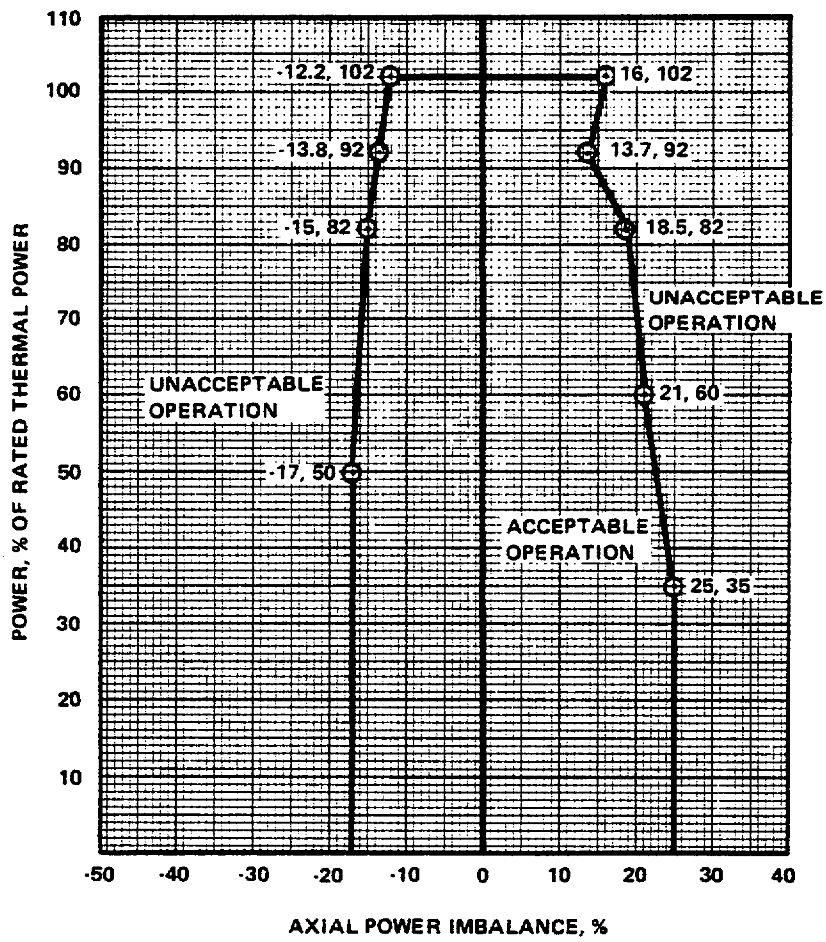



**Figure 3.2-1a AXIAL POWER IMBALANCE Envelope for Operation to 145$\pm$5 EFPD (Four Pumps)**

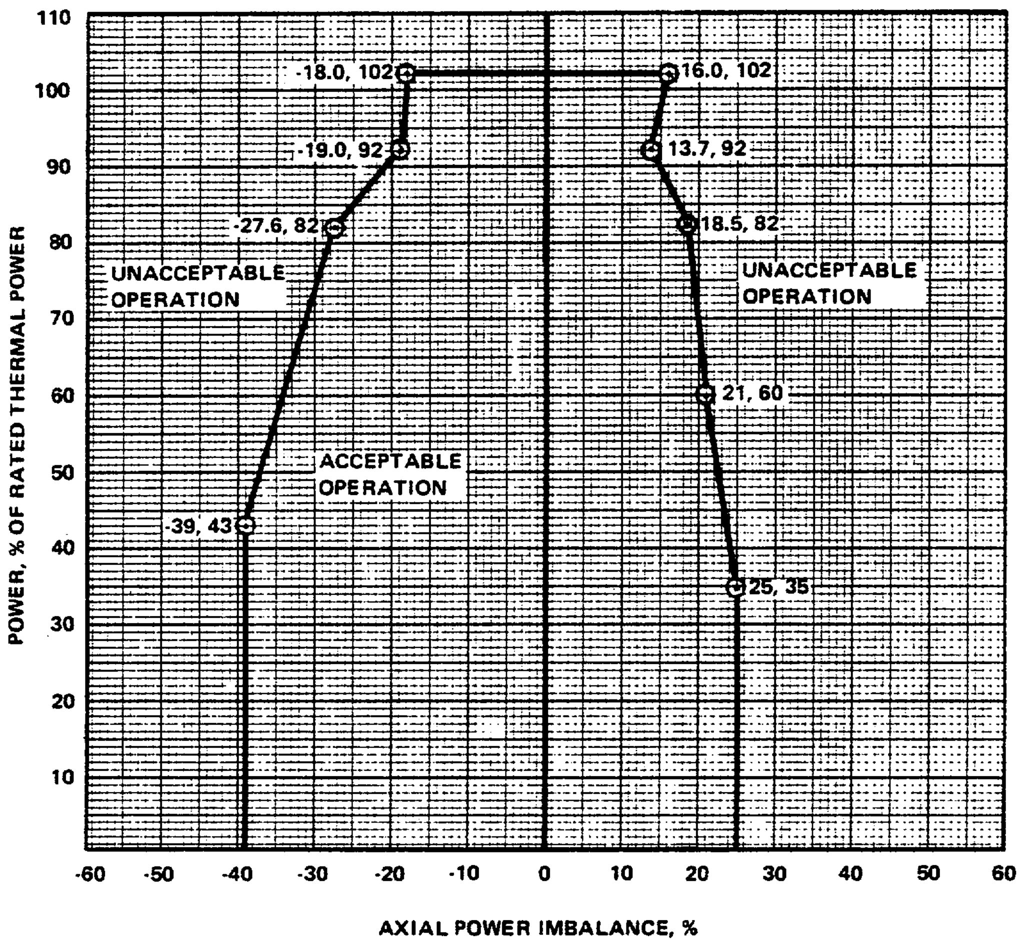



Figure 3.2-1b AXIAL POWER IMBALANCE Envelope for Operation After 145± 5 EFPD (Four Pumps)

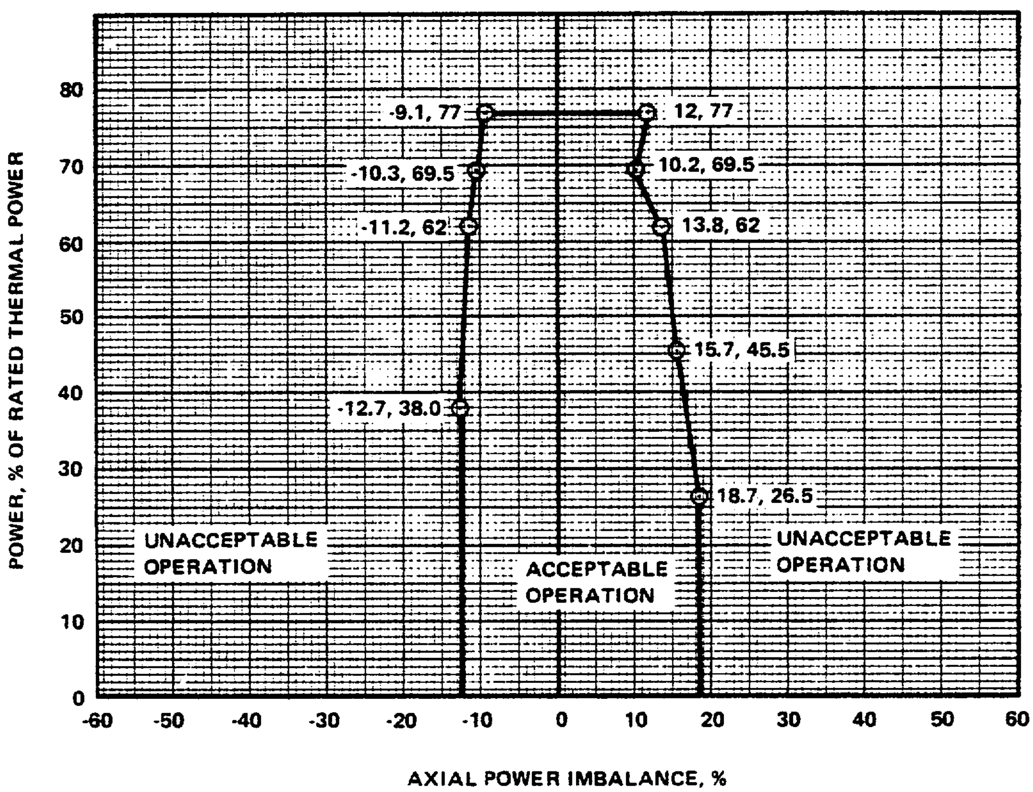



**Figure 3.2-2a AXIAL POWER IMBALANCE Envelope for Operation to 145 ± 5 EFPD (Three Pumps)**

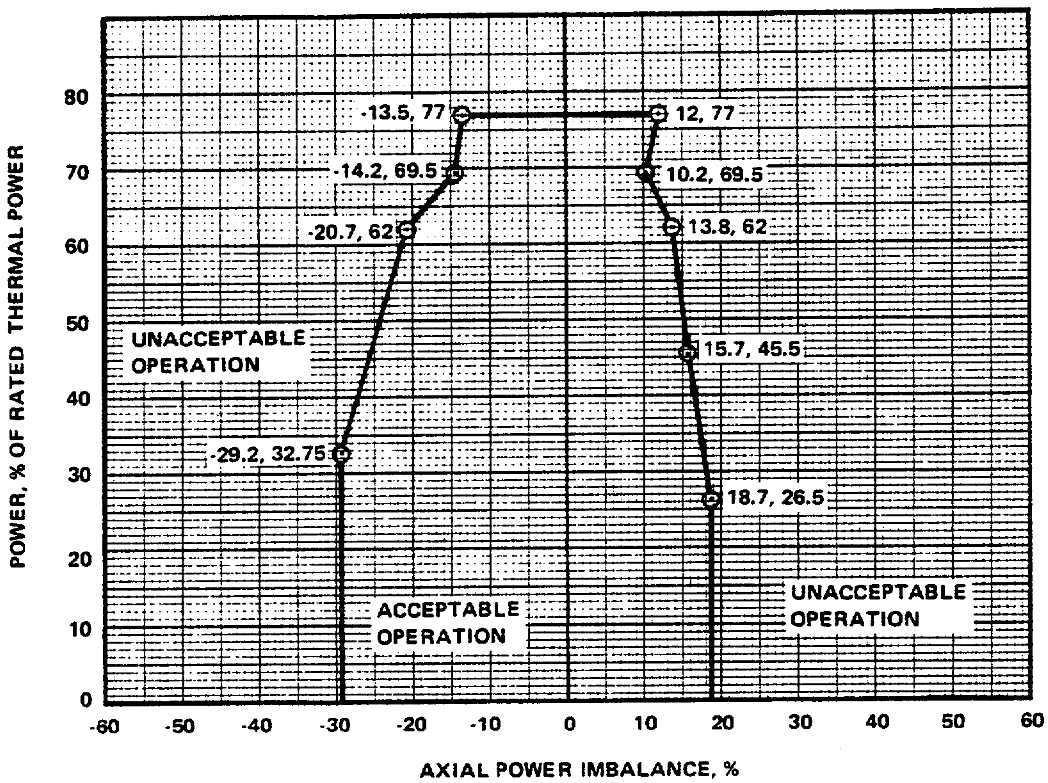



**Figure 3.2-2b AXIAL POWER IMBALANCE Envelope for Operation After 145$\pm$5 EFPD (Three Pumps)**

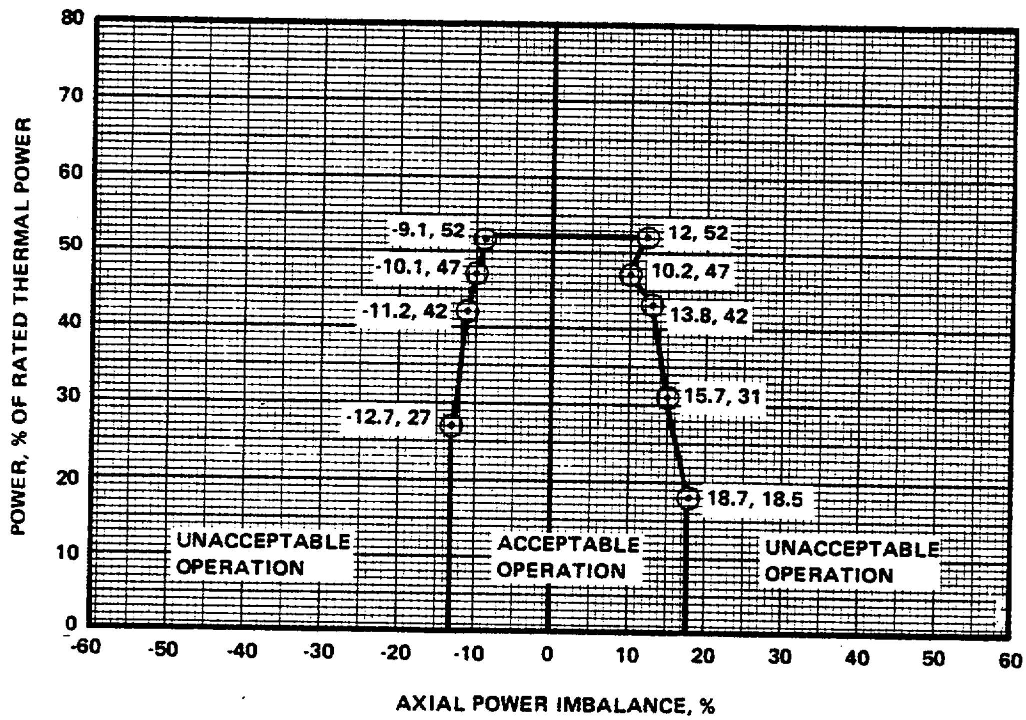



Figure 3.2-3a AXIAL POWER IMBALANCE Envelope for Operation to 145± 5 EFPD (Two Pumps)

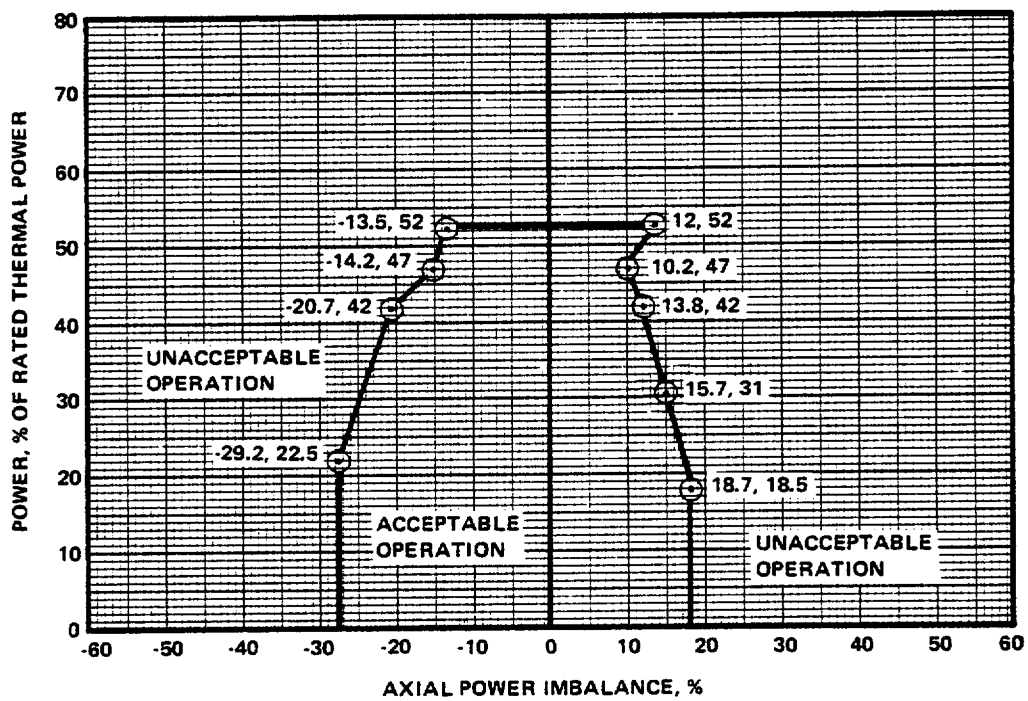



**Figure 3.2-3b** AXIAL POWER IMBALANCE Envelope for Operation After 145± 5 EFPD (Two Pumps)

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With the QUADRANT POWER TILT determined to exceed the Maximum Limit of Table 3.2-2, reduce THERMAL POWER to $\leq$ 15% of RATED THERMAL POWER within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT shall be determined to be within the limits at least once every 7 days during operation above 15% of RATED THERMAL POWER except when the QUADRANT POWER TILT alarm is inoperable, then the QUADRANT POWER TILT shall be calculated at least once per 12 hours.

## TABLE 3.2-2

## QUADRANT POWER TILT LIMITS

| | STEADY STATE LIMIT | TRANSIENT LIMIT | MAXIMUM LIMIT |
|---|---|---|---|
| Measurement Independent QUADRANT POWER TILT | 4.92 | 11.07 | 20.0 |
| QUADRANT POWER TILT as Measured by: | | | |
| Symmetrical Incore Detector System | 3.40 | 8.90 | 20.0 |
| Power Range Channels | 1.96 | 6.96 | 20.0 |
| Minimum Incore Detector System | 1.90 | 4.40 | 20.0 |

## 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 Hot Leg Temperature.

b. Reactor Coolant Pressure

c. Reactor Coolant 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 MARGIN

| Parameter | LIMITS: Four Reactor Coolant Pumps Operating | LIMITS: Three Reactor Coolant Pumps Operating | LIMITS: One Reactor Coolant Pump Operating in Each Loop |
|---|---|---|---|
| Reactor Coolant Hot Leg Temperature $T_H$°F | $\leq$ 611.1 | $\leq$ 611.1[1] | $\leq$ 611.1 |
| Reactor Coolant Pressure, psig.[2] | $\geq$ 2062.7 | $\geq$ 2058.7[1] | $\geq$ 2091.4 |
| Reactor Coolant Flow Rate, gpm[3] | $\geq$ 396,880 | $\geq$ 297,340 | $\geq$ 195,760 |

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These flows include a flow rate uncertainty of 2.5%.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

## 3/4.1.1 BORATION CONTROL

### 3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS $T_{avg}$. The most restrictive condition occurs at EOL, with $T_{avg}$ at no load operating temperature. The SHUTDOWN MARGIN required is consistent with FSAR safety analysis assumptions.

### 3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2800 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2800 GPM will circulate an equivalent Reactor Coolant System volume of 12,110 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

### 3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum $RT_{NDT}$ temperature.

2/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) makeup or DHR pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE emergency busses.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires the equivalent of either 7373 gallons of 8742 ppm borated water from the boric acid storage tanks or 52,726 gallons of 1800 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 434,650 gallons of borated water in the borated water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the

## 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 $\geq$ 1.32 during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.4 kw/ft during normal operation, and (c) maintaining the peak power density $\leq$ 20.4 kw/ft during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power-imbalance envelope defined in Figures 3.2-1, 3.2-2 and 3.2-3 and the insertion limit curves, Figures 3.1-1 and 3.1-3 are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power-imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power-imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined by Figures 3.1-1 and 3.1-3 and if the steady-state limit QUADRANT POWER TILT exists. Additional conservatism is introducted by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

e. Potential fuel rod bow effects.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

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

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

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

$$F_Q \leq 2.94; \quad F^N_{\Delta H} \leq 1.71$$

Power Peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 6.5% (indicated position) from the group average height.

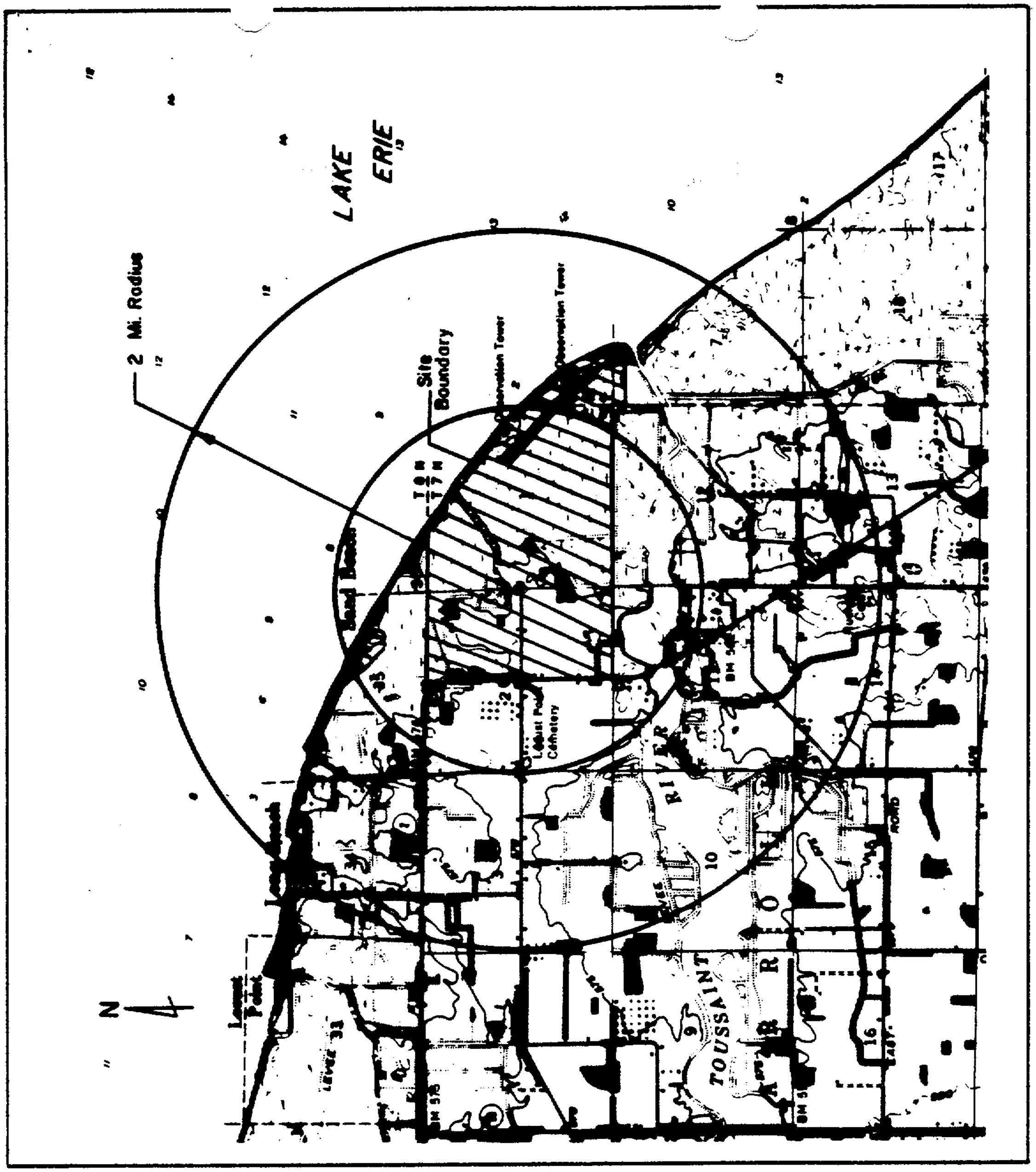
2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained between the limits specified in Specification 3.2.1.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine $F_Q$ and $F^N_{\Delta H}$:

a. The measurement of total peaking factor, $F_Q^{Meas}$, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.




DAVIS-BESSE NUCLEAR POWER STATION
LOW POPULATION ZONE
FIGURE 5.1-2

## DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 40 psig and a temperature of 264°F.

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2500 grams uranium. The initial core loading shall have a maximum enrichment of 3.0 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

### CONTROL RODS

5.3.2 The reactor core shall contain 53 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber manterial. The APSR's shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-346

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 11 to the Facility Operating License No. NPF-3, issued to the Toledo Edison Company and the Cleveland Electric Illuminating Company, for operation of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) located in Ottawa County, Ohio. The amendment is effective as of the date of its issuance.

This amendment revises the Technical Specifications to reflect plant operation at full-rated power (2772 Megawatts-thermal) with the burnable poison rod assemblies and and orifice rod assemblies (except two) removed from the core.

Also, this amendment deletes license condition 2.C.(3)(i) from the operating license No. NPF-3 which specified the penalties for the effects of fuel rod bowing on the departure from nucleate boiling.

The amendment also revises a technical specification regarding a change in the alarm setpoints on quadrant tilt.

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The application for the amendment 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 amendment.

The Commission has determined that the granting of this relief will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

For further details with respect to this action, see (1) the application for amendment dated April 10, 1978, as supplemented May 17, May 26, and June 2, 1978, and (3) the application for amendment dated May 18, 1978 as supplemented May 26, June 2, June 7, June 8, and June 13, 1978, (3) Babcock and Wilcox Report, BAW-1496, May 1978, (4) Amendment No. 11 to License NPF-3, and (5) the Commission's related Safety Evaluation Report. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, DC 20555 and at the Ida Rupp Public Library, 310 Madison Street, Port Clinton, Ohio 43452. A copy of items (4) and (5) may be obtained upon request

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addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland, this 16 day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION:

Original Signed by
John F. Stolz
John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

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| DATE | 6/14/78 | 6/14/78 | 6/16/78 | 6/16/78 | | |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

## INTRODUCTION

By letter dated April 10, 1978, the Toledo Edison Company requested a change in the Technical Specifications for the "DNB Margin" reactor coolant flow rates to accommodate the Departure From Nucleate Boiling Ratio (DNBR) penalty, as specified in license condition 2.C.(3)(i) of Facility Operating License NPF-3.

The proposed change involved balancing the Fuel Rod Bowing Penalty of 11.2 percent (described in Section 4.4 of Supplement No. 1 to our Safety Evaluation Report) by taking credit for: (1) a 1 percent DNBR credit for the Flow Area Reduction Factor; (2) a 1.1 percent credit for the DNBR Power Spike Factor, and (3) a 9.8 percent DNBR credit for increasing the required reactor coolant flow by 5 percent.

In addition, by letter dated May 18, 1978, the Toledo Edison Company requested changes in the Technical Specifications because of removal of the Burnable Poison Rod Assemblies (BPRA) following evidence of wear of the hold down devices for the BPRA's. On May 26, 1978, the Toledo Edison Company revised their May 18, 1978 request to include changes in the Technical Specifications because of the mechanical wear also observed on the Orifice Rod Assemblies (ORAs). The Toledo Edison Company stated that it was prudent to remove all BPRAs and all but two of the ORAs from the core internals of Davis-Besse, Unit 1 before the completion of the first cycle of operation to avoid the possible damage to the plant from a potential failure of the hold down devices.

The removal of the BPRAs and ORAs result in changes in various nuclear parameters, as well as resulting in an increase in core bypass flow. Changes to the Technical Specifications are required as a result of changes in the nuclear parameters, as well as an increase in core bypass flow.

The Toledo Edison Company provided, as an attachment to their letter of May 18, 1978, the Babcock and Wilcox document, BAW-1489, "Application to Amend Operating License for Removal of Burnable Poison Rod Assemblies - Davis-Besse Nuclear Generating Station, Unit 1," and by their letter of May 26, 1978 provided BAW-1489, Revision 1, "Application to Amend Operating

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License for Removal of Burnable Poison Rod and Orifice Rod Assemblies - Davis-Besse Nuclear Generating Station, Unit 1." BAW-1489 and BAW-1489, Revision 1 provided analyses supporting the proposed changes to the Technical Specifications.

Both requests for application to revise the Technical Specifications (i.e. the letters of April 10, 1978 and May 18, 1978, as supplemented) required staff evaluation of the thermal hydraulic design and the accuracy of the observed reactor coolant flow rate in excess of design flow rate. Therefore, our evaluation of both requests for changes to the Technical Specifications interface closely and are provided in the discussion and evaluation provided below.

DISCUSSION

BPRAs are used in the first cycle of B&W reactors to control part of the initial excess reactivity and to flatten the radial power distribution. The reactivity controlled by burnable poison reduces the amount which must be controlled by soluble boron and prevents the occurrence of a positive moderator coefficient above 95 percent of full power. The Davis-Besse, Unit 1 reactor has achieved a first cycle burnup of 87 Effective Full Power Days (EFPDs) and some of the burnable poison has been burned out. However, sufficient burnable poison remains to require core changes in order to offset the effect of its removal. These core changes were:

1. Interchange of four intermediate (2.63 w/o) enrichment bundles near the center of the core with 4 low (1.98 w/o) enrichment bundles near the core periphery.

2. Rearrangement of the control rod groupings and decoupling of group 7 from the withdrawal sequence. In the regrouping, control rod group 7 has been shifted toward the periphery and remains in the core until a burnup of 145 EFPDs has been reached. This arrangement serves to further flatten the radial power distribution and to replace some of the fixed poison in the core and thus prevent the moderator coefficient from becoming positive.

The Toledo Edison Company has performed an analysis of the modified core, assuming that the modification occurred at 80 EFPD and that the cycle length is increased from 433 to 485 EFPD. The analysis was performed using the same calculational methods and techniques that have been employed in the design of other B&W reactors--including Davis-Besse, Unit 1. The core physics parameters have been calculated for the modified cycle--80 to 145 EFPD with groups 5 and 6 partially inserted into the core and group 7 completely inserted followed by 145 to 485 EFPD with groups 5 through 7 nearly out of the core. The recalculated parameters included shutdown

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margins, rod bank worths, ejected and dropped rod worths, stuck rod worth, Doppler coefficient, moderator coefficient, xenon worth, boron worth, and critical boron concentration.

During removal of the BPRAs it was discovered that sufficient wear was present on the holddown devices for the orifice rod assemblies (ORAs) to warrant their removal. By letter dated May 26, 1978, the Toledo Edison Company submitted Revision 1 to BAW-1489 to encompass the removal of the ORAs from the Davis-Besse, Unit 1 core.

All of the ORAs will be removed with the exception of two modified orifice rod assemblies which are used with a primary neutron source. The removal of the ORAs increases the flow through the guide tubes but does not significantly alter the physics parameters. Thus, the analyses presented in BAW-1489 remain in effect.

EVALUATION

We have reviewed the information presented in BAW-1489 for the values of the physics parameters and core flow and their effect on the safety analyses for Davis-Besse, Unit 1. For the rod withdrawal transients at full and zero powers, the control rod misoperation transient, the rod ejection accident, the moderator dilution transient, cold water accident, steam line failure accident, loss-of-coolant accident, and loss-of-normal-feedwater transient, the significant parameters are shown to be bounded by those used in the Final Safety Analysis Report analysis. Thus, the consequences of these transients and accidents will not be greater than those described in the Final Safety Analysis Report.

The loss of electric power transient and the steam generator tube failure are independent of the significant parameter changes and the Final Safety Analysis Report analyses are, therefore, applicable for these transients.

By letter dated June 8, 1978 the Toledo Edison Company submitted Revision 2 to BAW-1489 providing a revised B&W analysis for the loss of flow transient and the feedwater system malfunction transient. The minimum DNBR transient is the one-pump loss-of-flow transient which results in a minimum DNBR of 1.45. It should be noted that the Davis-Besse, Unit 1 Final Safety Analysis Report and BAW-1489 indicated that the most limiting loss-of-flow transient was a four-pump loss of flow transient. The one-pump loss of transient became the most limiting transient when the power imbalance/flow reactor trip was adjusted to decrease inadvertent power imbalance/flow reactor trips. This trip adjustment was made prior to operation of Davis-Besse, Unit 1. It should also be pointed out that incorporating margin to compensate for fuel rod bow results in a minimum required DNBR of 1.445, and thus the limiting loss-of-flow transient resulting in a minimum DNBR of 1.45 is acceptable.

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Removal of all the BPRAs and all but two of the ORAs from the core results in a calculated increase of 4.7 percent in the maximum core bypass flow (from 6.04 percent to 10.75 percent). By letter dated April 10, 1978, the Toledo Edison Company requested that the minimum allowable reactor coolant flow be increased by 5 percent over the Final Safety Analysis Report design flow to compensate for the potential effects of fuel rod bowing. Therefore, modified operating conditions have been proposed to compensate for both the increased bypass flow and the potential effects of rod bow on the core thermal safety margin. An analysis has been performed, based on a minimum allowable flow rate of 110 percent of design flow and a slightly adjusted trip limit curve (Technical Specification Figure 2.1-1) for reactor coolant core outlet pressure and outlet temperature. The analysis results indicate that operation at the proposed limits with BPRAs and ORAs removed would not result in violation of acceptable fuel design limits. Reactor coolant system flow measurements have indicated an actual system flow rate of at least 113 percent of the previous limit (measurement errors not included).

In a B&W-designed Nuclear Steam Supply System (NSSS), Gentile flowmeters are used to measure Loop 1 and Loop 2 reactor coolant flow rates (B&W NSSS have 2 loops with 2 pumps each). These primary loop flowmeters are not calibrated prior to installation. Loop 1 and 2 feedwater flow rates are measured with calibrated flowmeters and a plant heat balance is used to calibrate the Gentile flow-meters.

The total reactor coolant flow rate for Davis-Besse, Unit 1, as determined from a plant heat balance, is 113.2 percent of the design flow rate. Based on the accuracies of primary and secondary side measurements reported in Table 1, the licensee calculated the reactor coolant flow rate accuracy to be ± 2.2 percent.

Measurement accuracies for primary and secondary side measurements used for calculation of reactor coolant flow rate are shown in Table 1. Except for the pressure uncertainty and flow $\Delta P$ uncertainty, these values are reasonable and consistent with industry practice. The most significant terms in calculating accurate values of reactor low rate are reactor coolant temperatures and feedwater flowmeter differential pressures.

The measurement accuracy reported for reactor coolant pressure is ±0.77 percent; however, staff experience indicates a ±1 percent is more reasonable. The change to ±1 percent pressure measurement accuracy does not affect the final reactor coolant flow accuracy as given to 3 significant digits.

TABLE 1

ACCURACY OF PRIMARY AND SECONDARY SIDE MEASUREMENTS

USED FOR CALCULATION OF TOTAL RC FLOWRATE

| PARAMETER | MEASUREMENT ACCURACY - PERCENT | SPAN | ACCURACY UNITS |
|---|---|---|---|
| RC hot leg temp. | ± 0.79 | 520 to 620F[a] | ± 0.79 F |
| RC cold leg temp. | ± 0.79 | 520 to 620F | ± 0.79 F |
| Steam temp. | ± 0.60 | 0 to 700F | ± 4.2 F |
| Feedwater temp | ± 1.13 | 0 to 600F | ± 6.8 F |
| Feedwater pressure | ± 1.0 | 0 to 1500 psig[b] | ± 15 psi[c] |
| Steam pressure | ± 1.89 | 0 to 1200 psig | ± 23 psi |
| RC pressure | ± 0.77 | 0 to 2500 psig | ± 19 psi |
| Feedwater Flow | ± 1.25 | 0 to 960 inches (Std. $H_2O$) | ± 12. inches |
| RC Flowrate | ± 1.046 | 0 to 910 inches (Std. $H_2O$) | ± 9.5 inches |

a = Temperature in degrees Fahrenheit

b = Pressure in pounds per square inch gauge

c = Pressure in pounds per square inch

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The measurement accuracy reported for reactor coolant flow rate $\Delta P$ ($\pm 1.046$ percent) is for the $\Delta P$ transmitter only. It is our position that a drift allowance for the flow element (Gentile tube) is also needed. Therefore, the staff has reevaluated the reactor coolant flow measurement accuracy, using a value of $\pm 2$ percent for the reactor coolant flow rate $\Delta P$ measurement. The effect of this change is to increase the total flow rate measurement accuracy from $\pm 2.2$ percent to $\pm 2.5$ percent.

An important element in the error analysis is the assumed independence of the uncertainties in measurement of feedwater flow for the two loops. The major potential source of dependency for the feedwater flow measurement uncertainties is crud buildup in the flow elements. Although crud buildup has been observed in the feedwater venturi's for at least one reactor vendor, the once-through steam generator feedwater chemistry control minimizes the increase of contaminants into the system and the buildup of crud on the flow elements for Davis-Besse, Unit 1. Therefore, it is reasonable to assume that the feedwater flow measurement accuracies are independent.

Flow requirements given in Table 3.2-1 of the proposed Technical Specification revision, as provided in Revision 1 to BAW-1489, included a measurement uncertainty of $\pm 2.2$ percent factored into the 110 percent design flow required for potential rod bow effects and increased bypass flow. Based on our determination that the measurement accuracy is $\pm 2.5$ percent, the Technical Specification, Table 3.2.1, has been revised to reflect the increase in total flow rate measurement accuracy from $\pm 2.2$ percent to $\pm 2.5$ percent.

Based on our calculations of bypass flow through the guide tubes with the BPRAs and ORAs removed, we have determined that an increase in the reactor vessel flow of 5 percent is sufficient to compensate for the increased bypass flow.

Also, we have reviewed and evaluated the Toledo Edison Company's request for balancing the Fuel Rod Bowing Penalty of 11.2 percent by taking credit for: (1) a 1 percent DNBR credit for the Flow Area Reduction Factor (as described in Section 4.4 of the Davis-Besse, Unit 1 Final Safety Analysis Report); (2) a 1.1 percent credit for the DNBR Power Spike Factor (as described in approved Topical Report BAW-1401); and a 9.8 percent DNBR credit for the effects of a 5 percent increase in reactor coolant flow.

We have reviewed each of these credits and find them acceptable. Also, we have determined that the revised Technical Specification on DNBR Margin with the allowed measurement uncertainty of 2.5 percent provides assurance that the treatment of the DNBR penalty required to address the effect of

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fuel rod bowing, as specified in License Condition 2.0.(3)(i) is acceptable. Therefore, we find that License Condition 2.C.(3)(i) is no longer necessary and can be removed from Facility License NPF-3.

We have also reviewed the modified orifice rod assembly (MORA) for acceptability. A MORA is a standard ORA modified for use with a primary neutron source. During the initial core operation of Davis-Besse, Unit 1, two primary neutron sources are located in individual guide tubes of two fuel assemblies. Each source is held in a shroud tube which rests on the bottom of a guide tube. A solid stainless steel rod is placed on top of the source to hold it down against hydraulic lift. To provide further assurance that the source will not come out of the guide tube during postulated accidents, and ORA is latched to the top of the fuel assembly. The rods of the ORA plug the top of each guide tube, including the guide tube containing the source.

To prevent the MORA from causing wear of the fuel assembly end fitting and coming loose, the Toledo Edison Company proposed to modify the primary source capturing arrangement. First, 12 of the rods in each of the two ORAs remaining in the core are being removed, leaving only the rod above the source and the 3 symmetrically-located rods. Secondly, a retainer is to be placed over the hub of the modified ORA and held down by the reactor internals.

The design and testing of this retainer device are described in the Babcock and Wilcox Report, BAW-1496, "BPRA Retainer Design Report," May, 1978. From a mechanical design standpoint, the basic concern is whether the retainer provides enough holddown force to preclude loosening of the MORAs. From analyses of the static and dynamic stresses on the retainer spring load arm and housing, results of prototype testing in a flow test facility, and in-air mechanical tests, criteria for use of the BPRA retainer device with modified ORAs have been established. The primary criterion is that the margin to component lift with the retainer, taking into account the hydraulic forces acting on the MORA, the MORA weight, and the retainer holddown force, should be greater than 30 pounds. This criterion is met with acceptable margin by the fact that when the retainer device is used with the modified ORA, the holddown force is greater than 35 pounds with all 4 reactor coolant pumps operating. A second criterion, which is related to fuel assembly growth, is based on a fuel assembly burnup design value that is used as a basis for the retainer design. Since the maximum burnup used in one cycle of operation will be less than the burnup used as a design basis, the fuel assembly growth criterion is met (note that the retainer will be used for only one cycle of operation).

Although failure is considered unlikely, the potential consequences of a retainer failure have also been addressed in a letter from J. Taylor (B&W) to S. Varga (NRC), dated June 7, 1978. The neutronic and thermal-hydraulic consequences are considered insignificant. Interference with control rod motion, for example, would not, according to analyses of stuck-out control rod transients for B&W 177-FA plants, prevent safe shutdown of the plant.

The major concern associated with retainer failure is plant damage and potential outages for repair. This damage should be precluded by the Loose Parts Monitoring System (LPMS). The LPMS is designed to detect a failed retainer in either the reactor vessel or steam generator. Even though the BPRA retainer is designed for only one cycle of operation, B&W has stated that it will recommend that surveillance inspections be made following retainer use. This should provide additional confirmation of acceptable operation. B&W has also stated that definite plans regarding surveillance will be provided to NRC as they are formulated.

In summation, we conclude that, based on (1) analyses and test results on the BPRA retainer device, (2) establishment and meeting of criteria for use of the device with ORAs modified for use with primary neutron sources in Davis-Besse, Unit 1, (3) analyses which indicate that failure of the retainers, however unlikely, would not prevent plant safe shutdown and (4) failure detection capability of the Loose Parts Monitoring System, there is reasonable assurance that the proposed use of the BPRA retainer with two MORAs in Davis-Besse, Unit 1 will pose no significant safety concern.

Because of the modification of core loading, some changes have been made in power distributions in the core. These changes necessitate changes in the technical specifications. Further changes are necessitated by the reprogramming of the rod groups.

The new technical specifications have been established, using procedures which have been previously employed. New safety limits (Spec. 2.1.2) and Trip Setpoints (Fig. 2.2-1) and Allowable Values (Fig. 2.2-2) have been specified. New rod insertion limits (Spec. 3.1.2.6) have been specified along with new axial imbalance limits (Spec. 3.2.1) to ensure that peaking factor limits used as input to the LOCA-ECCS analysis are not exceeded. The rod program description has been changed (Spec. 3.1.7) to reflect the modification in group assignments. The maximum boration capability requirements (page B3/4 1-2) has been changed to reflect the reactivity changes resulting from the removal of the BPRAs and the relocation of the fuel assemblies.

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The procedures used to establish the technical specifications on power distribution limits have been previously reviewed and approved. Based on this review and approval, we find the technical specification changes described above to be acceptable.

A further technical specification change, unrelated to the core modification, was requested in Toledo Edison Company's letter of May 18, 1978. This request concerns the modification of alarm setpoints on quadrant tilt to accommodate a recently-discovered increase in the measurement error associated with this quantity. The original uncertainty evaluation was performed in 1974, based on data obtained from prototype detectors. Observations of anomalies in operating reactors led to the reevaluation of this error. B&W has submitted (letter, Taylor to Reid, dated May 11, 1978) a document describing the methods used to perform the statistical analysis of the uncertainties and giving revised quadrant tilt alarm setpoints for Davis-Besse, Unit 1. We have reviewed the document and conclude that the analysis method is acceptable. We have not reviewed the data base used to obtain numerical results but we know of no data that would make the application of the method to Davis-Besse, Unit 1 nonconservative. We, therefore, find the revised alarm setpoints on quadrant tilt to be acceptable.

The Toledo Edison Company, in their submittals of May 18 and May 26, 1978, stated that, after completion of the core modifications, startup tests will be performed to assure that the various physics parameters are bounded by those in the Final Safety Analysis Report for Davis-Besse, Unit 1. Tests will be performed on rod drop times, critical boron concentration, temperature coefficients, control rod worths, power distributions, and power coefficients. Successful completion of tests at each power level will be required before proceeding to the next higher power level.

We reviewed the low power physics tests and startup tests proposed by the Toledo Edison Company and requested that additional tests be completed. By letters dated June 8 and June 13, 1978, the Toledo Edison Company committed to the additional low power physics tests and startup tests which we requested.

Based on the use of approved calculational methods, and on the augmented low power physics tests and startup tests, which we have reviewed and found acceptable, we find the analysis of the physics parameters of the core modification to be acceptable. We have also determined that the actual reactor coolant system flow exceeds the design flow by an amount sufficient to not only compensate for the increased bypass flow due to the removal of all the BPRAs and all but two of the ORAs, but also, that the excess flow is sufficient to accommodate the thermal margin

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required to address the rod bow effects on the departure from nucleate boiling ratio. We have also determined that the two modified ORAs to be used at Davis-Besse, Unit 1 will pose no significant safety concern. In addition, our review and evaluation had determined that revised limits necessary for safe operation have been incorporated in the revised Technical Specification. Therefore, we find that Davis-Besse, Unit 1 can be operated safely for the duration of Cycle No. 1 without BPRAs and ORAs at the rated core power level of 2772 Megawatts-thermal.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered or a significant decrease in any safety margin, it does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. Also, we reaffirm our conclusions as otherwise stated in our Safety Evaluation Report.

Dated: JUN 16 1978

minor changes

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Docket No. 50-346

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SUBJECT: **NOTICE OF ISSUANCE OF AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NO. NPF-3 FOR DAVIS-BESSE, UNIT 1**

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies ( 15 ) of the Notice are enclosed for your use.

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P.S. An extra copy of the Amendment Package is enclosed for the NRC PDR.

Office of Nuclear Reactor Regulation

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