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Docket No. 50-346

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

The Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consists of changes to the license and the Technical Specifications in response to your applications dated July 13, 1979 and February 11, 1980, as revised and supplemented.

This amendment modifies the Technical Specifications to permit operation for a second cycle and to reflect changes in the reactor coolant system high pressure and power operated relief valve setpoints. The amendment also deletes three satisfied License Conditions, 2.C.(3)(e), 2.C.(3)(f) and 2.C.(3)(g).

Two changes to the Technical Specifications requested in your February 11, 1980 submittal are not included in this amendment since no bases were provided. These are changes to the Action Statements for Sections 3.2.1 and 3.2.5. We will act upon your request to modify these Action Statements upon receipt of your bases.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 33
2. Safety Evaluation
3. Notice

8010210087

cc w/enclosures: See next page

Concur in amendment and Fed. Reg. notice only

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	STSG	OELD
SURNAME	RIngram	DGarner/cb	RReid	TNovak		M. Rothschild
DATE	9/19/80	9/19/80	9/19/80	9/19/80	9/ /80	9/24/80

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

October 2, 1980

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Docket No. 50-346

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: DAVIS-BESSE UNIT NO. 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 (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment No. 33
Referenced documents have been provided PDR

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#4:DL					
SURNAME →	RIngram/cb					
DATE →	10/3/89					

NRC FORM 102 (1-76)

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

October 1, 1980

Docket No. 50-346

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

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Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 33
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Toledo Edison Company

cc w/enclosure(s):

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Port Clinton, Ohio 43452

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Commissioners of Ottawa County
Port Clinton, Ohio 43452

Attorney General
Department of Attorney General
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Columbus, Ohio 43215

Harold Kahn, Staff Scientist
Power Siting Commission
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Mr. Rick Jagger
Industrial Commission
State of Ohio
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Licensing Engineer
Toledo Edison Company
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U.S. Nuclear Regulatory Commission
Resident Inspector's Office
5503 N. State Route 2
Oak Harbor, Ohio 43449

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

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

cc w/enclosure(s) and incoming dtd.:
2/11/80 & suppls.; 7/13/79

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

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. 33
License No. NPF-3

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

   A. The applications for amendment by the Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) dated February 11, 1980, as revised and supplemented, and July 13, 1979, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

8010210090

2. Accordingly, Facility Operating License No. NPF-3 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   A. Revise paragraph 2.C.(2) to read as follows:

      Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 33 , are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

   B. Delete paragraphs 2.C.(3)(e), 2.C(3)(f) and 2.C.(3)(g).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 1, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 33

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 2-5	
B 2-6	
B 2-8	
3/4 1-26	
3/4 1-28	
3/4 1-28a	
3/4 1-29	
3/4 1-29a	
3/4 1-29b	
3/4 1-29c	
3/4 1-31	
3/4 1-34	(New Page)
3/4 1-35	(New Page)
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3/4 2-1	
3/4 2-2	
3/4 2-2a	
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3/4 2-4a	
3/4 2-12	
3/4 2-14	
3/4 4-1	
3/4 4-4	
3/4 10-1	
3/4 10-2	
B 3/4 1-2	
B 3/4 1-4	
B 3/4 2-1	
B 3/4 2-3	
B 3/4 4-1	
B 3/4 4-1a	

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.

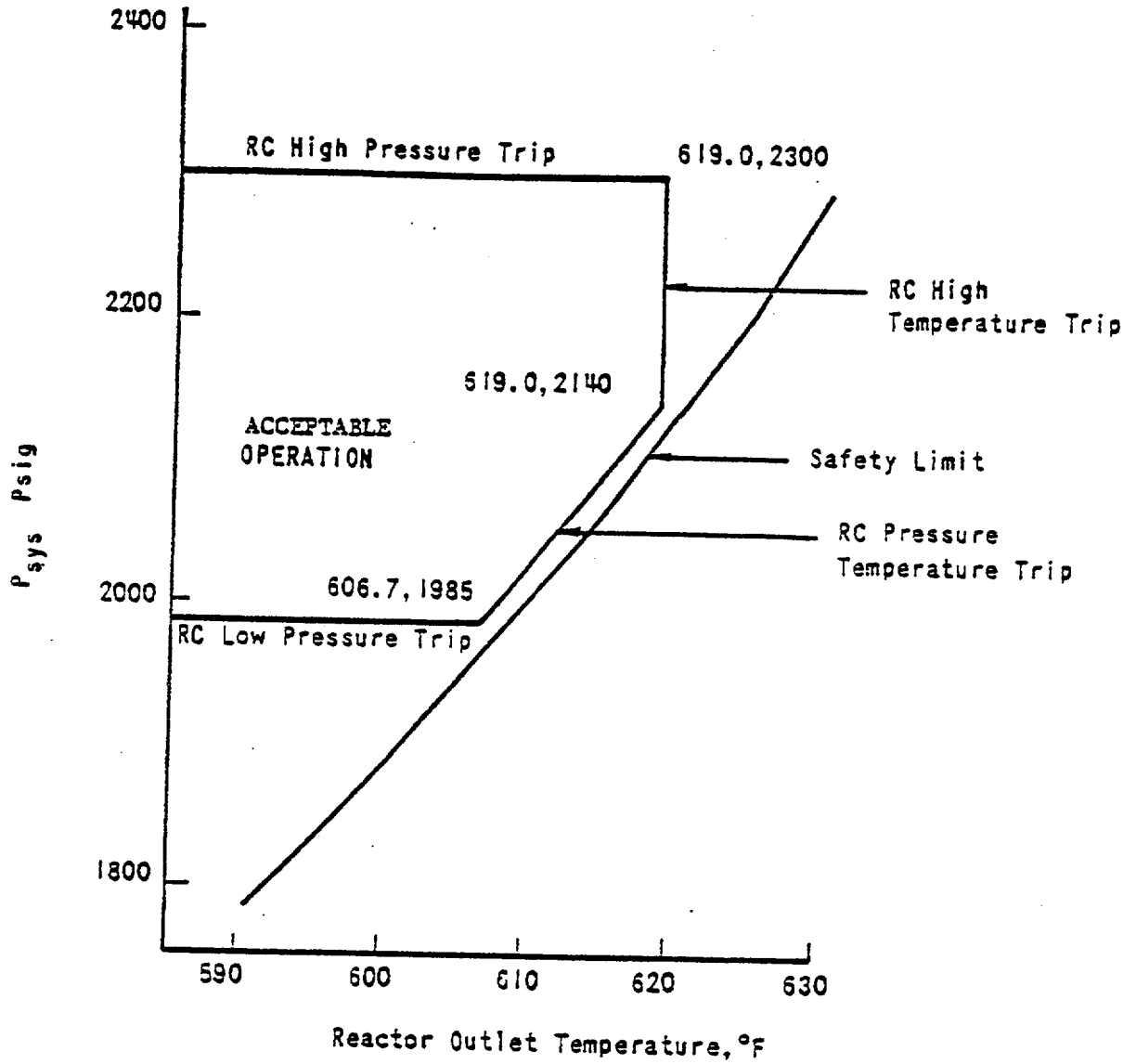


FIGURE 2.1-1 Reactor Core Safety Limit

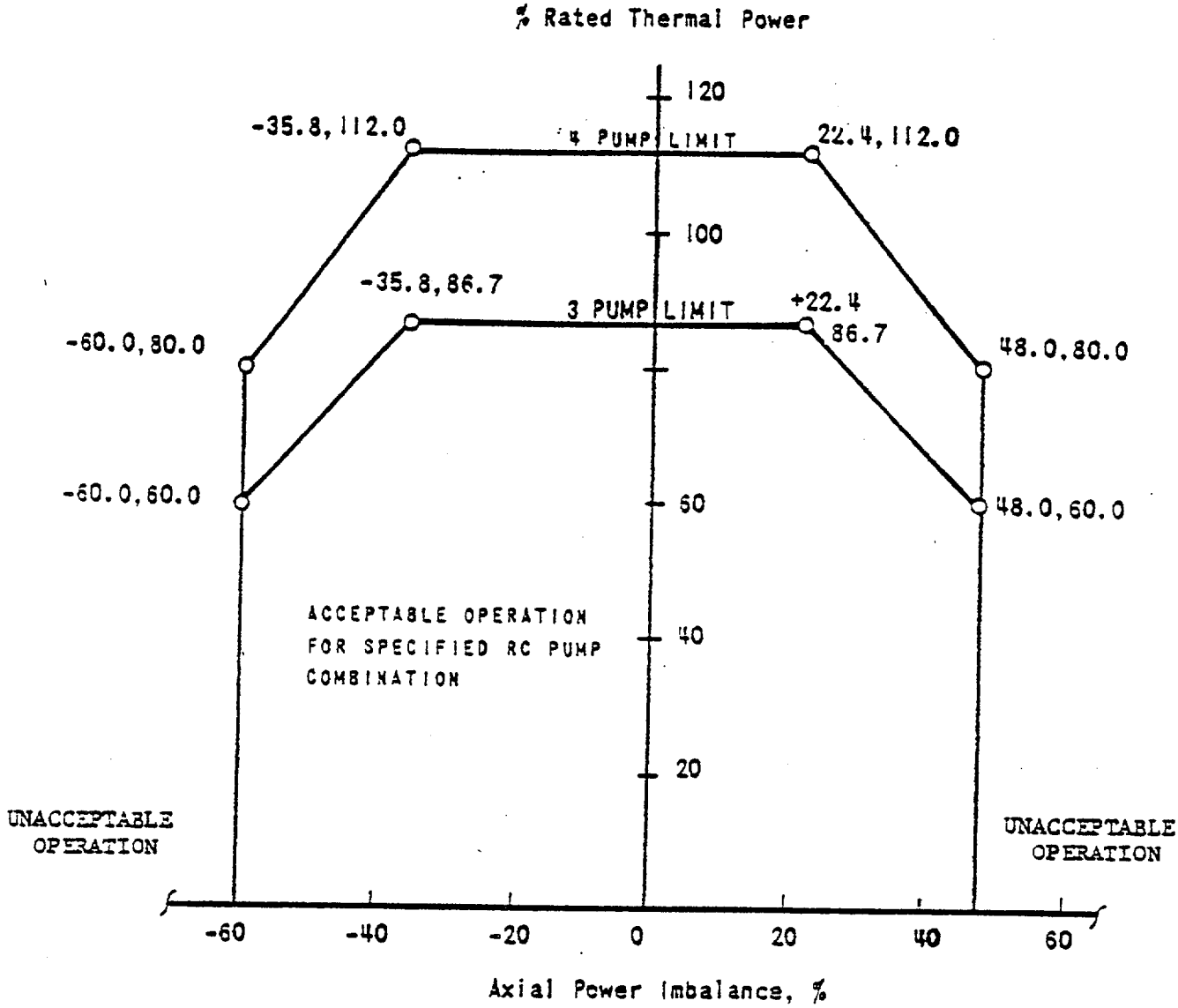


FIGURE 2.1-2 Reactor Core Safety Limit

PUMPS OPERATING	REACTOR COOLANT FLOW (GPM)
4	387,200
3	290,100

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	≤ 105.5% of RATED THERMAL POWER with four pumps operating	≤ 105.6% of RATED THERMAL POWER with four pumps operating#	
	≤ 80.2% of RATED THERMAL POWER with three pumps operating	≤ 80.3% of RATED THERMAL POWER with three pumps operating#	
3. RC High Temperature	≤ 619°F	≤ 619.08°F#	
4. Flux - Δ 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)	≥ 1985 psig	≥ 1984.0 psig*	≥ 1976.5 psig**
6. RC High Pressure	≤ 2300 psig	≤ 2301.0 psig*	≤ 2308.5 psig**
7. RC Pressure-Temperature(1)	≥ (12.60 T_{out} °F - 5660) psig	≥ (12.60 T_{out} °F - 5660.41) 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 pumps 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#

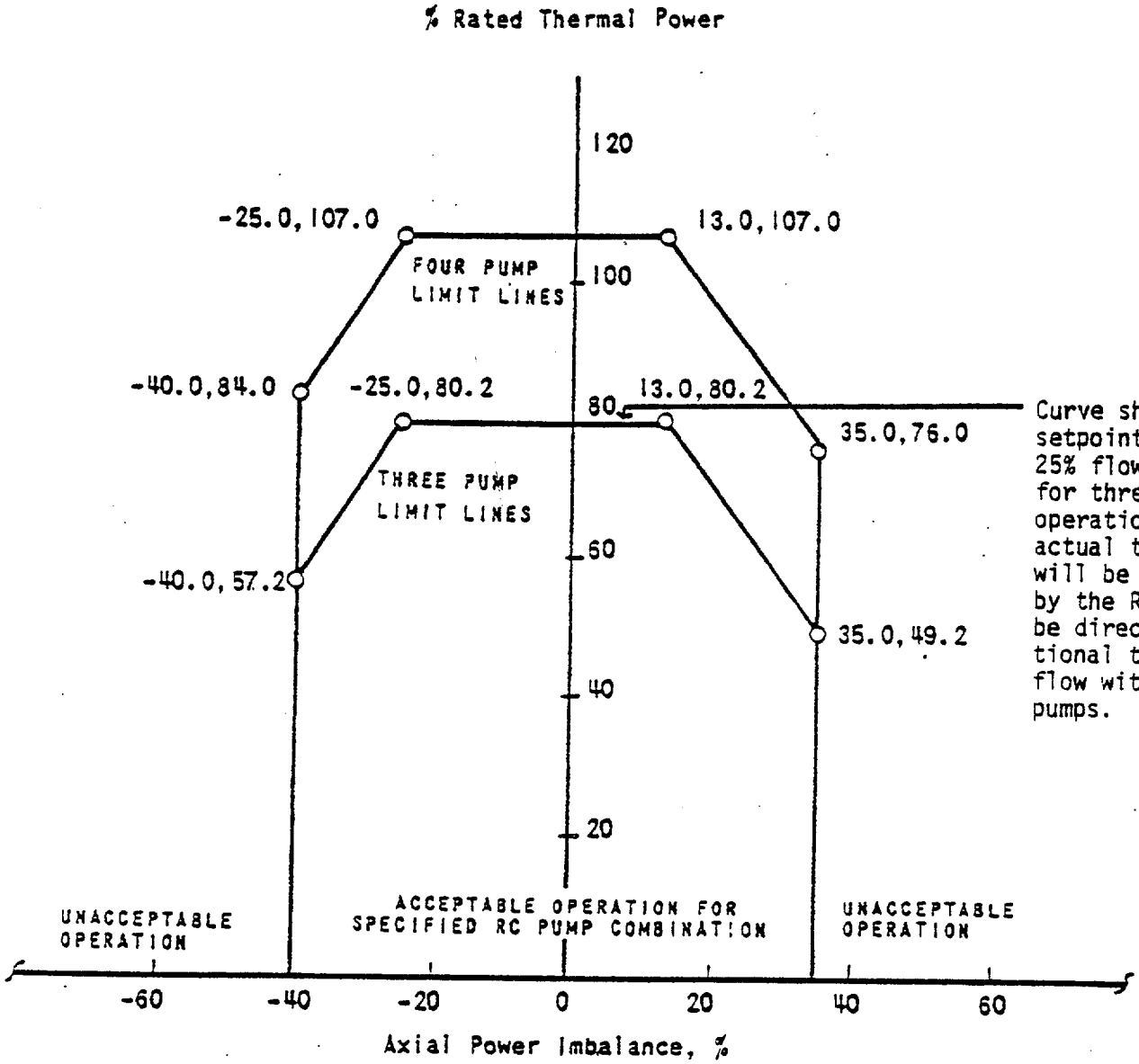
(1) Trip may be manually bypassed when RCS pressure ≤ 1820 psig by actuating Shutdown Bypass provided that:

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



Curve shows trip setpoint for a 25% flow reduction for three pump operation. The actual trip setpoint will be calculated by the RPS and will be directly proportional to the actual flow with three pumps.

FIGURE 2.2-1 Trip Setpoint for Flux-~~Δ~~ Flux-Flow

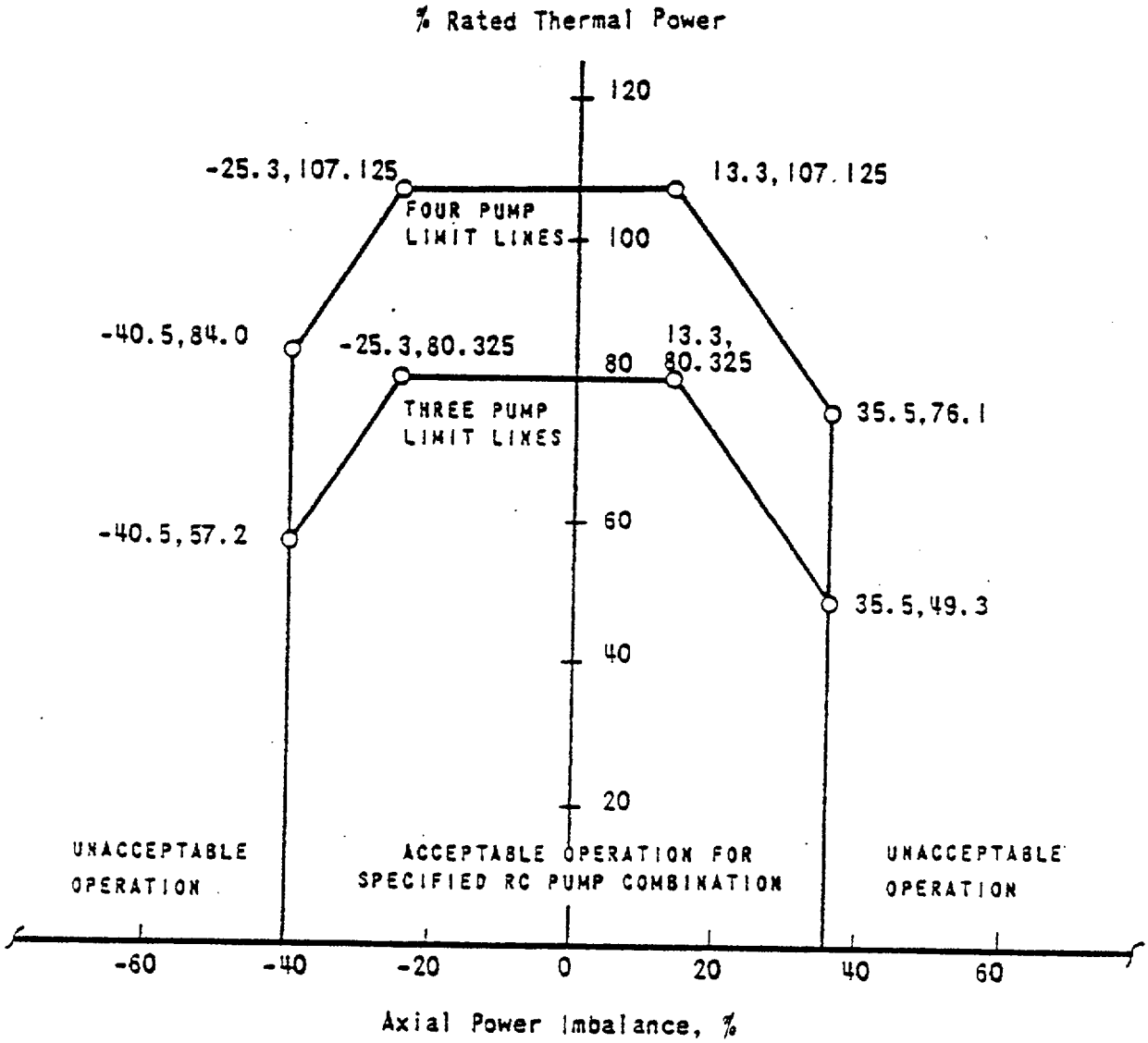


FIGURE 2.2-2 Allowable Value for Flux-Δ Flux-Flow

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 1.30. This value corresponds to a 95 percent probability at a 95 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 1.30 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.

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 1.30 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 1.30 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 and 2 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps and three pumps, 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 1.30 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.

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 1.30 or a local quality at the point of minimum DNBR less than +22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for 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 curve.

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 High Temperature

The RC High Temperature trip ≤ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux - Δ Flux-Flow

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the High Flux/Number of Reactor Coolant Pumps On Trips.

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

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 93.3% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.2% and reactor coolant flow rate is 74.9% of full flow rate, or flow rate is 69.8% of full flow rate and power is 75%.

   For safety calculations the maximum calibration and instrumentation errors for the power level were used. Full flow rate in the above two examples is defined as the flow calculated by the heat balance at 100% power.

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

RC Pressure - Low, High and Pressure Temperature

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

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC High Pressure setpoint is reached before the High Flux Trip Setpoint. The trip setpoint for RC High Pressure, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC High Pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2435 psig. The RC High Pressure trip also backs up the High Flux trip.

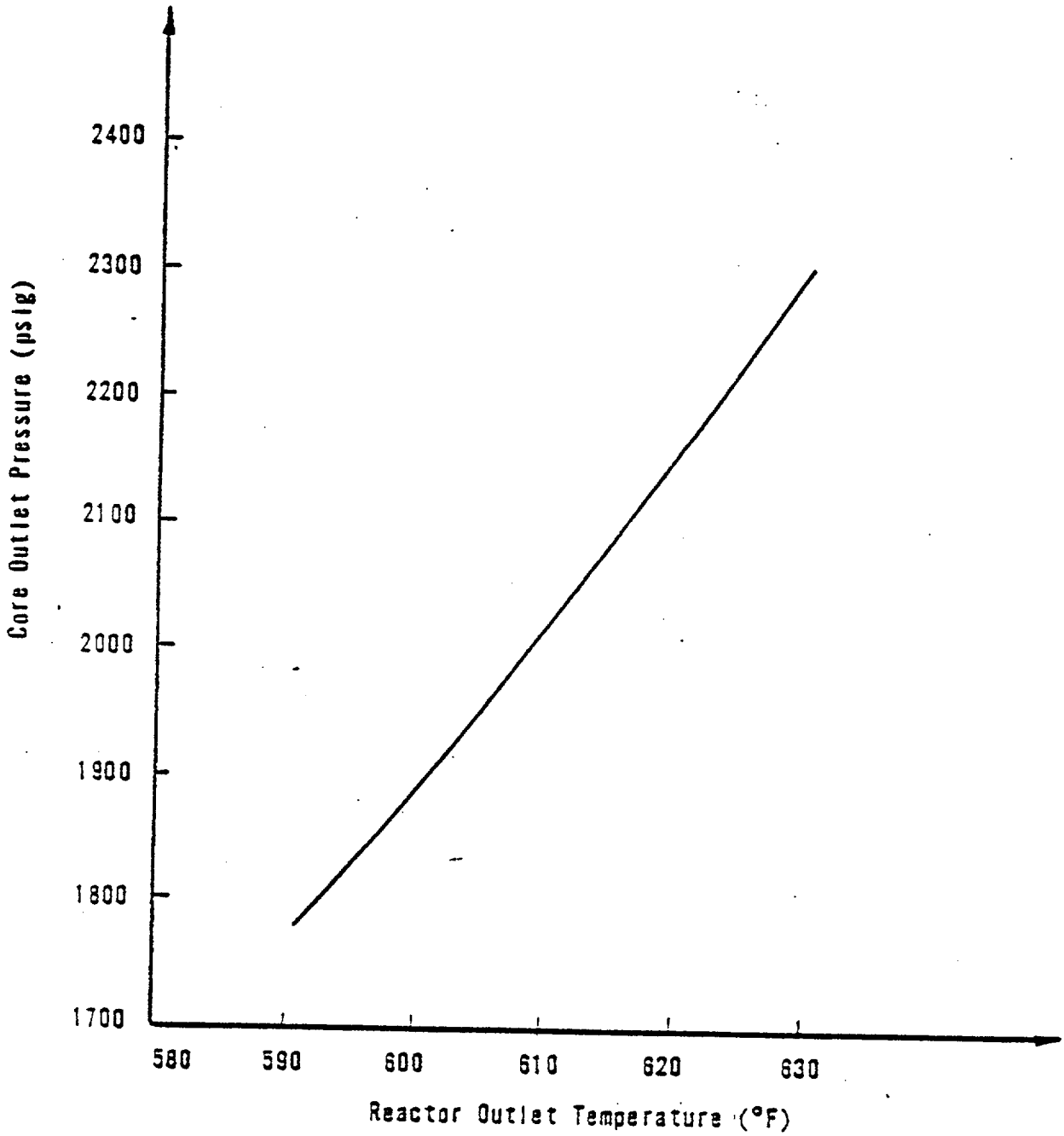
The RC Low Pressure, 1985 psig, and RC Pressure-Temperature (12.60 T_{out} °F-5660) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the Flux - Δ Flux-Flow trip the High Flux/Number of Reactor Coolant Pumps On trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

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.



RC FLOW
387,200 GPM

POWER
112%

Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR
BASES FIGURE 2.1

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 and -3b, with a rod group overlap of 25 ± 5% between sequential withdrawn groups 5, 6, and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits (in a region other than acceptable operation), 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.

NOTE: If in unacceptable region, also see Section 3/4.

*See Special Test Exceptions 3.10.1 and 3.10.2.

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

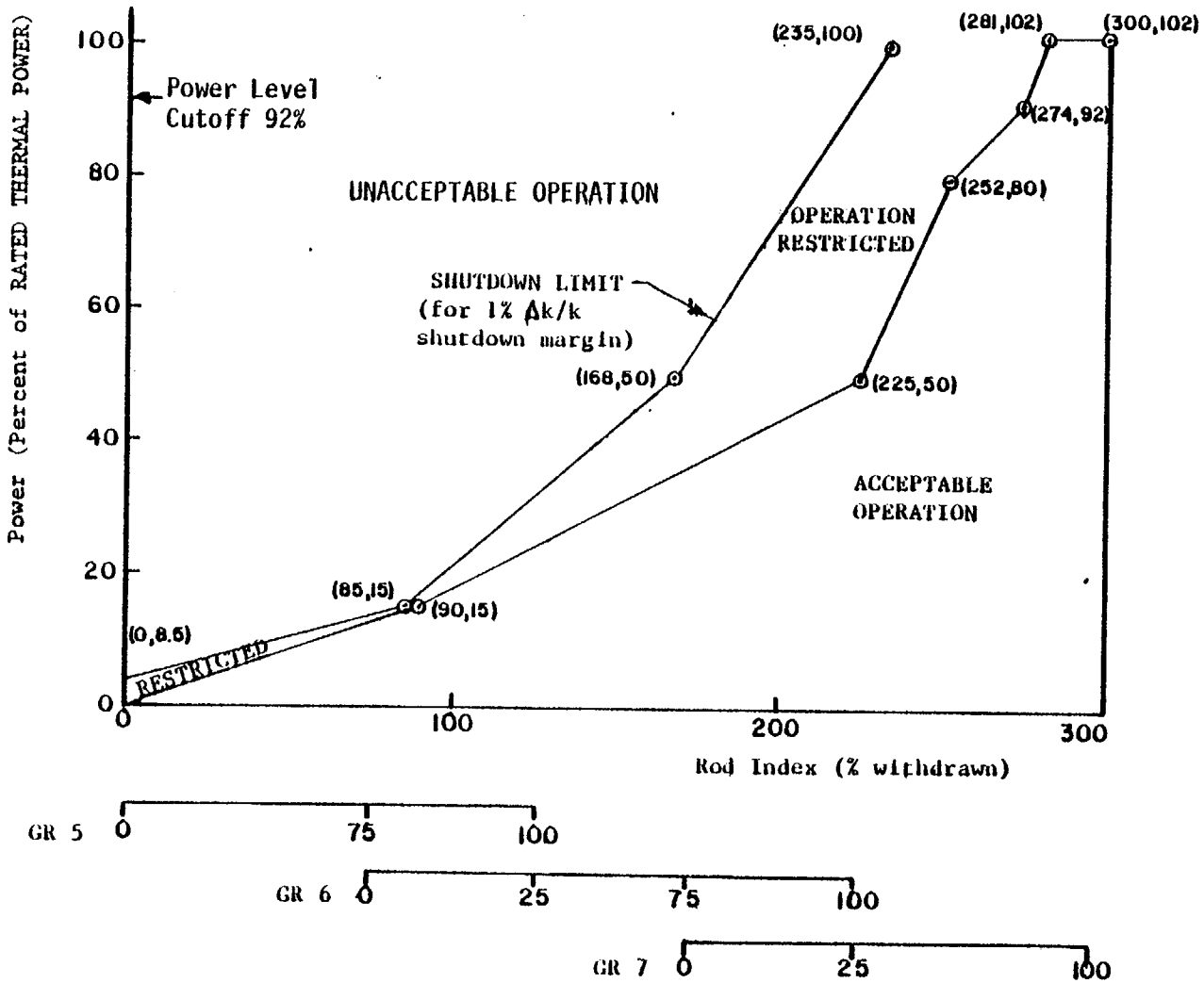
REGULATING ROD INSERTION LIMITS

SURVEILLANCE REQUIREMENTS

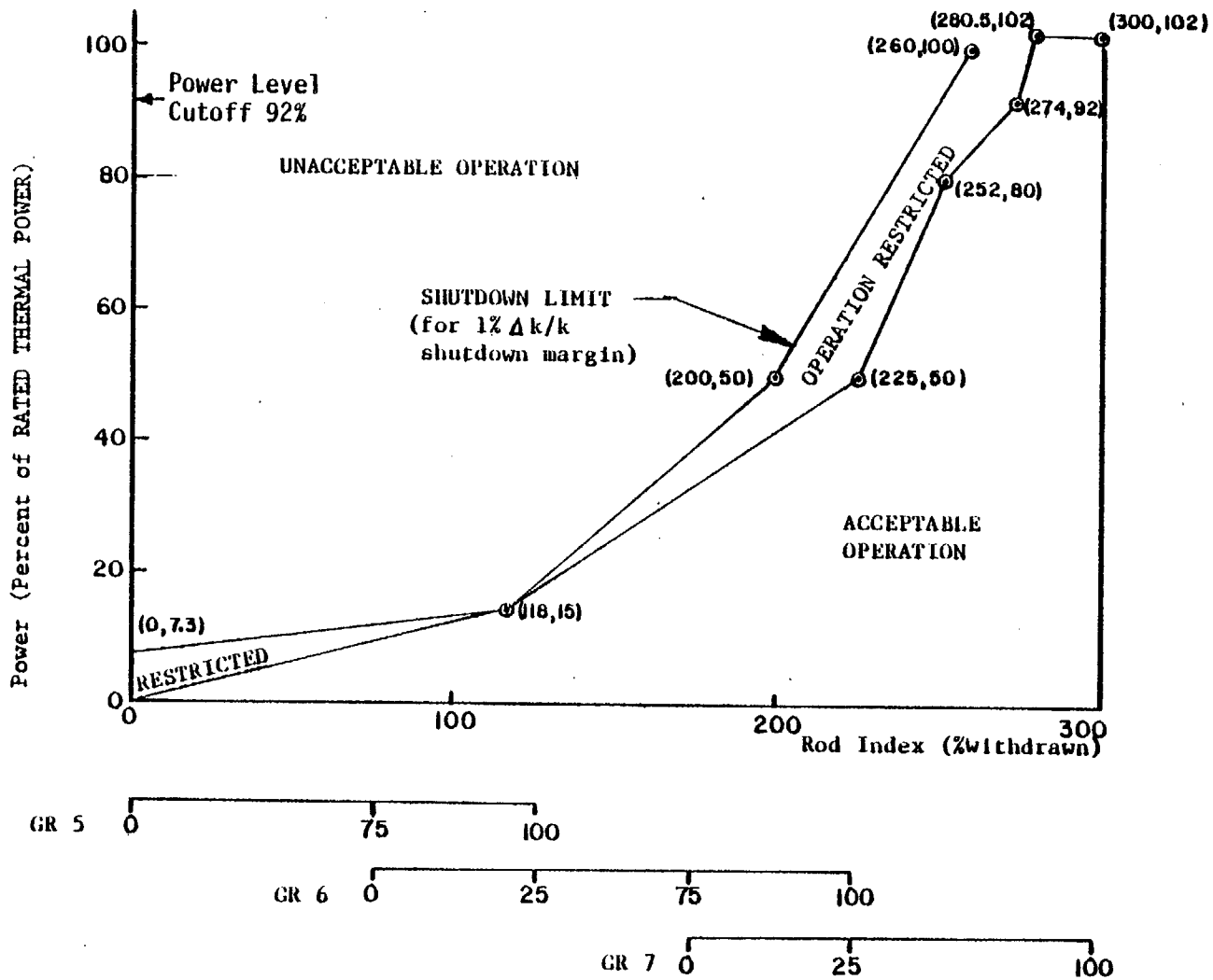
4.1.3.6 The position of each regulating group shall be determined to be within the insertion, sequence and overlap limits at least once every 12 hours except when:

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;

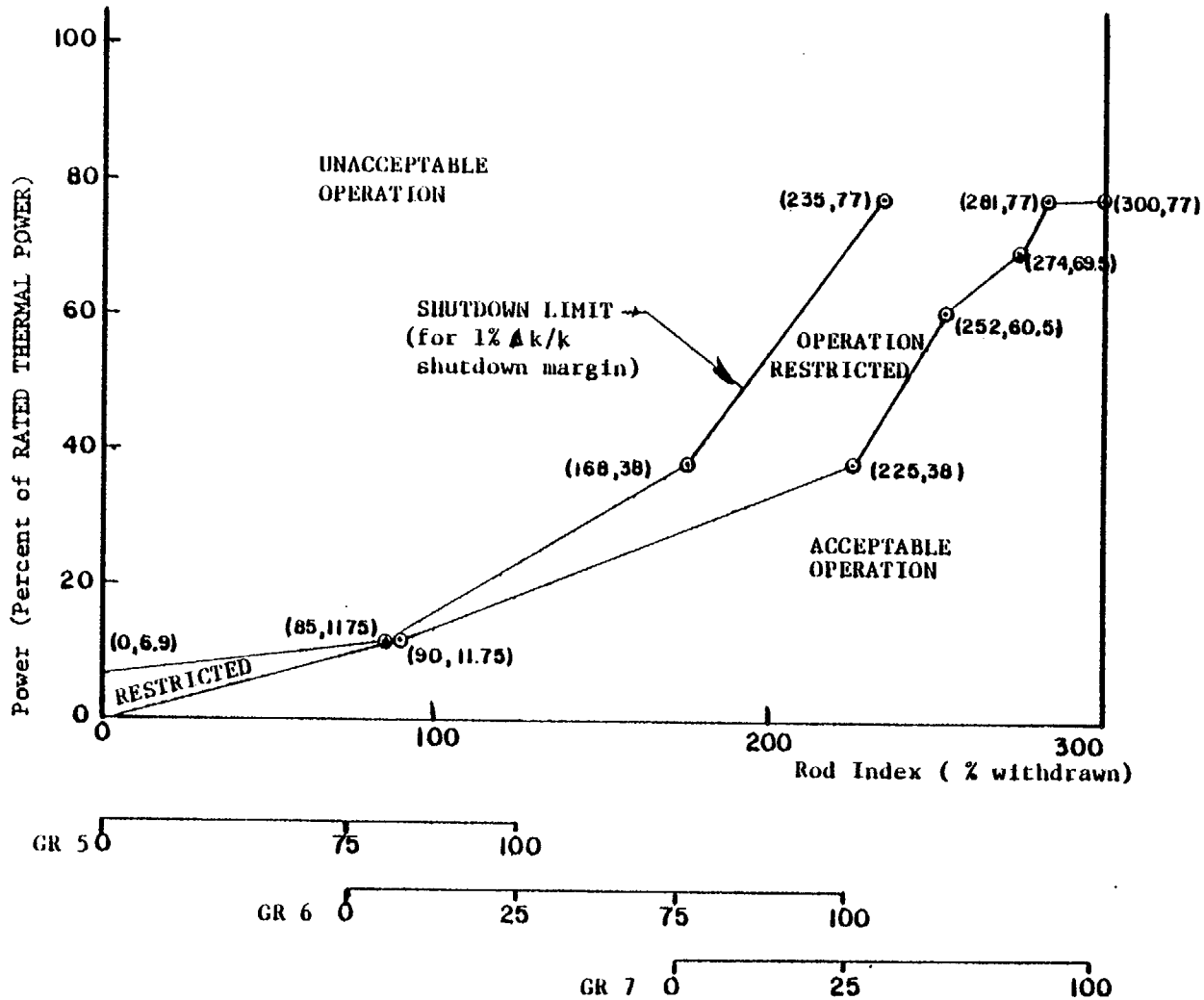
b. The control rod drive sequence alarm is inoperable, then verify the groups to be within the sequence and overlap limits at least once per 4 hours.



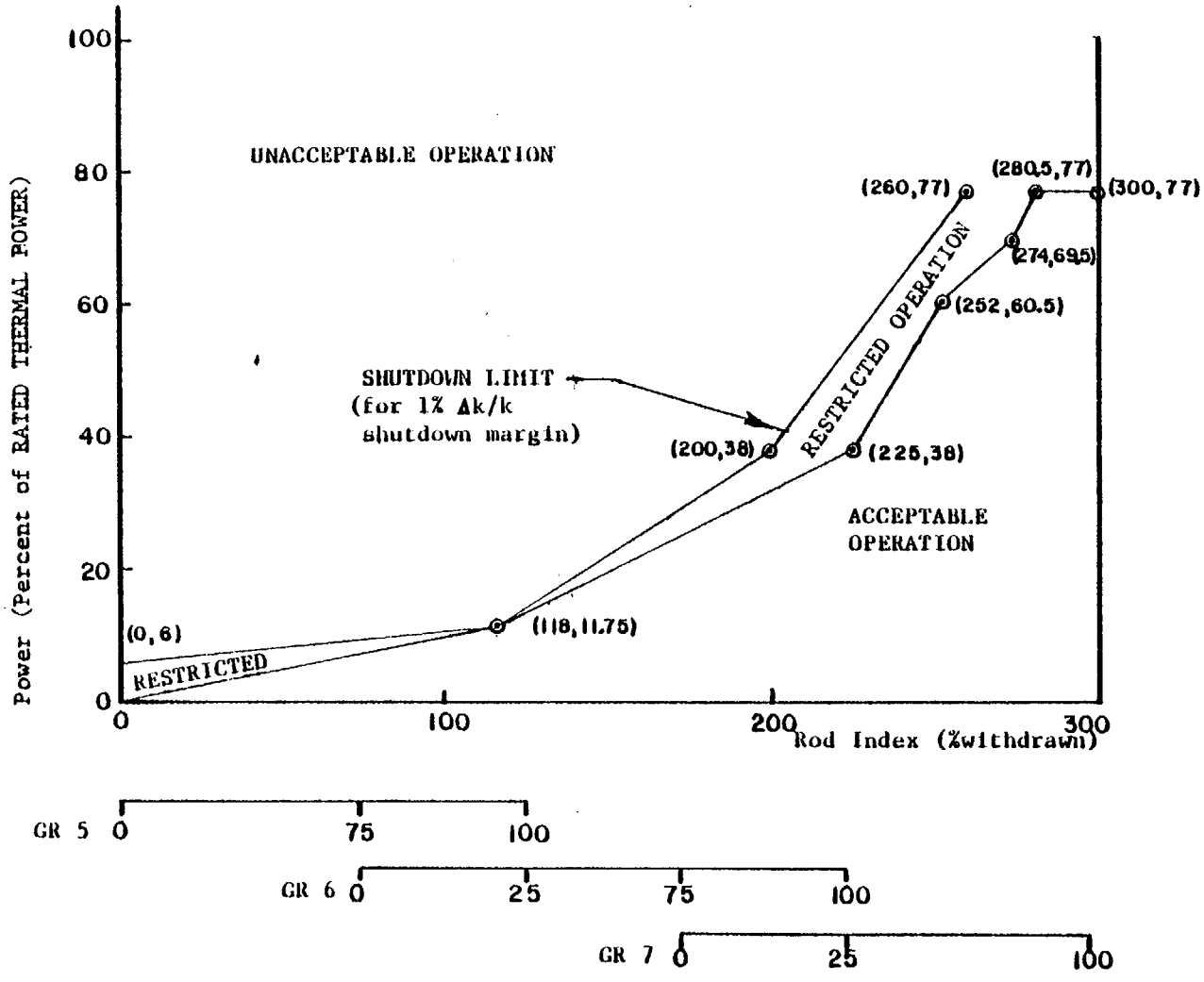
TECHNICAL SPECIFICATION FIGURE 3.1-2a Regulating Group Position Limits, 0-150±10 EFPD, Four RCPs - Davis-Besse 1, Cycle 2



TECHNICAL SPECIFICATION FIGURE 3.1-2b Regulating Group Position Limits, After 150 ±10 EFPD, Four RCPs - Davis Besse 1, Cycle 2



TECHNICAL SPECIFICATION FIGURE 3.1-3a Regulating Group Position Limits, 0 to 125 ±10 EFPD, Three RCPs - Davis Besse 1, Cycle 2



TECHNICAL SPECIFICATION FIGURE 3.1-3b Regulating Group Position Limits, After 150 ±10 EFPD, Three RCPs - Davis Besse 1, Cycle 2

Figure 3.1-3c Deleted

Figure 3.1-3d Deleted

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:

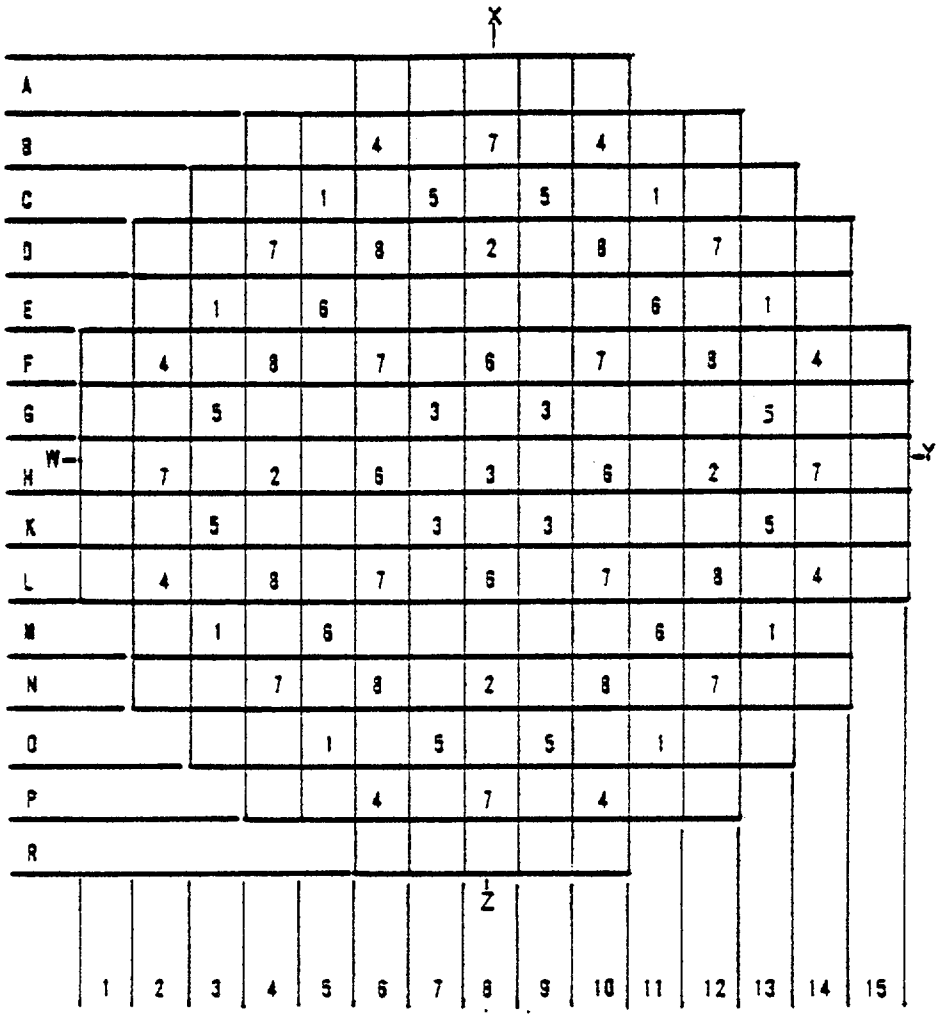
   a) For all control rods, after the control rod drive patches are locked subsequent to test, reprogramming or maintenance within the panels.

   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.



Group	No. of Rods	Purpose
1	8	Safety
2	4	Safety
3	5	Safety
4	8	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total	61	

FIGURE 3.1-4 Control Rod Core Locations and Group Assignments — Davis-Besse 1, Cycle 2

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XENON REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.3.8 THERMAL POWER shall not be increased above the power level cutoff specified in Figure 3.1-2 unless one of the following conditions is satisfied:

a. Xenon reactivity is within 10 percent of the equilibrium value for RATED THERMAL POWER and is approaching stability, or

b. THERMAL POWER has been within a range of 87 to 92 percent of RATED THERMAL POWER for a period exceeding 2 hours in the soluble poison control mode, excluding xenon free start-ups.

APPLICABILITY: MODE 1.

ACTION:

With the requirements of the above specification not satisfied, reduce THERMAL POWER to less than or equal to the power level cutoff within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.3.8 Xenon reactivity shall be determined to be within 10% of the equilibrium value for RATED THERMAL POWER and to be approaching stability or it shall be determined that the THERMAL POWER has been in the range of 87 to 92% of RATED THERMAL POWER for $\geq$ 2 hours, prior to increasing THERMAL POWER above the power level cutoff.

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.9 The axial power shaping rod group shall be limited in physical insertion as shown on Figures 3.1-5a, -5b, -5c, and -5d.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the axial power shaping rod group outside the above insertion limits, either:

a. Restore the axial power shaping rod group 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.

SURVEILLANCE REQUIREMENTS

4.1.3.9 The position of the axial power shaping rod group shall be determined to be within the insertion limits at least once every 4 hours.

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

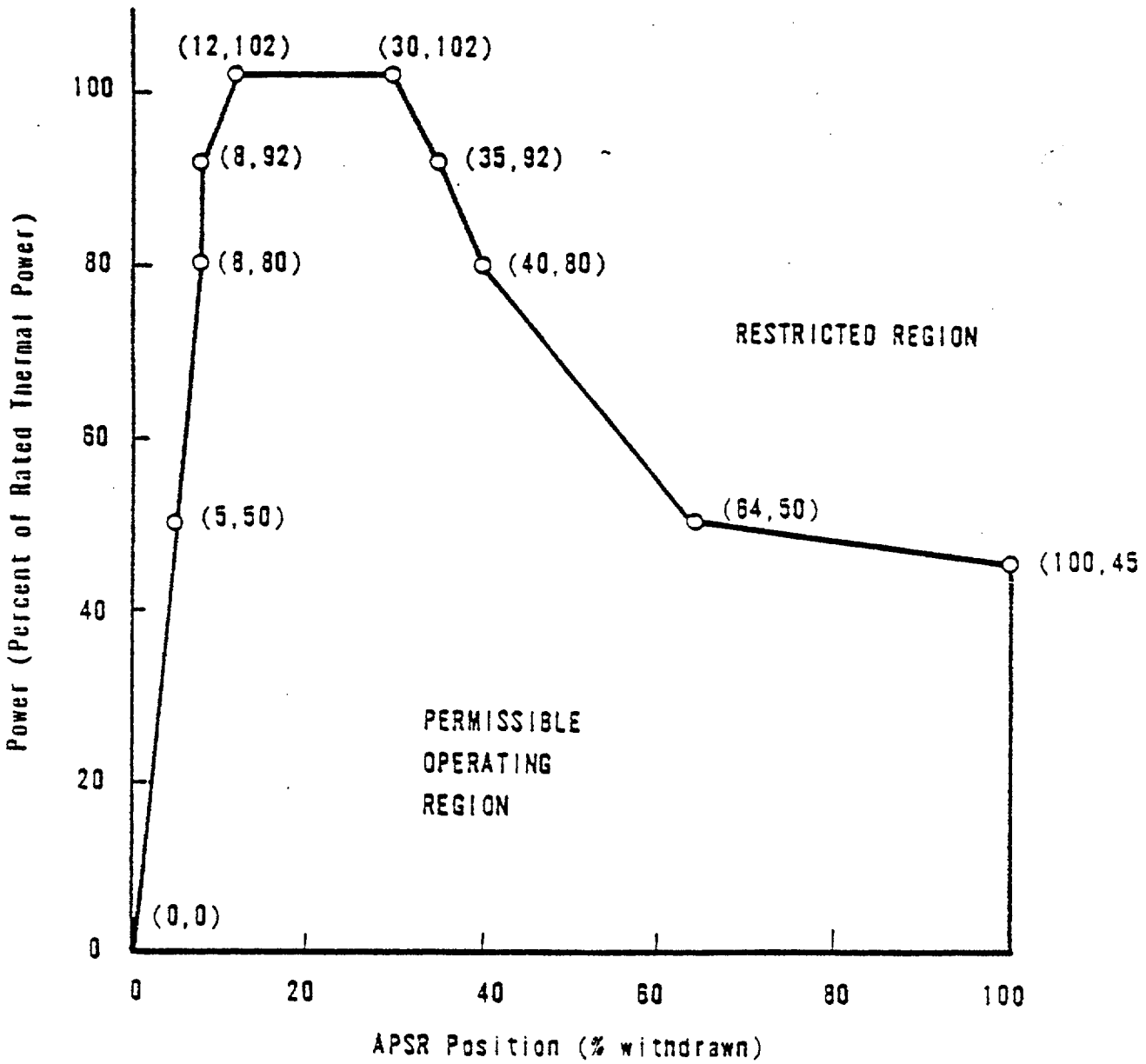

FIGURE 3.1-5a

APSR Position Limits, 0 to 150 ±10 EFPD,
Four RCPs - Davis-Besse 1, Cycle 2

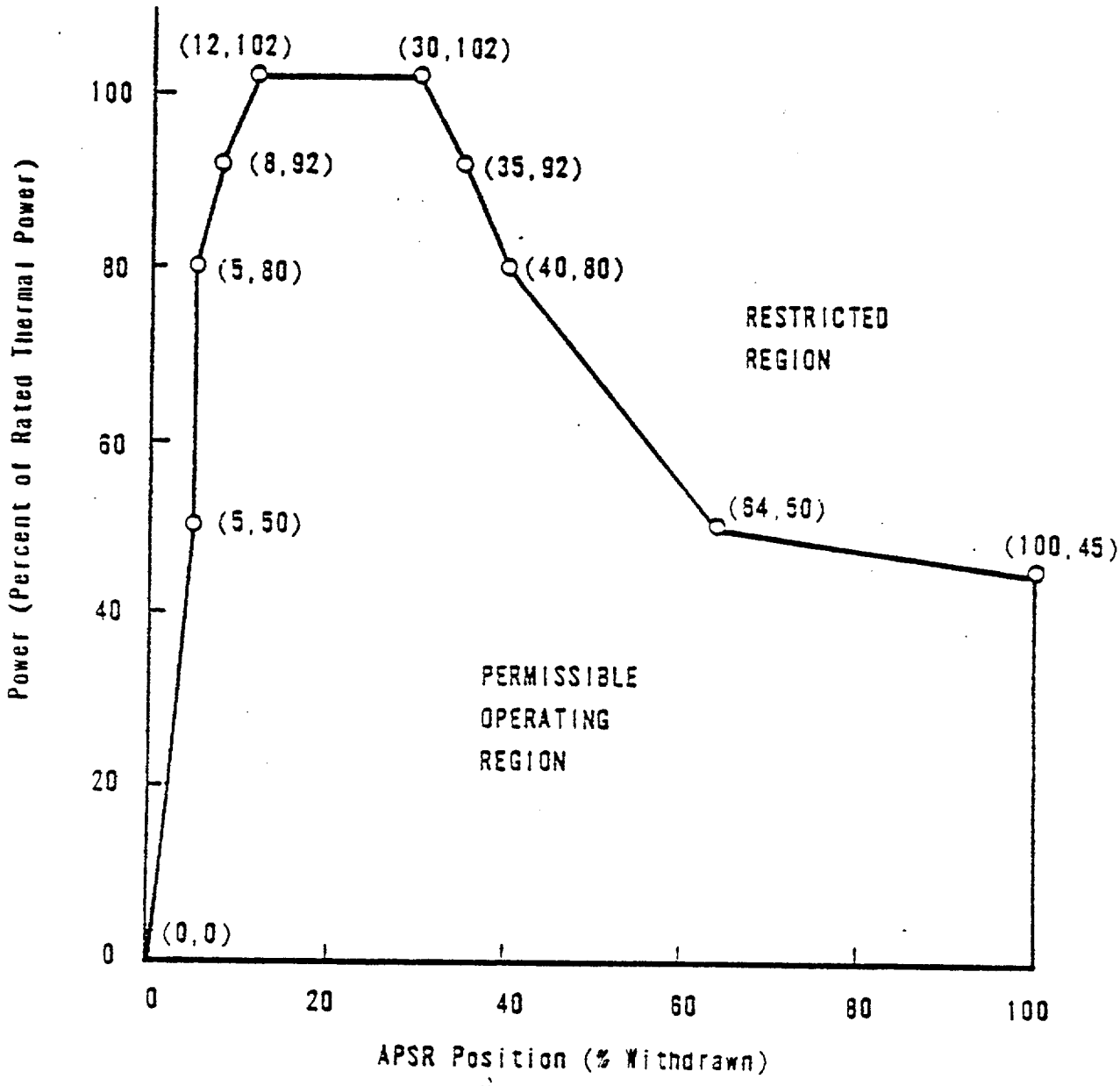

FIGURE 3.1-5 b

APSR Position Limits, After 150 ±10 EFPD,
Four RCPs - Davis-Besse 1, Cycle 2

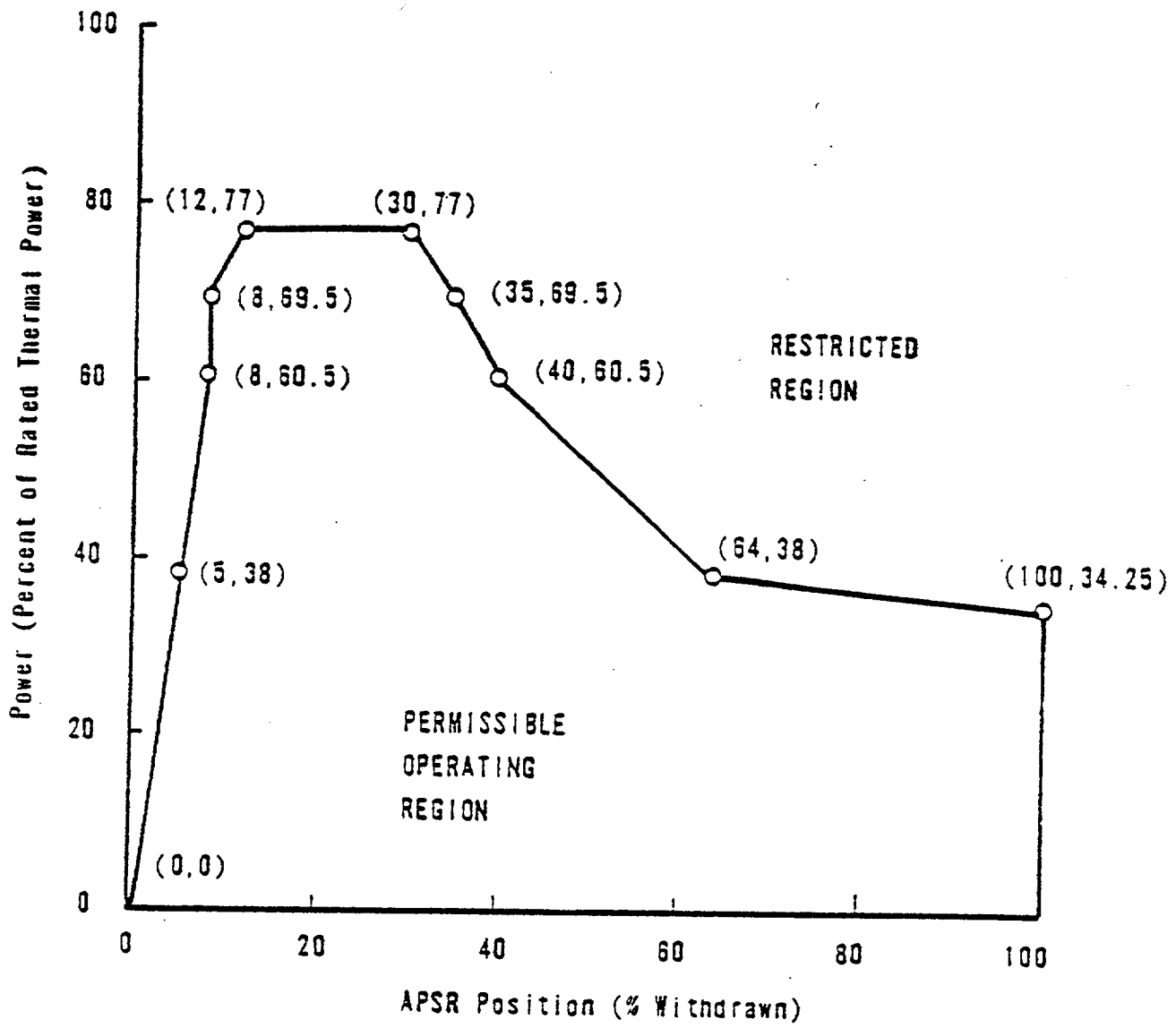

FIGURE 3.1-5 c

APSR Position Limits, 0 to 150 ±10 EFPD,
Three RCPs - Davis-Besse 1, Cycle 2

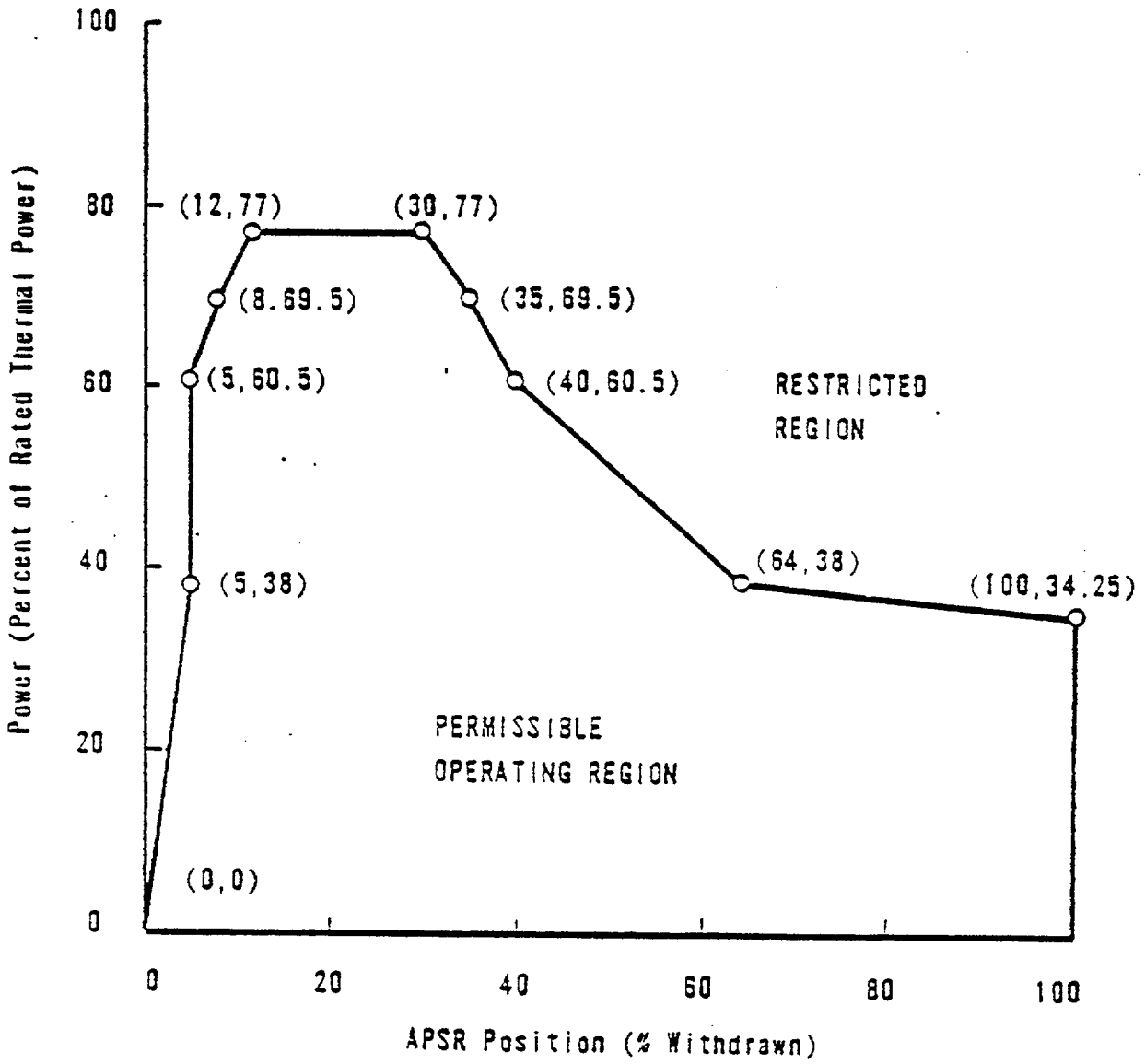

FIGURE 3.1-5 d

APSR Position Limits, After 150 ±10 EFPD,
Three RCPs - Davis-Besse 1, Cycle 2

3/4.2 POWER DISTRIBUTION LIMITS

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 and 3.2-2.

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

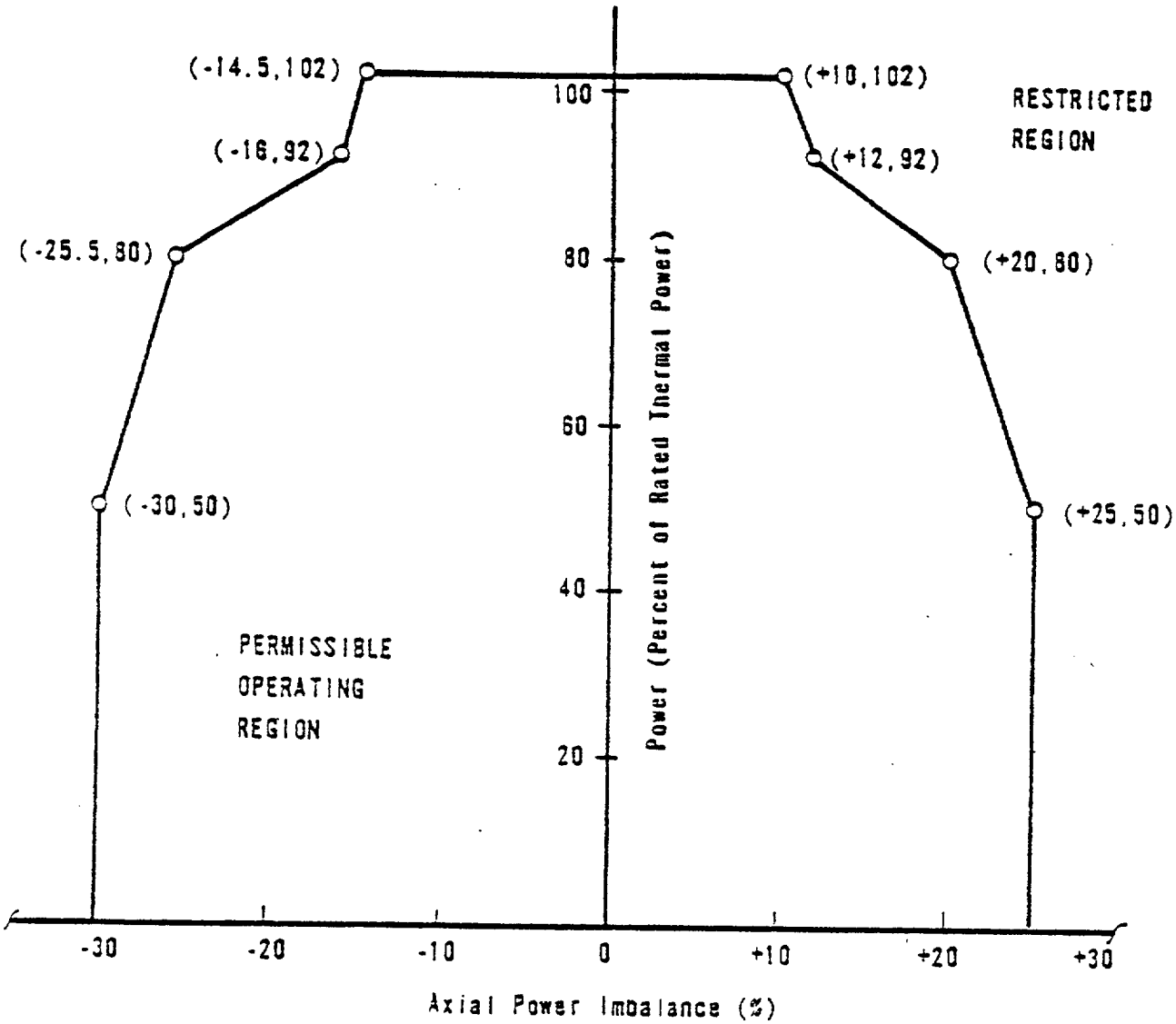

FIGURE 3.2-1a

Axial Power Imbalance Limits, 0 to 150 ±10 EFPD,
Four RCPs - Davis-Besse 1, Cycle 2

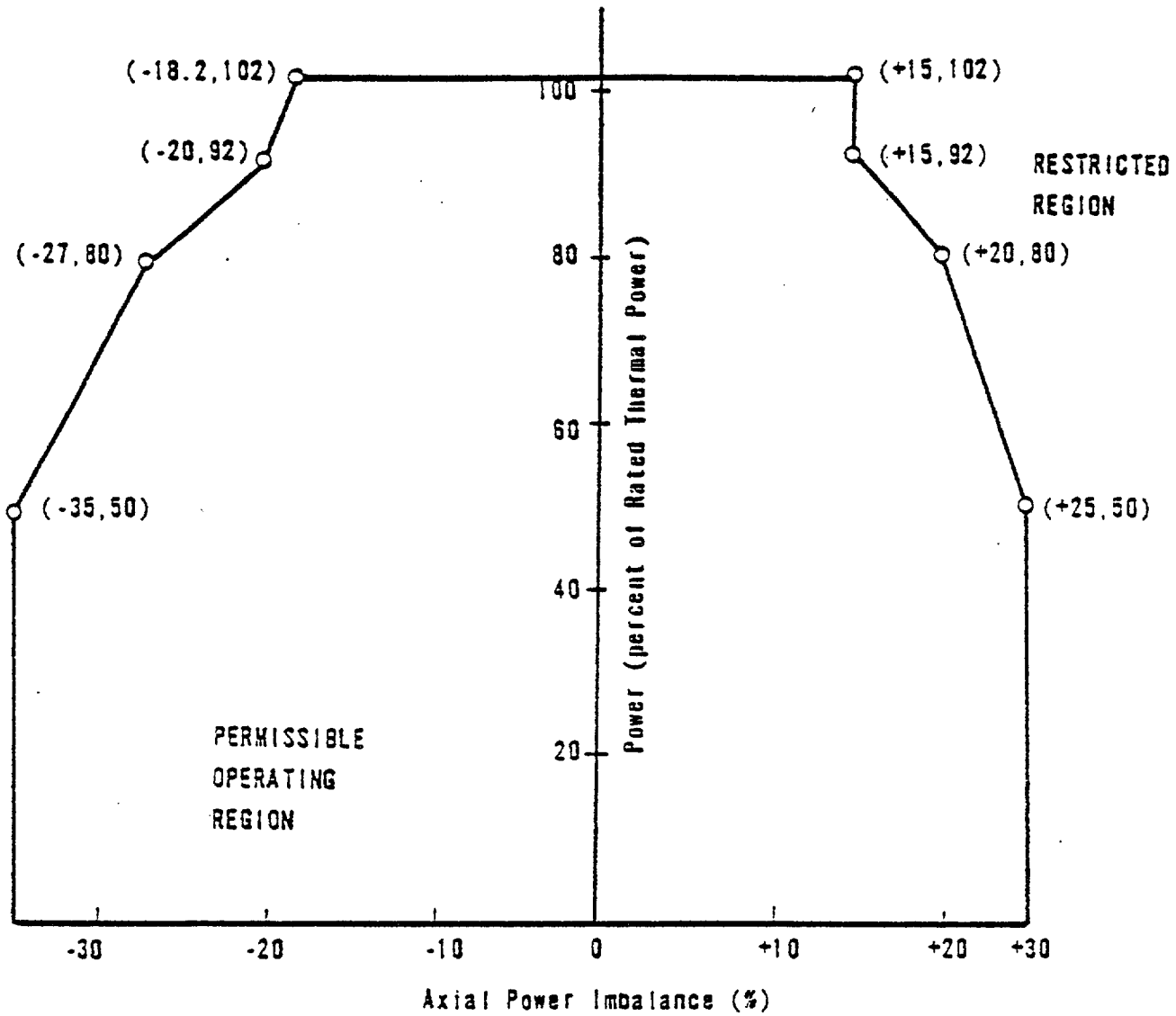

FIGURE 3.2-1b

Axial Power Imbalance Limits, After 150 ±10 EFPD,
Four RCPs - Davis-Besse 1, Cycle 2

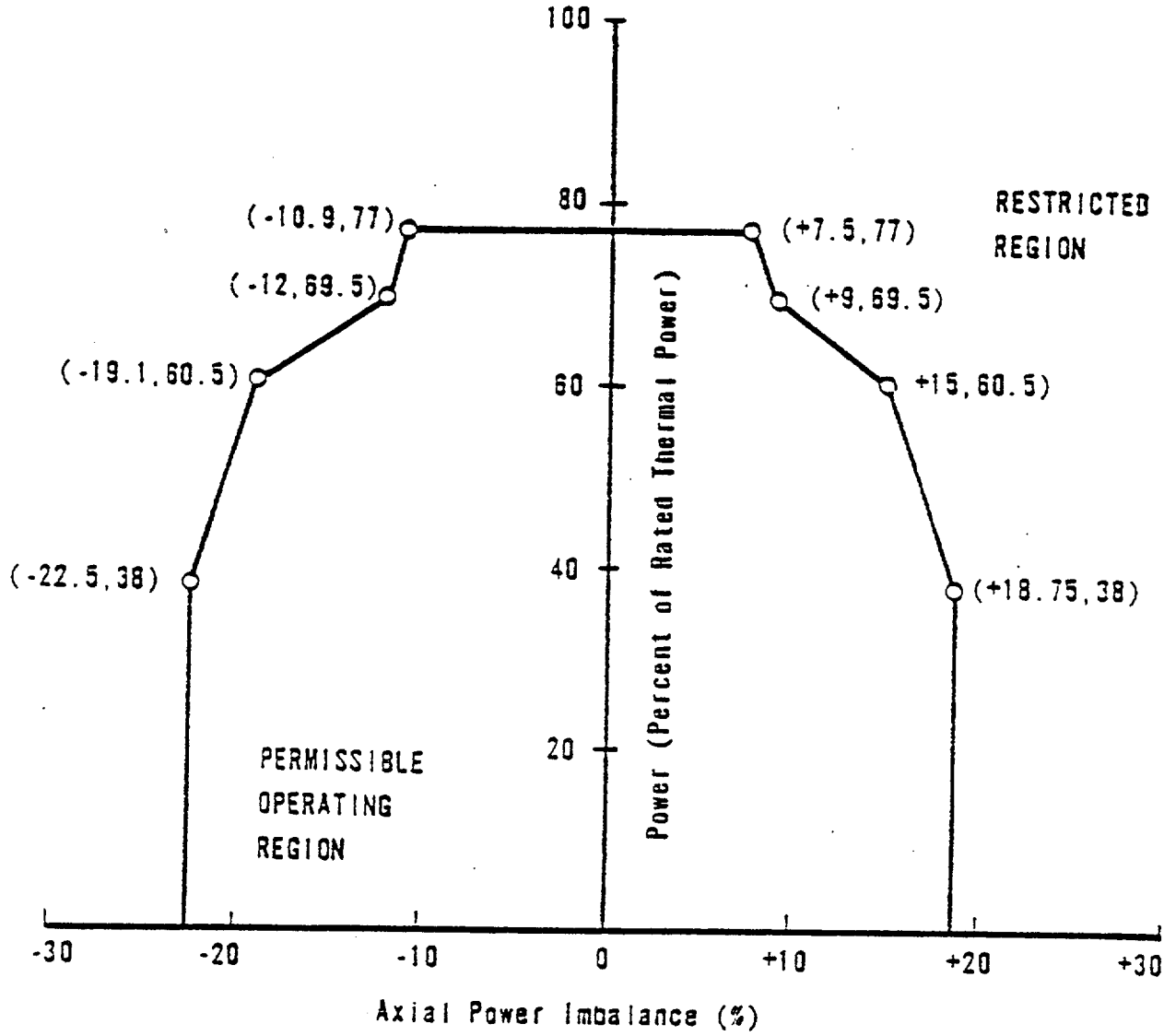

FIGURE 3.2-2a

Axial Power Imbalance Limits, 0 to 150 ±10 EFPD,
Three RCPs - Davis-Besse 1, Cycle 2

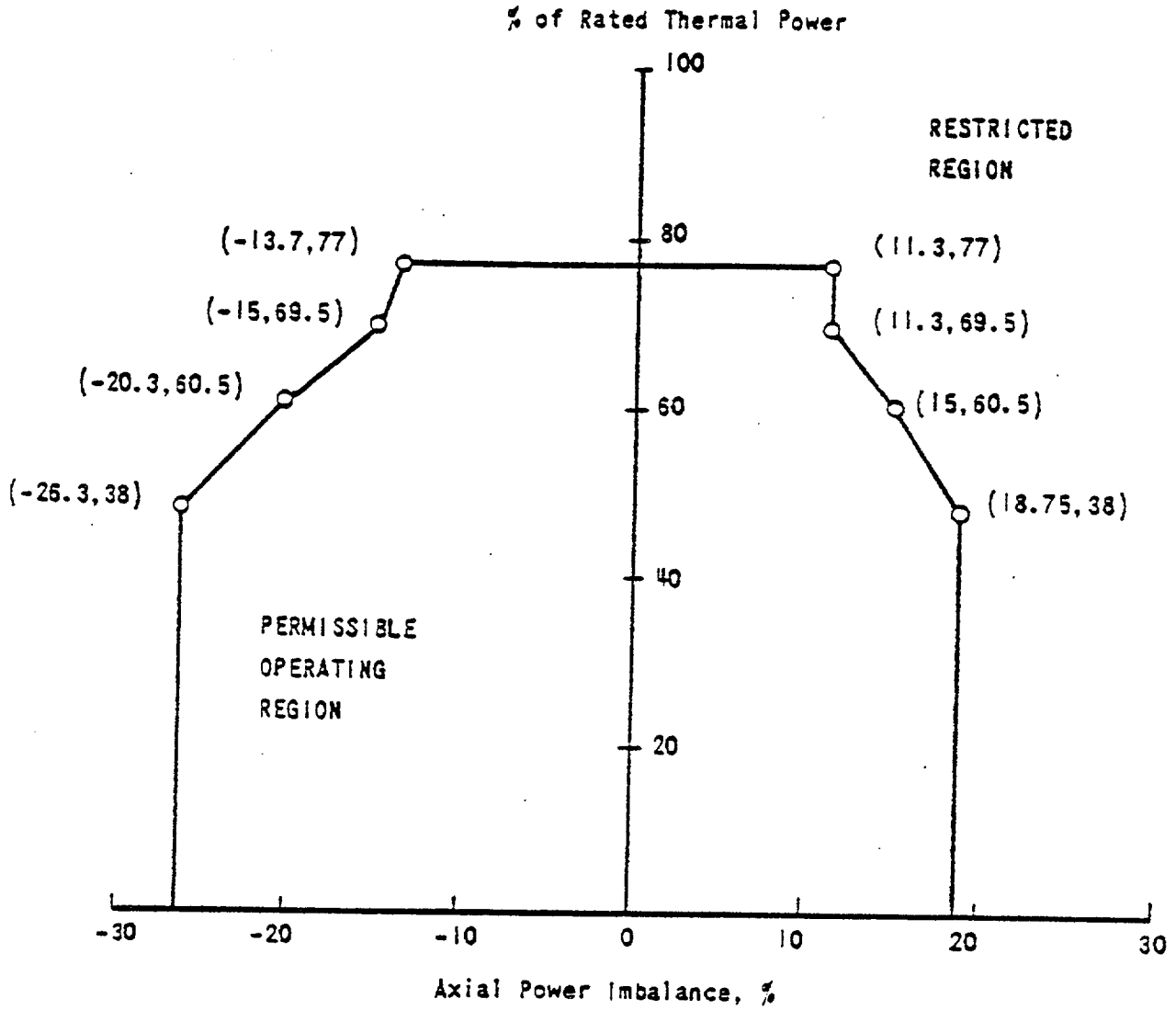

FIGURE 3.2-2b

Axial Power Imbalance Limits, After 150 ±10 EFPD,
Three RCPs - Davis-Besse 1, Cycle 2

Figure 3.2-3a Deleted

Figure 3.2-3b Deleted

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 ≤ 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	3.21	8.71	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
Reactor Coolant Hot Leg Temperature T_H°F	≤ 610	≤ 610[1]
Reactor Coolant Pressure, psig.[2]	≥ 2062.7	≥ 2058.7[1]
Reactor Coolant Flow Rate, gpm[3]	≥ 396,880	≥ 297,340

(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%.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 80.2% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. High Flux
   2. Flux-ΔFlux-Flow

*See Special Test Exception 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

MODES 3, 4 and 5:

a. Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or decay heat removal pump.*

b. Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operators, or manual valves DH-21 and DH-23 are opened.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*All reactor coolant pumps and decay heat removal train pumps may be de-energized for up to 1 hour to accommodate surveillance testing & pre-operational testing, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2435 PSIG ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE DHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SAFETY VALVES AND ELECTROMATIC RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2435 PSIG ± 1%.* When not isolated, the pressurizer electromatic relief valve shall have a trip setpoint of ≥ 2390 PSIG and an allowable value of ≥ 2385.5 PSIG.**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer electromatic relief valve a channel calibration check shall be performed every 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Allowable value for channel calibration check.

3/4.10 SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.1 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.1.3.9, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER,

b. The High Flux Trip Setpoint is ≤ 10% of RATED THERMAL POWER higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% of RATED THERMAL POWER, and

c. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in 4.10.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.1.3.9, 3.2.1 or 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

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

SURVEILLANCE REQUIREMENTS

4.10.1.1 The High Flux Trip Setpoint shall be determined to be set within the limits specified within 8 hours prior to the initiation of and at least once per 8 hours during PHYSICS TESTS.

4.10.1.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at least once per two hours during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.2 The limitations of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, and 3.1.3.9 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE High Flux Channels are set at ≤ 25% of RATED THERMAL POWER.

c. The nuclear instrumentation Source Range and Intermediate Range high startup rate control rod withdrawal inhibit are OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the control rod drive trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 Each Source and Intermediate Range and High Flux Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

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 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

3/4.1.2 BORATION SYSTEMS (Continued)

stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 8603 gallons of 8742 ppm borated water from the boric acid storage system or 28,200 gallons of 1800 ppm borated water from the borated water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution recirculated within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analyses. Measurement with $T_{avg} \geq 525°F$ and with reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The limitation on THERMAL POWER based on xenon reactivity is necessary to ensure that power peaking limits are not exceeded even with specified rod insertion limits satisfied.

The limitation on axial power shaping rod insertion is necessary to ensure that power peaking limits are not exceeded.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 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 and 3.2-2 and the insertion limit curves, Figures 3.1-2 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 3.1-2 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 ± 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.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding 20.4 kw/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, AXIAL POWER IMBALANCE, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the Reactor Protection System.

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

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

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

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one loop, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE and the Nuclear Overpower Based on Pump Monitors trip, ensuring that the DNBR will be maintained above 1.30 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 equal to 22%, whichever is more restrictive.

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a DHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 336,000 lbs per hour of saturated steam at the valve's setpoint.

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating DHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

For a RPS high pressure trip setpoint of 2300 psig, the maximum overshoot of the Reactor Coolant System pressure for a loss of feedwater (LOFW) event would be 2350 psig. Also, the LOFW is the maximum over-pressure anticipated transient. The string inaccuracies and drift for the RPS high pressure trip are 15.29 psi, or 16 psi conservatively. The maximum pressure peak for an anticipated transient is then 2366 psig.

The inaccuracies and drift for the string that controls the electromatic relief valve for the pressurizer are 16.75 psi, or 17 psi conservatively. Included in this value is an inaccuracy of 4 psi and a drift of 7.5 psi for the transmitter. The 4 psi and 7.5 psi were combined by taking the square root of the sum of the squares, giving 8.5 psi. Subtracting 4 psi from 8.5 psi gives a value of 4.5 psi that is attributable to only the drift. The 8.5 psi was then added to inaccuracy and drift values for other components in the string to obtain a total of 16.75 psi.

The allowable value of $\geq$2385.5 psig is obtained by subtracting 4.5 psi due to the drift from the trip setpoint of $\geq$2390 psig. The minimum lift pressure for the pressurizer electromatic relief valve is then (2400 - 10 - 17) psig = 2373 psig. Consequently, the resultant margin between the maximum pressure peak of 2366 psig and minimum lift pressure of 2373 psig for the pressurizer electromatic relief valve following an anticipated transient is 7 psi.

Thus, a 2300 psig RPS high pressure trip setpoint and the above values for the pressurizer electromatic relief valve will avoid actuation of the pressurizer electromatic relief valve during anticipated transients.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Safety Feature Actuation System. The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NO. NPF-3

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1. Introduction

By letters dated February 11, 1980 (Reference 2-1), May 21, 1980 (Reference 2-2) and August 22, 1980, the Toledo Edison Company (TECo or the licensee) made application to modify the Technical Specifications for the Davis-Besse Nuclear Power Station, Unit No. 1, to permit operation for a second cycle. Cycle 1 was terminated after 360 effective full power days (EFPD) and Cycle 2 has a design length of 248 EFPD. Our evaluation of this application follows.

2. Evaluation of Fuel System Design

2.1 Fuel Assembly Mechanical Design

The fresh Babcock and Wilcox (B&W) Mark B-4 15x15 fuel assemblies loaded as Batch 4 at the end of Cycle 1 (EOC 1) are mechanically interchangeable with Batches 1, 2 and 3 fuel assemblies previously loaded at Davis-Besse Unit 1. Forty-four Batch 1 assemblies have been discharged and an identical number of Batch 4 assemblies will be loaded for Cycle 2. This reload scheme is a revision (2-2) to that originally proposed (2-1) by the licensee. The change allows the reinsertion of 12, rather than 4, Batch 1 fuel assemblies into the Cycle 2 core. This revision was necessitated by the lower than anticipated exposure accumulated on the Cycle 1 core and was not due to fuel design considerations. The designation 1B is

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now used to identify the Batch 1 assemblies being reused for Cycle 2. Batch 1A is now the remainder of the Batch 1 assemblies which have not been scheduled for reinsertion.

The Mark B-4 fuel assembly was previously described (2-3) in the Final Safety Analysis Report (FSAR) for Davis-Besse Unit 1. The design has been approved (2-4) by the NRC staff and is used in other B&W nuclear steam supply systems. Two assemblies will contain primary neutron sources and two assemblies will contain regenerative neutron sources in Cycle 2. Retainers will be used on the four fuel assemblies that contain the neutron sources. Justification for the design and use of the neutron source retainers is described in the "Burnable Poison Rod Assembly Retainer Design Report" (2-5). A discussion of the burnable poison rods themselves is presented in Section 2.1.1.

2.1.1 Reactivity Control System

Davis-Besse Unit 1 will be operated in a feed-and-bleed mode during Cycle 2. That is, the core reactivity control will be supplied mainly by soluble boron in the reactor coolant. This mode of reactivity control results in reduced power peaking and, therefore, allows the core to be operated at an increased power density compared to that permitted in "rodded" reactors such as Crystal River Unit 3. Reactivity control at Davis-Besse Unit 1 is further supplemented by 53 full-length control rod assemblies (CRAs) composed of silver-indium-cadmium alloy clad in stainless steel. In addition to the full-length control rods, eight axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The axial power shaping rods are similar in design to the full-length rods. The locations of all 61 control rods and the group designations are indicated in Figure 3-3 of Ref. 2-2. Although

the rod group designations differ, the core locations of the control rods for Cycle 2 are identical to those of Cycle 1. The mechanical aspects of the control rods are also identical to that described in the Davis-Besse Unit 1 FSAR. The control rod design was found to be acceptable (2-4) for Cycle 1 and no further review is required for Cycle 2 operation.

In addition to the permanent reactivity control systems (soluble boron and control rods), 68 burnable poison rod assemblies (BPRAs) were added to the first cycle to control reactivity changes due to fuel burnup and fission product buildup. The BPRAs are normally removed from the reactor at the end of first cycle. In April 1978, two BPRAs were accidently ejected from the core of another B&W-designed reactor at Crystal River (2-6). The ejected BPRAs were carried out of the reactor vessel by the coolant flow to the steam generator, where significant damage to the steam generator tube ends resulted. B&W determined that the ejection of the BPRAs from the core resulted from fretting wear in the holddown latching mechanism. In order to avoid similar problems at Davis-Besse Unit 1, the licensee met with the NRC staff (2-7), submitted a proposed license amendment (2-8), and removed all BPRAs from the core before the end of Cycle 1. This change was approved by the NRC staff (2-9) and does not impact Cycle 2 operation. We conclude that changes to the core reactivity control system have been adequately considered for Cycle 2 operation.

2.1.2 Fuel Rod Design

Although all batches in Davis-Besse Unit 1 Cycle 2 utilize the same Mark B-4 fuel, the Batch 4 assemblies incorporate a slightly lower initial fuel density. The change, from 96 to 94 percent of theoretical density, is a consequence of using a modified fuel fabrication process. The stability (densification resistance) of both fuel types is similar. As a consequence, the initial active fuel length is virtually unchanged for the Batch 4 assemblies. Densification in Davis-Besse Unit 1 Cycle 2 fuel is discussed further in Section 2.3.1.

The fuel pellet end configuration has also changed to a truncated cone dish for Batch 4 as opposed to a spherical dish for the previous three batches. The dish volume remains unchanged. This minor change facilitates manufacturing and does not significantly alter the performance characteristics of the fuel.

2.2.1 Cladding Collapse

Due to the cumulative nature of cladding deformation, creep collapse analyses were performed for the previous first cycle as well as the proposed second cycle of operation. Batches 1B, 2 and 3 are more limiting than Batch 4 due to their previous incore exposure time. That analysis was performed for the most limiting fuel assembly power history using the CROV computer code and procedures described in the topical report BAW-10084PA, Rev. 2 (2-10). The analysis conservatively determined a creep collapse time of 30,000 effective

full power hours (EFPH) of operation. Since the collapse time is greater than the estimated maximum three cycle residence time for any assembly in the Cycle 2 core, we conclude that cladding creep collapse has been adequately considered.

2.2.2 Cladding Stress

The Davis-Besse Unit 1 stress parameters are enveloped by a conservative fuel rod stress analysis. For design evaluation the primary membrane stress must be less than two-thirds of the minimum specified unirradiated yield strength and all stresses must be less than the maximum specified unirradiated yield strength. In all cases, the margin is in excess of 30%. We have examined the Davis-Besse Unit 1 FSAR (Section 4.2.1.4.3) and find the cladding stress analyses were performed for both beginning-of-life (BOL) and end-of-life (EOL) conditions. The results, shown in Table 4.2 of the report, compare cladding circumferential stress levels with the yield and ultimate strength of Zircaloy under a variety of conditions. The cladding stress levels are strongly dependent on the pressure differential across the cladding wall and are limiting (maximum) for BOL when the rod internal pressure is minimum.

We agree that the pressure differential across the cladding wall is a major contributor to the cladding stress level. The external system pressure remains relatively constant (2.200 psia) during normal operation. The differential across the cladding wall is the greatest, therefore, when the rod internal pressure is much less, or

much greater, than the coolant pressure. As discussed in Section 2.2.4, the rod internal pressure does not exceed system pressure during normal operation. Therefore, limiting cladding stress conditions based on rod internal gas pressure exist at BOL. As a result, the analyses presented in the Davis-Besse Unit 1 FSAR apply to Cycle 2 operation.

We also note, however, that fuel swelling, cladding creep, and fuel-cladding mechanical interaction may also contribute to the effects of internal gas pressure on cladding stress levels. In general, these effects are localized and the licensee's design bases (DB-1 FSAR Section 4.2.1.1.1.2 (2-3)) state that "secondary stresses, which are relieved by small material deformation, are permitted to exceed the yield strength." We do not believe that the design criterion for cladding stress will limit the operational flexibility of Davis-Besse Unit 1. Therefore, we conclude that cladding stress limits will not be exceeded during normal operation of Cycle 2 fuel at Davis-Besse Unit 1.

2.2.3 Cladding Strain

The fuel design criteria (DB-1 FSAR Section 4.2.1.1.1.3 (2-3)) specify a 1% limit on cladding plastic strain due to diameter increases resulting from fuel swelling, thermal ratcheting, creep and internal gas pressure. Strain limits were established on the basis of low-cycle fatigue techniques, not to exceed 90% of material fatigue life. The design evaluation, discussed in Section 4.2.1.4

of the DB-1 FSAR (2-3) and Section 4.2.3 of the Reload submittal (2-2), was performed for design pellet burnup and heat generation rate as well as limiting dimensional tolerances. These conditions are considerably beyond those expected for Cycle 2 at Davis-Besse Unit 1. The results show circumferential plastic strain is less than 1% at design EOL burnup, and cumulative fatigue damage after three cycles of operation is less than 90% of material fatigue life. We conclude that the cladding strain and fatigue limits have been adequately considered for Cycle 2 operation.

2.2.4 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (2-11) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Not all of these have been addressed in the licensee's reload application or previous reports. Among those which may affect the operation of the fuel rod is the internal pressure limit. Our current criterion (Standard Review Plan (SRP) 4.2, Section II.A.1(f)) states that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Meeting this criterion is also a condition of acceptance as discussed previously in Section 2.2.2 (cladding stress).

Although the Davis-Besse Unit 1 FSAR (Table 4-21 (2-3)) shows that maximum fuel rod internal pressure does not exceed approximately 2,000 psi, it also describes the use of an internal gas pressure of 3,300 psi to determine fuel cladding internal design conditions. It is not clear whether the limit of rod internal pressure at

2,000 psi is a design criterion or simply an analytical result. The analysis is not described in the reload submittal. Furthermore, we believe (2-12) that some of the analytical methods utilized by B&W may be deficient at high burnups.

In response to a question on this criterion, the licensee has stated (2-13) that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 2. This analysis is based on the use of the B&W TAFY code (2-14) rather than a newer B&W code called TACO (2-15). Although both of these codes are currently approved for use in safety analyses, we believe that only the newer TACO code is capable of correctly calculating fission gas release (and therefore rod pressure) at very high burnups. B&W has responded (2-16) to this concern with an analytical comparison between both codes. In this response, they have stated that the internal fuel rod pressure predicted by TACO is lower than that predicted by TAFY for fuel rod exposures of up to 42,000 MWd/mtU. Although we have not examined the comparison, we note that the analyses exceed the expected exposure in Davis-Besse Unit 1 Cycle 2 by a large margin. We conclude that the rod internal pressure limits have been adequately considered.

2.3 Fuel Thermal Design

There are no major changes between the new Batch 4 fuel and previous batches reinserted in the Cycle 2 core. The decrease in initial fuel density (94% T.D.) results in a slightly altered linear heat rating (LHR) for the fuel based on centerline melt. However, the linear heat rate capacity, as shown

in Table 4-2 of the reload submittal (2-1), was established on the basis of design, rather than as-fabricated, fuel conditions. The fuel conditions applicable to Batch 4 fuel, including resintering data, have been checked with respect to design conditions and found acceptable.

From Table 4-2 of the reload submittal, we also note that two Batch 1B and five Batch 3 fuel assemblies have LHR to centerline melt limits of 20.17 and 20.35 kW/ft, respectively. These LHR limits were established with the TAFY code (2-14) and are below the design value of 20.4 kW/ft established for the remaining Cycle 2 assemblies, including Batch 4. This is due to the fabrication process used for the Cycle 1 fuel. All of Batches 1, 2 and 3 were resintered after initial manufacturing in order to obtain a high density, stable fuel. Subsequently, some, but not all, of the resintered pellets were ground to meet design dimensions. The remaining material was checked on an assembly-by-assembly basis to determine noncompliance with design specifications. The slightly reduced LHR limits resulted for some assemblies. The LHR limit is therefore limiting for these previously irradiated fuel assemblies. The LHR limits are maintained by reactor protection system setpoints.

2.3.1 Fuel Densification

The Davis-Besse Unit 1 reload submittal (2-1) states that the initial fuel pellet density for Batches 1B, 2 and 3 is 96% and that for Batch 4 is 94%. It further states that thermal design limits were established using a terminal density value of 96.5%. The initial-to-final density change, particularly for high density batches, appears to be very small when compared to both recent B&W (2-17) and NRC (2-18)

estimates. However, our examination of the Davis-Besse Unit 1 Fuel Densification Report (2-19) indicates that the analysis was prepared in accordance with an earlier B&W topical report (2-20). The earlier report describes a method wherein an uncertainty in fabricated density is added to the total density change. This method was assumed to apply only to early B&W fuels (i.e., pellets fabricated prior to 1976). However, both the earlier (2-20) and the more recent (2-17) densification models have been approved by the NRC staff (2-21, 2-22). Because one of these methods, the earlier (2-20), was used for Davis-Besse Unit 1, we conclude that the densification process has been adequately considered.

2.3.2 Loss of Coolant Accident (LOCA) Initial Conditions

The average fuel temperature as a function of LHR and lifetime pin pressure data used in the LOCA analysis (Section 7.2 of the Reload submittal) are also calculated with the TAFY code (2-14). B&W has stated (2-1) that the fuel temperature and pin pressure data used in the generic LOCA analysis (2-23) are conservative compared to those calculated for Cycle 2 at Davis-Besse Unit 1.

As previously mentioned in Section 2.2.4 of this evaluation, B&W currently has two fuel performance codes, TAFY (2-14) and TACO (2-15), which could be used to calculate the LOCA initial conditions. The older code, TAFY, has been used for the Cycle 2 LOCA analysis. Recent information (2-24) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO for all Cycle 2 conditions as suggested in Ref. 2-16. The issue involves calculated fuel rod internal gas pressures that are too low at BOL. The rod internal pressures are used to determine swelling and rupture behavior during LOCA. B&W has proposed (Attachment 3 of Ref. 2-24) a method of resolving this issue which has not yet been accepted by the NRC staff. While we have not yet completed the review, we believe the Cycle 2 LOCA initial conditions are acceptable as submitted.

2.4 Material Compatibility

The chemical and material compatibility of possible fuel, cladding and coolant interactions is unchanged from the previous cycle of operation. The impact of this issue on the operational safety of Davis-Besse Unit 1 need not be reconsidered for Cycle 2 operation.

2.5 Operating Experience

B&W has accumulated operating experience with the Mark B 15x15 fuel assembly at all of the eight operating B&W 177-fuel assembly plants. A summary of this operating experience as of September 30, 1979 is given on page 4-3 of Ref. 2-1.

2.5.1 Guide Tube Wear

Significant wear of Zircaloy control rod guide tubes has been observed in facilities designed by Combustion Engineering. Similar wear has also been reported in those facilities designed by Westinghouse. In a letter dated June 13, 1978, we requested information from B&W on the susceptibility of the facilities designed by B&W to guide tube wear. The information provided by B&W in a letter dated January 12, 1979, was insufficient for us to conclude that guide tube wear was not a significant problem in B&W plants. This was documented in our letter to B&W dated August 22, 1979.

Because significant guide tube wear could impede the control rod scram capability and also affect the required coolable geometry of the reactor core, we consider this wear phenomenon a potential safety concern. Therefore, we requested (2-25) additional information from TECo on the wear characteristics of the control rods on the guide tubes in that reactor. The response to this request has not yet been received. The licensee has stated (2-13) that a generic response to this request has been prepared by B&W. The report, B&W Control Rod Guide Tube Wear Generic Report (BAW-1623), has been concurred with by the licensee but has not been received by the NRC.

We have, however, received preliminary information on post-irradiation examinations of identical guide tubes for wear in Rancho Seco spent fuel (2-26). The results of these measurements indicate that through-

wall wear or excessive wall degradation will not likely occur during anticipated fuel residence time for rodded assemblies. On the basis of this preliminary information and the imminent documentation of a complete generic evaluation, we conclude that guide tube wear has been adequately addressed for the Davis-Besse Unit 1 during Cycle 2.

2.5.2 Holddown Spring Failures

The upper end fitting of the B&W Mark B-4 fuel assembly contains a holddown spring to accommodate length changes due to thermal expansion and irradiation growth while providing a positive holddown force for the assembly. On May 14, 1980, a failed holddown spring was discovered by remote video inspection at Davis-Besse Unit 1 (2-27). Further examination ultimately identified a total of 19 failed springs in the Cycle 1 fuel assemblies. Subsequent examination of spent fuel assemblies at other B&W reactors revealed a small number of similar failures at Crystal River 3 (2-28) and Oconee 1 (2-29).

A metallurgical investigation of the spring material (Inconel) indicated that the holddown springs had a high susceptibility to fatigue and stress corrosion cracking. On June 10, 1980, the licensee and B&W met with the NRC staff (2-30) to discuss the problem and the procedure to be utilized for spring replacement. Three potential concerns were raised as a result of the spring failures: (1) loss of holddown force; (2) loose parts; and (3) interference with normal control rod assembly movement. Following

this meeting, we prepared a list of questions (2-31) which were sent to all B&W licensees, including TECo, requesting further information on the holddown spring problem.

TECo responded to our questions in their letter of July 18, 1980 (2-32). With respect to loss of holddown force, results of test data were described in which partial holddown force was maintained even for springs with failures in more than one location. This partial holddown force, coupled with the fuel assembly weight and frictional forces, was judged sufficient to maintain assembly position under all reasonable operating conditions. However, under the most adverse conditions considered, fuel assembly liftoff may occur. Liftoff under these latter conditions was also analyzed. Some upper and lower end fitting wear would be expected to occur, but reactivity changes and impacted loads were expected to be very minor. Our own estimates (2-33) of reactivity change confirm those provided by the licensee. In addition, the fuel assemblies containing broken holddown springs at Davis-Besse Unit 1 were examined for signs of wear at the end fittings and other areas. None was found.

In regard to loose parts (spring fragments) causing damage to reactor vessel internals, B&W has stated (2-30) that all but the very small fragments of a failed spring would be retained within the upper assembly end fitting. This was the case observed at Davis-Besse Unit 1. Any additional pieces would be carried by normal core flow to the

steam generator, where there exists little potential for damage to occur because of the necessarily small size of the pieces. In addition, the licensee has stated (2-32) the Loose Parts Monitoring System (VLPM) at Davis-Besse Unit 1 would be capable of detecting broken springs and pieces of cladding if the mass of the part is 0.007 pounds or greater. We agree that most failed spring fragments would be retained within the fuel assembly. Because of their small size, loose fragments would therefore constitute little additional threat of damage and would most likely be detected by the VLPM.

Concerning control rod insertion capability, the licensee stated that horizontal and vertical motion of a fuel assembly with a failed hold-down spring would be limited by the spacer grid pads, baffle plates and adjacent seated assemblies. Control rod insertion difficulties are not expected with even maximum fuel assembly repositioning and there have been no observed problems in this regard to date. Smaller spring fragments are not expected to block the control rod insertion path since the rods are partially inserted in the guide tubes at all times. Periodic control rod movement tests would confirm CRA insertion and drop time. We conclude that the licensee has adequately addressed all of the concerns in our letter of July 1, 1980 (2-31).

Although the licensee has concluded that the holddown spring failures do not constitute a significant safety concern, those issues concerning additional fuel assembly damage (e.g., lower end fitting wear) still remain. As a consequence, the licensee has replaced the holddown springs in all (133) Cycle 1 fuel assemblies scheduled

for reinsertion in Cycle 2. This modification, which is described in the licensee's letter of July 10, 1980 (2-34), eliminates all failed and suspect holddown springs in the Cycle 2 core. TECo has also committed (2-34) to inspection of all Cycle 2 assemblies at the second refueling outage.

On the basis of the licensee's analysis of the consequences of operating with failed holddown springs, the replacement of all failed and suspect springs, and the licensee's commitment to continued surveillance of the fuel assemblies, we conclude that there is reasonable assurance that the holddown spring issue has been correctly analyzed and does not result in a safety concern for Cycle 2 operation.

2.6 Rod Bow

The licensee has stated that a rod bow penalty has been calculated according to the procedure approved in reference 2-35. The burnup used is the maximum fuel assembly burnup of the batch that contains the limiting (maximum radial x local peak) fuel assembly. For Cycle 2, this burnup is 26,654 MWd/mtU in a Batch 3 assembly. The resultant net rod bow penalty after inclusion of the 1% flow area reduction factor credit is 1.8% reduction in departure from nucleate boiling ratio (DNBR). However, this rod bow penalty is offset by the DNBR margin included in trip setpoints and operating limits. (See Section 3.2.)

2.7 Cladding Strain and Flow Blockage

The licensee has responded (2-13) to our request for information concerning the new fuel cladding strain and fuel assembly flow blockage models described in NUREG-0630.

TECo has reviewed all of the subject information supplied by B&W and is in agreement with the results that calculated peak fuel cladding temperature will remain unchanged or lowered with the use of the new NRC ramp-rate-dependent correlations, and that compliance with 10 CFR 50.46 is assured for Davis-Besse Unit 1.

REFERENCES

2-1 R. P. Crouse (Toledo Edison) letter dated February 11, 1980 to R. Reid (NRC) transmitting Davis-Besse Nuclear Power Station, Unit 1 Reload Report (BAW-1598) dated January, 1980 and revised by Toledo Edison, February 6, 1980.

2-2 R. P. Crouse (Toledo Edison) letter to R. Reid (NRC) dated May 21, 1980 with revision to BAW-1598.

2-3 Davis-Besse Nuclear Power Station Unit 1, Final Safety Analysis Report, Docket 50-346, Toledo Edison Company.

2-4 Safety Evaluation Report related to operation of Davis-Besse Nuclear Power Station Unit 1, U. S. Nuclear Regulatory Commission Report NUREG-0136, December 1976.

2-5 BPRA Retainer Design Report, Babcock and Wilcox Company Report BAW-1496, May 1978.

2-6 W. P. Stewart (Florida Power Corporation) letter to C. Nelson (NRC) on "Crystal River Unit Three Status Report - May 1, 1978," dated May 4, 1978.

2-7 L. Engle (NRC), Meeting Summary of "Tentative Proposal to Amend Operating License for Removal of Burnable Poison Rod Asemblies - Davis Besse, Unit No. 1," dated May 23, 1978.

2-8 L. E. Roe (Toledo Edison) letter to J. F. Stolz (NRC) transmitting "Davis-Besse Nuclear Power Station Unit No. 1 Attachment 1 to Amend Operating License for Removal of the Burnable Poison Rod Assemblies and Orifice Rod Assemblies," (BAW-1489) dated May 26, 1978.

2-9 J. F. Stolz (NRC) letter to L. E. Roe (Toledo Edison) dated June 16, 1978, transmitting Amendment No. 11 to the Davis-Besse Unit 1 operating license.

2-10 Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, Babcock and Wilcox Company Report, BAW-10084P-A, October 1978.

2-11 Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U. S. Nuclear Regulatory Commission Report NUREG-75/087.

2-12 D. F. Ross, Jr. (NRC) letter to J. H. Taylor (B&W) dated January 18, 1978.

2-13 R. P. Crouse (Toledo Edison) letters to T. M. Novak (NRC) dated July 29 and July 30, 1980.

REFERENCES (Continued)

2-14 C. D. Morgan and H. S. Kao, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," Babcock and Wilcox Company Report BAW-10044, May 1972.

2-15 "TACO-Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10087P-A, Rev. 2, August 1977.

2-16 J. H. Taylor (B&W) letter to P. S. Check (NRC), dated July 18, 1978.

2-17 B. J. Buescher and J. W. Pegram, Babcock and Wilcox Model for Predicting In-Reactor Densification, BAW-10083P-A Revision 1, July 1977.

2-18 R. O. Meyer, The Analysis of Fuel Densification, NUREG-0085, July 1976.

2-19 Davis Besse Unit 1 - Fuel Densification Report, BAW-1401 (proprietary), April 1975 and BAW-1420 (non-proprietary), September 1975.

2-20 R. A. Turner, Fuel Densification Report, BAW-10054 Revision 2 (proprietary), May 1973 and BAW-10055 (non-proprietary), February 1973.

2-21 "Technical Report on Densification of Babcock and Wilcox Reactor Fuels," Regulatory Staff, U. S. Atomic Energy Commission, July 6, 1973.

2-22 S. A. Varga (NRC) letter to J. H. Taylor (B&W) on "Evaluation of BAW-10083P, Revision 1," dated May 16, 1977.

2-23 W. L. Bloomfield, et.al., "ECCS Analysis of B&W's 177-FA Raised-Loop NSS," Babcock and Wilcox Company Report BAW-10105, June 1975.

2-24 R. O. Meyer (NRC) memorandum to L. S. Rubenstein (NRC) on "TAFY/TACO Fuel Performance Models in B&W Safety Analyses," dated June 10, 1980.

2-25 R. W. Reid (NRC) letter to L. E. Roe (Toledo Edison) dated November 23, 1979.

2-26 J. J. Mattimoe (Sacramento Municipal Utility District) letter to R. W. Reid (NRC) on "Guide Tube Wear Measurements - Preliminary Results" dated February 15, 1980.

2-27 T. D. Murray (Toledo Edison) letter to J. G. Keppler (NRC/Reg. III) dated May 23, 1980.

2-28 J. A. Hancock (Florida Power) letter to J. P. O'Reilly (NRC/Reg. II) dated May 29, 1980.

2-29 W. O. Parker, Jr. (Duke Power) letter to J. P. O'Reilly (NRC/Reg. II) dated June 6, 1980.

REFERENCES (Continued)

2-30 D. Garner (NRC), "Summary of June 10, 1980 Meeting with Toledo Edison and Babcock & Wilcox Regarding Fuel Assembly Holddown Spring Failures at Davis-Besse 1 and Other B&W Plants," June 30, 1980.

2-31 T. M. Novak (NRC) letter to R. P. Crouse (Toledo Edison) dated July 1, 1980.

2-32 R. P. Crouse (Toledo Edison) letter to T. M. Novak (NRC) dated July 18, 1980.

2-33 V. Stello (NRC) memorandum to K. Goller (NRC) on "Oconee 2 and 3 Core Lift Potential with Higher Core Pressure Drop," dated August 1, 1975.

2-34 R. P. Crouse (Toledo Edison) letter to R. W. Reid (NRC) dated July 10, 1980.

2-35 L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reductions Due to Rod Bow," dated October 18, 1979.

3. Evaluation of Nuclear and Thermal-Hydraulic Design

3.1 Nuclear Design

A core loading diagram for Cycle 2 of Davis-Besse Unit 1 is presented in the reload report (BAW-1598 and Revision 1 of that document) along with enrichment and burnup distributions. The nuclear parameters for Cycle 2 are compared to those for Cycle 1 including reactivity coefficients, boron worths and rod group worths. An analysis of the shutdown margin capability and a radial power map at BOC are also given.

The core physics calculations are performed with the PDQ07 code* which has been reviewed and approved by the NRC staff. This code has been used for analysis of the previous cycle of Davis-Besse Unit 1. The results of the analysis show small differences between Cycle 2 and Cycle 1 values, occasioned by the difference in design cycle lengths (248 EFPD for Cycle 2 vs. 433 EFPD for Cycle 1) and by the fact that the core is not yet in its equilibrium configuration. The analysis of shutdown margin shows that 1.76% Δ k/k exists at EOC compared to the required 1.0% $\Delta k/k$ for hot shutdown.
The calculated radial power distribution at BOC shows adequate margin to limits.

Based on the fact that approved methods have been used to obtain the core characteristics, that margin exists to limiting values of the parameters, and that startup testing will be used to obtain measured values of important parameters, we find the analysis of core parameters to be acceptable.

* PDQ07 Users Manual, BAW-10117PA, January 1977.

3.2 Thermal-Hydraulic Design

The licensee states that the Batch 4 fuel for Cycle 2 is hydraulically and geometrically similar to the fuel remaining in the core from Cycle 1. The thermal-hydraulic design evaluation supporting Cycle 2 operations used the models and methods described in references 3-1, 3-2 and 3-3.

A rod bow penalty was calculated according to the procedure approved in reference 3-4. The resulting rod bow penalty is 1.8 percent after a credit for one percent flow area reduction factor is included.

Table 3.2-1 shows a comparison of the thermal-hydraulic design conditions for Cycles 1 and 2. The flux/flow trip setpoint for Cycle 2 operation has been established as 1.07. This setpoint and other plant operating limits are based on DNBR criteria that contain sufficient margin to offset the rod bow penalty and meet the design minimum DNBR limit of 1.30 calculated using the BAW-2 correlation.

The design coolant flow rate is used in the analysis. However, the minimum flow rate permitted by the Technical Specifications is a factor of 1.025 times the design flow to account for flow rate uncertainty.

The minimum DNBR at 112 percent of full power is 1.79 for Cycle 2 vs. 1.81 for Cycle 1B to account for the slightly different power distribution in Cycle 2. We find the methods used in the above analyses to be acceptable.

Table 3.2-1. Cycles 1B[b] and 2 Thermal-Hydraulic Design Conditions - Davis-Besse

	Cycle 1B	Cycle 2
Design power level, MWt	2772	2772
System pressure, psia	2200	2200
Reactor coolant flow, % design	110	110
Vessel inlet/outlet coolant temp., 100% power, F	557.7/606.3	557.7/606.3
Ref design radial-local power peaking factor	1.71	1.71
Ref design axial flux shape	1.5 cosine with tails	1.5 cosine with tails
Hot channel factors		
Enthalpy rise (F_q)	1.011	1.011
Heat flux (F''_q)	1.014	1.014
Flow area	0.98	0.98
Avg heat flux, 100% power, Btu/h-ft^2	1.86×10^5[a]	1.89×10^5[a]
Max heat flux, 100% power, Btu/h-ft^2	4.78×10^5[a]	4.83×10^5[a]
CHF correlation	BAW-2	BAW-2
Minimum DNBR, (% power)	1.81 (112%)	1.79 (112%)

[a] With thermally expanded fuel rod OD of 0.43075 inch.

[b] After removal of burnable poison and orifice rod assemblies.

REFERENCES

3-1 BPRA Retainer Design Report, BAW-1496, Babcock & Wilcox, Lynchburg, Virginia, May 1978.

3-2 Attachment 1 to Application to Amend Operating License for Removal of Burnable Poison Rod and Orifice Rod Assemblies, BAW-1489, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, May 1978.

3-3 Davis-Besse Unit 1 Fuel Densification Report, BAW-1401, Babcock & Wilcox, Lynchburg, Virginia, April 1975.

3-4 L. S. Rubenstein (USNRC) to J. H. Taylor (B&W), Letter, "Evaluation of Interim Procedure for Calculating DNBR Reductions Due to Rod Bow," October 18, 1979.

4. Evaluation of Transients and Accidents

4.1 Loss of Coolant Flow Transients

The licensee states that each FSAR accident analysis has been examined with respect to changes in Cycle 2 parameters to determine the effect on the Cycle 2 reload and to ensure that thermal performance during hypothetical transients is not degraded. The licensee concludes that in comparing the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of Cycle 2, including the four- and single-pump coastdowns, is bounded by the previously accepted analyses. The initial conditions of the transients in Cycle 2 are bounded by the FSAR and/or the fuel densification report (reference 3-3). We find this evaluation to be acceptable.

4.2 Rod Misoperation Events

The input parameters for the rod misoperation events - uncontrolled rod withdrawal, rod drop and rod ejection - have been compared to those in Cycle 1 and those used in the FSAR analysis. In all cases the Cycle 2 values are bounded by those used in the safety analyses. We conclude that the consequences of these events will not be greater than was shown to be acceptable in the FSAR.

4.3 Fuel Misloading Event

The misloading of fuel in the core is made very unlikely by careful procedures during manufacture and installation of the fuel. Further, analysis of a large number of possible misloadings for another B&W reactor having 177 fuel assemblies shows that any misloading that could produce a violation of fuel design limits when operating at full power would be detected by the

incore system. TECo asserts that this analysis applies to Davis-Besse Unit 1. On this basis, we conclude that the consequences of a potential fuel misloading are acceptable.

4.4 Other Transients and Accidents

Other transients and accidents are treated by examination of the input parameters for each analysis and noting that the Cycle 2 values are bounded by values used in the FSAR safety analysis. Table 4.4-1 presents a comparison of important parameters between the two cycles. It is to be noted that in all cases the Cycle 2 values are bounded by values used in safety analyses.

5. Modifications to Trip Setpoints

IE Bulletin 79-05B, issued on April 21, 1979 requested that all operators of B&W designed reactors modify plant design and procedures to assure a reduction of the likelihood of automatic actuation of the pressurizer power operated relief valve (PORV) during anticipated transients. In the licensee's analyses of potential modifications, consideration of changes in the high reactor coolant system (RCS) pressure reactor trip setpoint and the PORV setpoint was to be included. It was further stated that the modifications shall not increase the frequency of pressurizer code safety operation for these anticipated transients.

In a response to the above request dated May 18, 1979, TECo stated that the high RCS pressure trip would be reduced from a nominal 2,355 psig to 2,300 psig, and that the PORV setpoint would be raised to 2,400 psig. In a letter dated July 6, 1979, we requested that TECo develop Technical Specifications which would, in part, reflect the changes in the two modified setpoints. In accordance with this request, TECo forwarded proposed TS changes in a submittal dated July 13, 1979. The licensee's analysis of the effect of the modified setpoints shows that for a loss of

feedwater event, which TECo considers the maximum overpressure anticipated transient, there is still margin between the maximum RCS pressure achieved and the lowest possible actual actuation point of the PORV. This analysis includes allowances for drift and other potential inaccuracies in the setpoints for both the RCS high pressure trip and the PORV.

While we cannot conclude that actuation of the PORV will no longer take place for anticipated transients, we can conclude the following:

1. The adjustment in setpoints greatly reduces the probability that the PORV will be opened by an anticipated transient.
2. Simultaneous reduction in the RCS high pressure trip and increase in the PORV setpoint will not increase the probability that the pressurizer code safety valves will be operated during anticipated transients.
3. Since no credit was taken for PORV operation in the FSAR accident analyses, the change in setpoint for the PORV does not change any accident analysis for the plant.

Based on the above, we find the modified setpoints and the proposed Technical Specification changes to be acceptable.

6. <u>License Conditions</u>

The Davis-Besse Unit 1 license currently contains the following conditions:

2.C.(3)(f) Prior to startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall modify the auxiliary feedwater system by providing diverse direct current power to one of the redundant auxiliary feedwater trains.

2.C.(3)(g) Prior to startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall modify the emergency core cooling system by providing motor operated valves with control and position indication in the control room in lieu of the manually operated valves in each of the

two crossover connection lines installed between the high pressure makeup pump suction and the low pressure injection discharge.

The design modifications required by these conditions have been reviewed and previously approved. In a letter dated August 6, 1980, TECo informed the NRC that the required modifications are being completed, and that the license conditions should be deleted upon confirmation of completion by the NRC Resident Inspector. Confirmation has been received from the Resident Inspector and the conditions may, therefore, be deleted.

Another license condition reads as follows:

2.C.(3)(e) Prior to startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall modify the reactor coolant system flow indication to meet the single failure criterion with regard to pressure sensing lines to the flow differential pressure transmitters.

The licensee, in its letters dated July 21, August 4, and August 25, 1980, has proposed certain modifications to the Reactor Protection System (RPS) of Davis-Besse Unit 1 to correct the single failure deficiency identified in the above license condition. The present RPS overpower trip based upon reactor coolant flow (differential pressure or delta P) and axial power imbalance utilizes a common set of instrument sensing lines for the four reactor protection channels (flow transmitters) in each of the two reactor coolant loops. As stipulated in the condition to the operating license, this deficiency was to be corrected before the plant returns to power following its first refueling shutdown.

In each primary loop, the reactor coolant flow is detected by measuring delta P developed across a flow tube that is an integral part of the outlet piping of the loop. Each flow tube has two differential pressure taps, a high pressure (HP) tap upstream and a low pressure (LP) tap downstream of the flow tube. Each instrument tap has a root valve. Instrument piping runs from the root valves through the secondary shield wall to HP and LP headers. Four delta P transmitters are connected between the two headers. Each of these transmitters provides a flow signal directly to its corresponding RPS input channel.

The licensee's proposed design includes the installation of a second set of instrument sensing lines from the differential pressure taps in each of the primary loops to the differential pressure transmitters.

The new design begins immediately downstream of existing pressure tap root valves located inside the secondary shield wall where one inch tees will be welded in place. Downstream of these tees each set of redundant lines is routed to two flow transmitters. The redundant sets of sensing lines are separated and routed independently, including a new penetration of the secondary shield wall. These new sensing lines are stainless steel and seismically supported.

One half inch thick 18" x 4' steel plate barriers separate the redundant transmitters. Loops A and B are located on opposite sides of containment. The electrical circuits to these redundant transmitters are being seismically installed and the channel identity maintained in accordance with the requirements of the RPS as described in the Davis-Besse Unit 1 FSAR.

Based on our review of the licensee's submittals, we conclude that the installation of the redundant flow sensing lines to measure flow in each of the primary loops satisfies the requirements of seismic qualification, separa-

tion, and isolation. We, therefore, conclude that with the completion of the installation, as verified by the NRC Resident Inspector, the reactor coolant flow sensing system will satisfy our requirements for meeting the single failure criterion. License Condition 2.C.(3)(e) has been satisfied and may be deleted.

7. Technical Specification Changes

We have reviewed the proposed technical specifications for Cycle 2 operation which include the following changes:

1. All references to two-pump operation have been removed since this mode of operation is not allowed by the license.
2. The reduction of the fuel rod bow DNB penalty (reference 3-4) has permitted the relaxation of certain operating limits.
3. The DNB limits in the bases have been changed from 1.32 to 1.30 to be consistent with the approved limit of the BAW-2 correlation. This results from changing the bases from a 95% reliability at a 99% confidence level to a 95% reliability at a 95% confidence level.

Table 4.4-1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR, densif value	Predicted Cycle 2 value
BOL Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.28	-1.44
EOL Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.45[a]	-1.53
BOL moderator coeff, 10^{-4}, $\Delta k/k/°F$	+0.13	-0.56
EOL moderator coeff, 10^{-4}, $\Delta k/k/°F$	-3.0	-2.73
All rod bank worth (HZP), % $\Delta k/k$	10.0	7.09
Boron reactivity worth (HFP), ppm/1% $\Delta k/k$	100	111
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.39
Max dropped rod worth (HFP), % $\Delta k/k$	0.65	0.20
Initial boron conc (HFP), ppm	1407	1197

[a] -1.77×10^{-5} $\Delta k/k/°F$ was used for steamline failure analysis.

4. The power level trip setpoints relative to the power-to-flow ratio were slightly changed. As typical examples, the power setpoints on Figure 2.2-1, Trip Setpoint for Flux - ΔFlux/Flow was changed from 107.88% to 107% for four-pump operation and from 80.59% to 80.2% for three-pump operation.
5. Limits have been placed on the axial position of the part-length Axial Power Shaping Rods. The limits preclude certain axial power shapes which permit relaxation of other limits (e.g., imbalance). Such limits were not employed in the first cycle.
6. The limiting conditions of operation on axial imbalance and full-length rod insertion have been altered from those for Cycle 1.

The same techniques and models were used to derive the revised Technical Specifications as were used to derive those for Cycle 1. The procedure used for the part-length rod limits has been used on other B&W designed reactors.
On the basis that previously approved methods are used and that the changes from the first cycle Technical Specifications are not great, we conclude that the new specifications for Cycle 2 operation are acceptable.

In addition to the changes described above, the Technical Specifications issued by this amendment reflect the modification to the RCS high pressure trip setpoint and the inclusion of the trip setpoint for the PORV. Two changes proposed by the licensee have not been incorporated since no bases for the changes were submitted. These are changes to the Action statements for Sections 3.2.1 and 3.2.5.

8. 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 con-

cluded 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 or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

9. 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 and does not involve a significant decrease in a safety margin, the amendment 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.

Dated: October 1, 1980

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-346

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. NPF-3, issued to The Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees), which revised the license and Technical Specifications 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 its date of issuance.

This amendment modifies the Technical Specifications to permit operation for a second cycle and to reflect changes in the reactor coolant system high pressure and power operated relief valve setpoints. The amendment also deletes three satisfied License Conditions, 2.C.(3)(e), 2.C.(3)(f) and 2.C.(3)(g).

The applications for the amendment comply 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment 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 the issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated February 11, 1980, as revised and supplemented, and July 13, 1979, (2) Amendment No. 33 to License No. NPF-3, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, DC, and at the Ida Rupp Public Library, 310 Madison Street, Port Clinton, Ohio. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 1st day of October 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing