SOUTHERN COMPANY
Energy to Serve Your World
FNP-125-NRC-DC
JANUARY 4, 2002

MR. FRANK RINALDI
U. S. NUCLEAR REGULATORY COMMISSION
MAIL STOP O-8H12 PD II-1
WASHINGTON, DC 20555

DEAR SIR,

ATTACHED ARE THE FOLLOWING AMENDMENTS FOR FARLEY NUCLEAR PLANT TECHNICAL SPECIFICATIONS

UNIT 1 AMENDMENT NO. 151

UNIT 2 AMENDMENT NO. 143

UNIT 1 CORE OPERATING LIMITS REPORT REVISION 1 CYCLE 18 DATED DECEMBER 2001

UNIT 2 CORE OPERATING LIMITS REPORT REVISION 1 CYCLE 15 DATED DECEMBER 2001

REVISION 10 OF THE TECHNICAL SPECIFICATIONS BASES FOR FARLEY NUCLEAR PLANT.

PLEASE REPLACE YOUR COPY OF THE EFFECTIVE PAGES WITH THE ATTACHED REVISED COPIES.

IF YOU HAVE QUESTIONS PLEASE CALL ME AT 334-899-5156 EXTENSION 3402.

SINCERELY,

Donnie Hardy

DONNIE HARDY

DOCUMENT CONTROL SUPERVISOR

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A001

THE FOLLOWING AMEMDMENTS ARE BEING ISSUED AND ARE TO BE FILED IN THE TECHNICAL SPECIFICATIONS.

UNIT 1 AMEMDMENT 151
UNIT 2 AMENDMENT 143

THE TRANSMITTAL SHOWS UNIT 1 AMENDMENT 151 AND 152
UNIT 2 AMENDMENT 143 AND 144

YOU WILL NOT HAVE ANYTHING TO FILE FOR AMENDMENT 152 AND 144 – THIS DID NOT AFFECT ANY TECHNICAL SPECIFICATION PAGE – THIS IS AN UPDATE FOL ONLY.

UNIT 1 AMENDMENT 150 AND UNIT 2 AMENDMENT 142 WILL BE IMPLEMENTED AT A LATER DATE.

ATTACHMENT TO LICENSE AMENDMENT NO. 151

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

ATTACHMENT TO LICENSE AMENDMENT NO. 143

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2.0-1	2.0-1
2.0-2	-
3.3.1-20	3.3.1-20
3.3.1-21	3.3.1-21
3.4.1-1	3.4.1-1
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B 3.4.1-2	B 3.4.1-2
B 3.4.1-3	B 3.4.1-3

Insert Colr:

U1 Core Operating Limits Report Cycle 18

U2 Core Operating Limits Report Cycle 15

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2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Table 3.3.1-1 (page 6 of 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
18.	Reactor Trip Breakers [j]	1,2	2 trains	R, V	SR 3.3.1.4	NA	NA
		3 [a], 4 [a], 5 [a]	2 trains	C, V	SR 3.3.1.4	NA	NA
19.	Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	U	SR 3.3.1.4	NA	NA
		3 [a], 4 [a], 5 [a]	1 each per RTB	C	SR 3.3.1.4	NA	NA
20.	Automatic Trip Logic	1,2	2 trains	Q, V	SR 3.3.1.5	NA	NA
		3 [a], 4 [a], 5 [a]	2 trains	C, V	SR 3.3.1.5	NA	NA

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(j) Including any reactor trip bypass breaker that is racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_O \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[T \frac{1}{(1+\tau_6 s)} - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ * °F.

P is the measured pressurizer pressure, psig.
P' is the nominal pressurizer operating pressure = * psig.

K_1 = *	K_2 = * /°F	K_3 = * /psi
$\tau_1 \geq$ * sec	$\tau_2 \leq$ * sec	
τ_4 = * sec	$\tau_5 \leq$ * sec	$\tau_6 \leq$ * sec

$f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

$f_1(\Delta I)$ =	*{* + (q_t - q_b)}	when (q_t - q_b) ≤ * % RTP
	* % of RTP	when * % RTP < (q_t - q_b) ≤ * % RTP
	*{(q_t - q_b) - *}	when (q_t - q_b) > * % RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

* as specified in the COLR

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_O \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T" is the reference T_{avg} at RTP, ≤ * °F.

K_4 = *

K_5 = */°F for increasing T_{avg}
K_5 = */°F for decreasing T_{avg}

K_6 = */°F when T > T"
K_6 = */°F when T ≤ T"

$\tau_3 \geq$ * sec

$\tau_4 =$ * sec

$\tau_5 \leq$ * sec

$\tau_6 \leq$ * sec

$f_2(\Delta I)$ = *% RTP for all ΔI.

* as specified in the COLR

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method, and ≥ the limit specified in the COLR.

APPLICABILITY: MODE 1.

---NOTE---

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within the limits.	12 hours
SR 3.4.1.4	-----NOTE----- Not required to be performed until 7 days after ≥ 90% RTP. ----- Verify by measurement that RCS total flow rate is within the limits.	18 months

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, the monthly operating report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient, and any corrective action necessary to prevent recurrence.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Reactor Core Safety Limits for THERMAL POWER, Reactor Coolant System highest loop average temperature and pressurizer pressure for Safety Limit 2.1.1,

2. SHUTDOWN MARGIN limit for MODES 2 (with $k_{eff} < 1$), 3, 4, and 5 for LCO 3.1.1,

3. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm and 100 ppm surveillance limits for LCO 3.1.3,

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Shutdown Bank Insertion Limits for LCO 3.1.5,

5. Control Bank Insertion Limit for LCO 3.1.6,

6. Heat Flux Hot Channel Factor F_Q^{RTP} limits, K(Z) figure, W(Z) values, and $F_Q(Z)$ Penalty Factors for LCO 3.2.1,

7. Nuclear Enthalpy Rise Hot Channel Factor limits, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, for LCO 3.2.2,

8. Axial Flux Limits for LCO 3.2.3,

9. Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for Table 3.3.1-1,

10 Reactor Coolant System pressure, temperature, and flow in LCO 3.4.1,

11. Refueling Operations Boron Concentration for LCO 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

   (Methodology for LCOs 3.1.1 - SHUTDOWN MARGIN, 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limit, 3.1.6 - Control Bank Insertion Limits, 3.2.3 - Axial Flux Difference, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor and 3.9.1 – Boron Concentration.)

2. WCAP-10216-P-A, Rev.1A, "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification," February 1994 (W Proprietary).

   (Methodology for LCOs 3.2.3 - Axial Flux Difference and 3.2.1 - Heat Flux Hot Channel Factor.)

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor and LCO 3.4.1- RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

(Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs. (Westinghouse Proprietary)

(Methodology for minimum RCS flow determination using the elbow tap measurement)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

(Methodology for LCO 3.9.1 – Boron Concentration)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance

(continued)

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report (continued)

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.10 Steam Generator Tube Inspector Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging effort.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission within 12 months following the completion of the inspection. This Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a Reportable Event and shall be reported pursuant to 10 CFR 50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

Changed Pages List
For ITS Bases
Revision 10

Replace the following pages of the Technical Specifications Bases with the attached revised pages. The revised pages are identified as **Revision 10.** They contain vertical lines indicating area of changes (except LOEP has no rev bars).

Remove	**Instructions**
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B 2.1.1-3	Replace
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B 3.8.10-2	0
B 3.8.10-3	0
B 3.8.10-4	0
B 3.8.10-5	0
Section B 3.9 Tab	
B 3.9.1-1	0
B 3.9.1-2	0
B 3.9.1-3	0
B 3.9.1-4	0
B 3.9.2-1	3
B 3.9.2-2	3
B 3.9.2-3	3
B 3.9.2-4	3
B 3.9.2-5	3
B 3.9.3-1	0
B 3.9.3-2	0
B 3.9.3-3	0
B 3.9.3-4	0
B 3.9.3-5	4
B 3.9.4-1	0
B 3.9.4-2	0
B 3.9.4-3	0
B 3.9.4-4	0
B 3.9.5-1	0
B 3.9.5-2	0
B 3.9.5-3	0
B 3.9.5-4	0
B 3.9.6-1	0
B 3.9.6-2	0
B 3.9.6-3	0

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur on the limiting fuel rod and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience DNB; and

b. The hottest fuel pellet in the core must not experience centerline fuel melting.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNB limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The Reactor Trip System Functions (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions (i.e., resulting from a Condition I or II event) for Reactor Coolant System (RCS) temperature, pressure, flow, ΔI and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses provide more restrictive limits to ensure that the SLs are not exceeded.

BASES

SAFETY LIMITS

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience DNB; and

b. The hottest fuel pellet in the core must not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operations, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 7.2.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. The indicated limit is based on the average of two control board readings. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. The indicated limit is based on the average of two control board readings. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNB design criterion throughout

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

each analyzed transient. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables — pressurizer pressure, RCS average temperature, and RCS total flow rate — to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

RCS total flow rate is based on two elbow tap measurements from each loop and contains a measurement error of 2.3% based on Δp measurements from the cold leg elbow taps, which are correlated to past precision heat balance measurements or performing a precision heat balance at the beginning of the current cycle. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 2.4%.

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

BASES

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." The conditions that define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

(continued)

BASES

ACTIONS (continued)

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is a qualitative verification performed using the installed flow indicators on the main control board fed by elbow tap measurements. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential significant flow degradation and to verify operation within safety analysis assumptions.

(continued)

JOSEPH M. FARLEY NUCLEAR PLANT

CORE OPERATING LIMITS REPORT

UNIT 1 - CYCLE 18

DECEMBER 2001

REVISION 1

APPROVED FOR ISSUE:

CDCollins 1.3.02

OPERATIONS MANAGER / DATE

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for FNP UNIT 1 CYCLE 18 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The Technical Requirement affected by this report is listed below:

13.1.1 SHUTDOWN MARGIN - MODES 1 and 2 (with $k_{eff} \geq 1$)

The Technical Specifications affected by this report are listed below:

2.1.1 Reactor Core Safety Limits for THERMAL POWER
3.1.1 SHUTDOWN MARGIN - MODES 2 (with $k_{eff} < 1$), 3, 4 and 5
3.1.3 Moderator Temperature Coefficient
3.1.5 Shutdown Bank Insertion Limits
3.1.6 Control Bank Insertion Limits
3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$
3.2.2 Nuclear Enthalpy Rise Hot Channnel Factor - $F_{\Delta H}^{N}$
3.2.3 Axial Flux Difference
3.3.1 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Setpoint Parameter Values for Table 3.3.1-1
3.4.1 RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature, and RCS Total Flow Rate
3.9.1 Boron Concentration

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using NRC-approved methodologies, including those specified in Technical Specification 5.6.5.

2.1 SHUTDOWN MARGIN - MODES 1 AND 2 (with $k_{eff} \geq 1.0$) (Technical Requirement 13.1.1)

2.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77 percent $\Delta k/k$.

2.2 SHUTDOWN MARGIN - MODES 2 (with $k_{eff} < 1.0$), 3, 4 and 5 (Specification 3.1.1)

2.2.1 Modes 2 ($k_{eff} < 1.0$), 3 and 4 - The SHUTDOWN MARGIN shall be greater than or equal to 1.77 percent $\Delta k/k$.

2.2.2 Mode 5 - The SHUTDOWN MARGIN shall be greater than or equal to 1.0 percent $\Delta k/k$.

2.3 Moderator Temperature Coefficient (Specification 3.1.3)

2.3.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less than or equal to $+0.7 \times 10^{-4}$ $\Delta k/k/°F$ for power levels up to 70 percent RTP with a linear ramp to 0 $\Delta k/k/°F$ at 100 percent RTP.

The EOL/ARO/RTP-MTC shall be less negative than -4.3×10^{-4} $\Delta k/k/°F$.

2.3.2 The MTC Surveillance limits are:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to -3.65×10^{-4} $\Delta k/k/°F$.

The 100 ppm/ARO/RTP-MTC should be less negative than -4.0×10^{-4} $\Delta k/k/°F$.

where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero THERMAL POWER
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER

2.4 Shutdown Bank Insertion Limits (Specification 3.1.5)

2.4.1 The shutdown banks shall be withdrawn to a position greater than or equal to 225 steps.

2.5 Control Bank Insertion Limits (Specification 3.1.6)

2.5.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.6 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3.2.1)

2.6.1 $F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z)$ for P > 0.5

$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z)$ for P ≤ 0.5

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

2.6.2 $F_Q^{RTP} = 2.50$

2.6.3 K(Z) is provided in Figure 2.

2.6.4 $F_Q(Z) \leq \frac{F_Q^{RTP} * K(Z)}{P * W(Z)}$ for P > 0.5

$F_Q(Z) \leq \frac{F_Q^{RTP} * K(Z)}{0.5 * W(Z)}$ for P ≤ 0.5

2.6.5 W(Z) values are provided in Figures 4 through 7.

2.6.6 The $F_Q(Z)$ penalty factors are provided in Table 1.

2.7 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^{N}$ (Specification 3.2.2)

2.7.1 $F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} * \left(1 + PF_{\Delta H} * (1 - P)\right)$

where: $P = \dfrac{THERMAL\,POWER}{RATED\,THERMAL\,POWER}$

2.7.2 $F_{\Delta H}^{RTP} = 1.70$

2.7.3 $PF_{\Delta H} = 0.3$

2.8 Axial Flux Difference (Specification 3.2.3)

2.8.1 The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 3.

2.9 Boron Concentration (Specification 3.9.1)

2.9.1 The boron concentration shall be greater than or equal to 2000 ppm.[1]

2.10 Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)

2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the safety limits specified in Figure 8.

2.11 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Setpoint Parameter Values for Table 3.3.1-1 (Specification 3.3.1)

2.11.1 The Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Setpoint Parameter Values for TS Table 3.3.1-1 are listed in COLR Tables 2 and 3.

2.12 RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature, and RCS Total Flow Rate (Specification 3.4.1)

2.12.1 RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature and RCS Total Flow Rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2209 psig;
b. RCS average temperature ≤ 580.3°F; and
c. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method and ≥ 264,200 GPM when using the elbow tap method.

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1 This concentration bounds the condition of $k_{eff} \leq 0.95$ (all rods in less the most reactive rod) and subcriticality (all rods out) over the entire cycle. This concentration includes additional boron to address uncertainties and B^{10} depletion.

Table 1

$F_Q(Z)$ PENALTY FACTOR

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor
4644	1.020
4848	1.023
5052	1.021
5256	1.020

Notes:

1. The Penalty Factor, to be applied to $F_Q(Z)$ in accordance with SR 3.2.1.2, is the maximum factor by which $F_Q(Z)$ is expected to increase over a 39 EFPD interval (surveillance interval of 31 EFPD plus the maximum allowable extension not to exceed 25% of the surveillance interval per SR 3.0.2) starting from the burnup at which the $F_Q(Z)$ was determined.
2. Linear interpolation is adequate for intermediate cycle burnups.
3. For all cycle burnups outside the range of the table, a penalty factor of 1.020 shall be used.

Table 2

Reactor Trip System Instrumentation - Overtemperature ΔT (OTΔT)
Setpoint Parameter Values

$T' \leq 577.2°F$	$P' = 2235$ psig	
$K_1 = 1.17$	$K_2 = 0.017/°F$	$K_3 = 0.000825/psi$
$\tau_1 \geq 30$ sec	$\tau_2 \leq 4$ sec	
$\tau_4 = 0$ sec	$\tau_5 \leq 6$ sec	$\tau_6 \leq 6$ sec

$f_1(\Delta I) =$	$-2.48\ \{23 + (q_t - q_b)\}$	when $(q_t - q_b) \leq -23\%$ RTP
	0% of RTP	when -23% RTP $< (q_t - q_b) \leq 15\%$ RTP
	$2.05\ \{(q_t - q_b) - 15\}$	when $(q_t - q_b) > 15\%$ RTP

Table 3

Reactor Trip System Instrumentation - Overpower ΔT (OPΔT)
Setpoint Parameter Values

$T'' \leq 577.2°F$

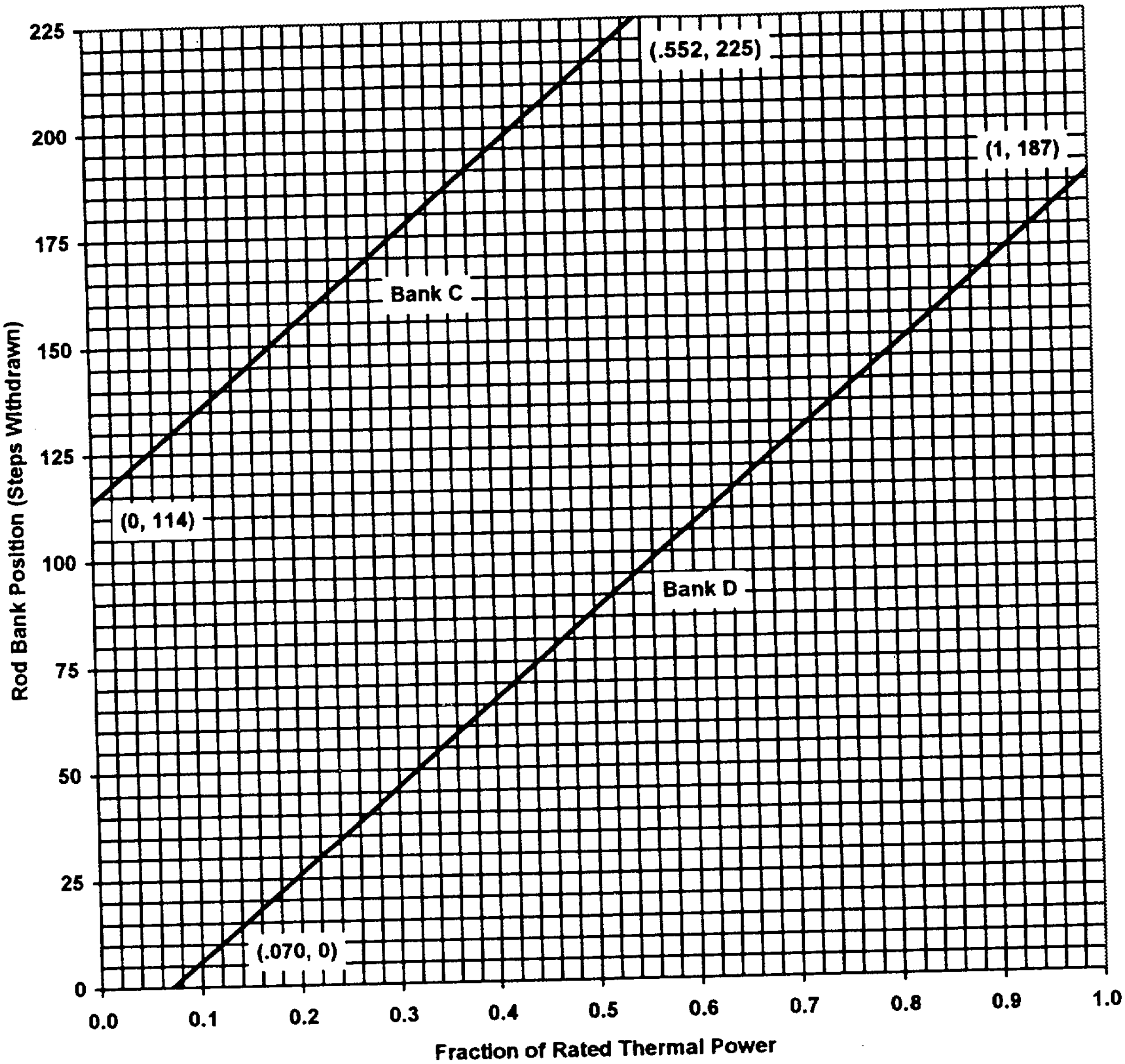
$K_4 = 1.10$	$K_5 = 0.02/°F$ for increasing T_{avg} $K_5 = 0/°F$ for decreasing T_{avg}	$K_6 = 0.00109/°F$ when $T > T''$ $K_6 = 0/°F$ when $T \leq T''$

$\tau_3 \geq 10$ sec
$\tau_4 = 0$ sec
$\tau_5 \leq 6$ sec
$\tau_6 \leq 6$ sec

$f_2(\Delta I) = 0\%$ RTP for all ΔI

Figure 1
Rod Bank Insertion Limits versus Rated Thermal Power

Fully Withdrawn – 225 to 231 steps, inclusive



Fully Withdrawn shall be the condition where control rods are at a position within the interval ≥ 225 and ≤ 231 steps withdrawn.

Note: The Rod Bank Insertion Limits are based on the control bank withdrawal sequence A, B, C, D and a control bank tip-to-tip distance of 128 steps.

Figure 2
K(Z) – Normalized $F_Q(Z)$ as a Function of Core Height

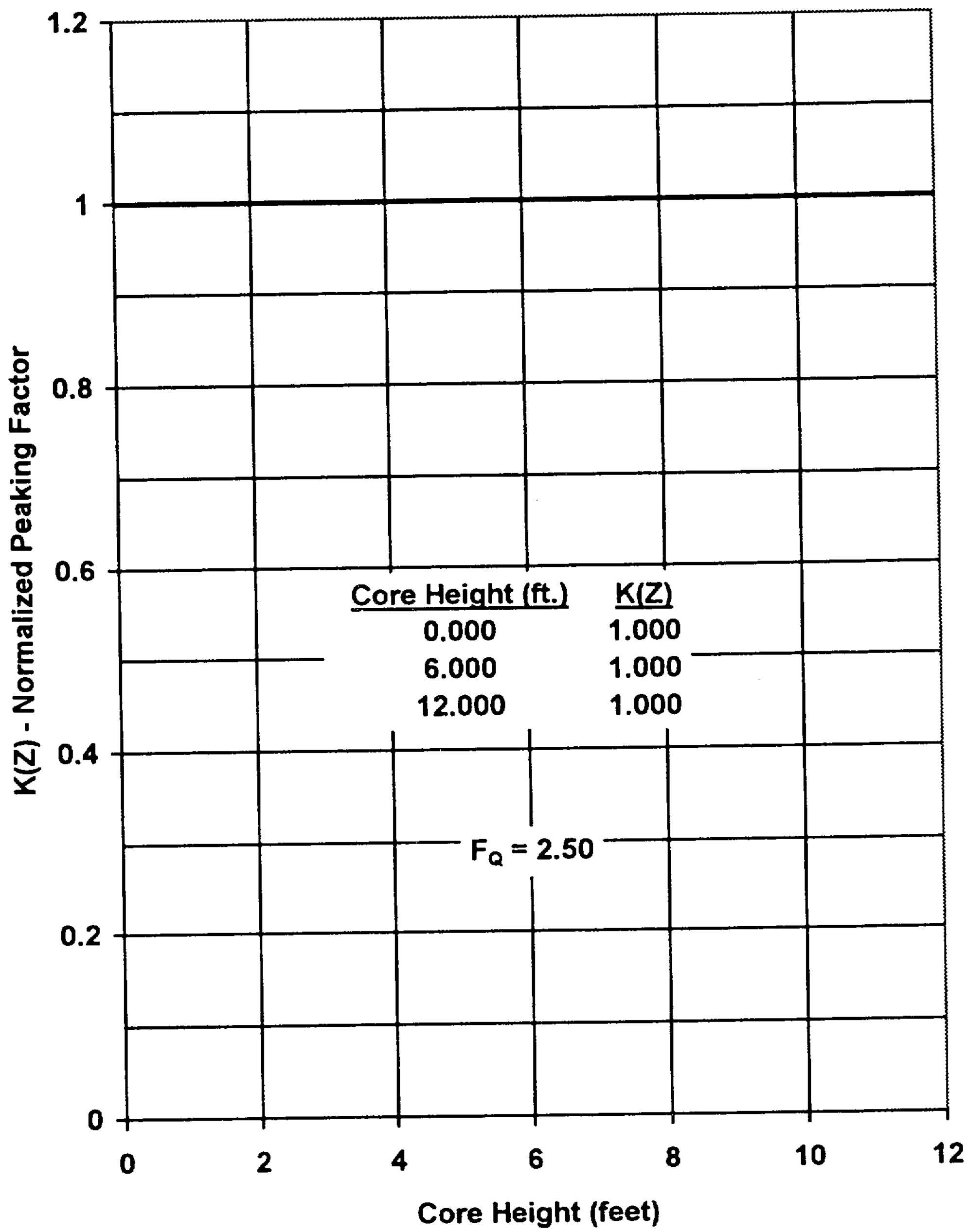


Figure 3
Axial Flux Difference Limits as a Function of Rated Thermal Power for RAOC

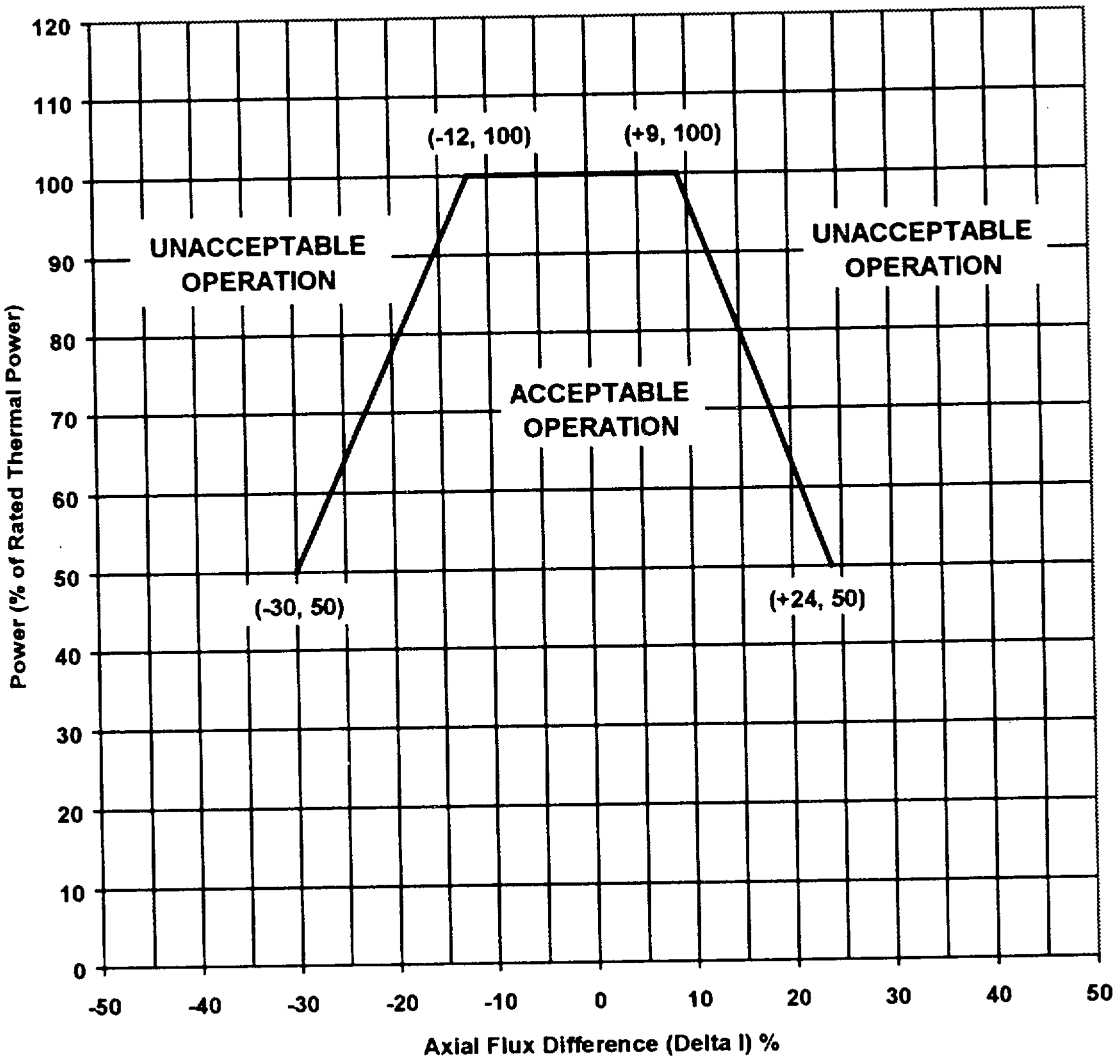
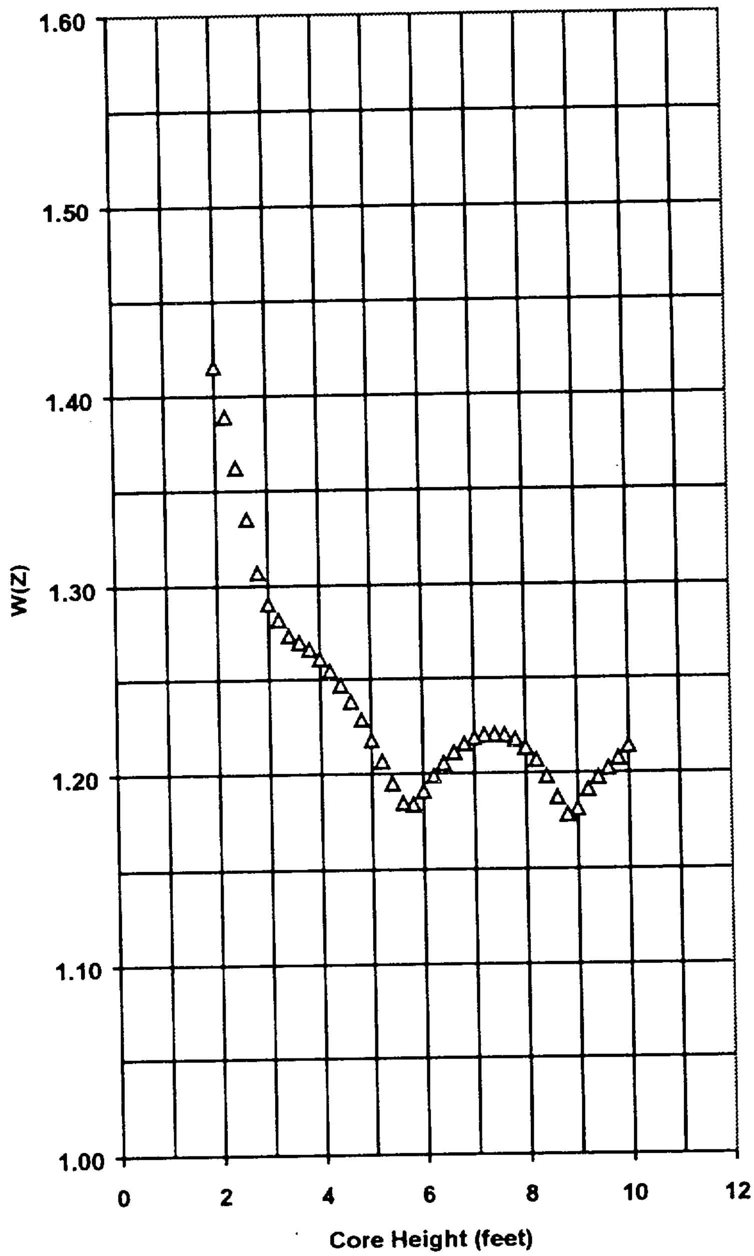


Figure 4
RAOC W(Z) at 150 MWD/MTU

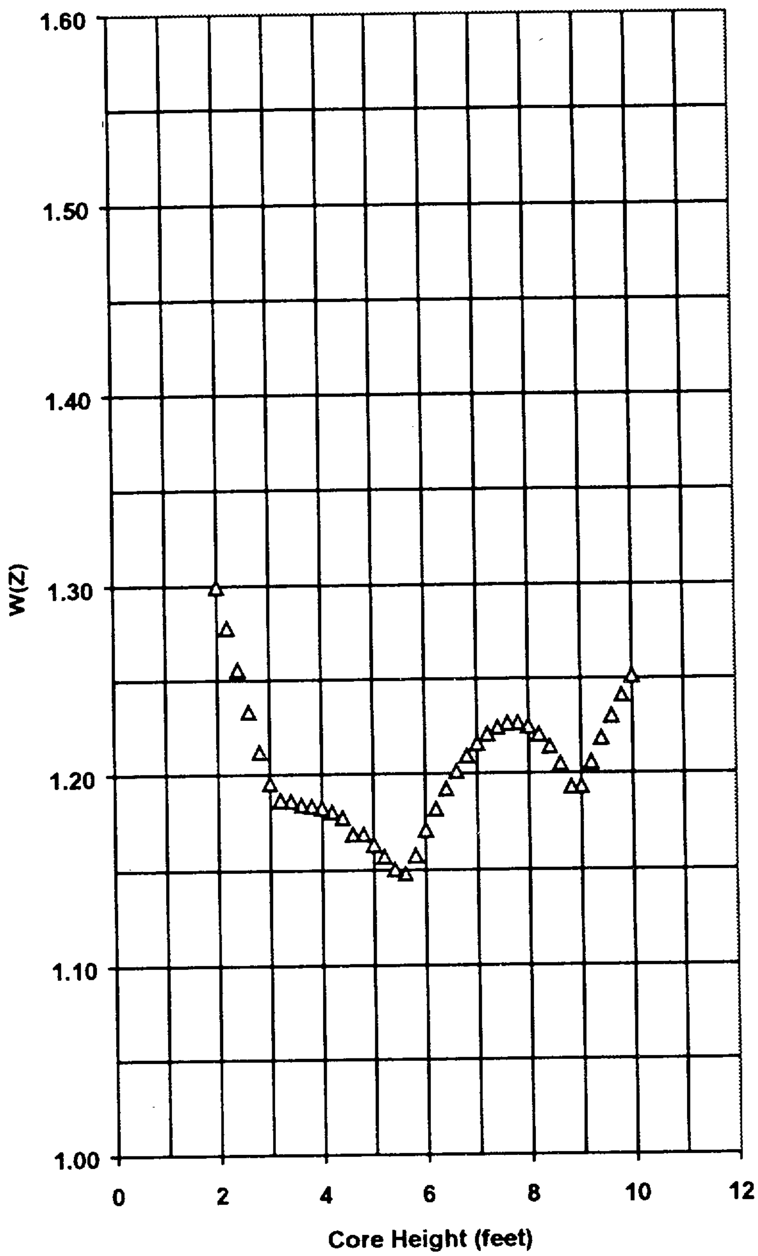


	Axial Point	Elevation (feet)	BOL W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2141
	12	9.80	1.2076
	13	9.60	1.2024
	14	9.40	1.1975
	15	9.20	1.1911
	16	9.00	1.1812
	17	8.80	1.1783
	18	8.60	1.1871
	19	8.40	1.1982
	20	8.20	1.2069
	21	8.00	1.2134
	22	7.80	1.2178
	23	7.60	1.2201
	24	7.40	1.2205
	25	7.20	1.2201
	26	7.00	1.2186
	27	6.80	1.2156
	28	6.60	1.2111
	29	6.40	1.2055
	30	6.20	1.1989
	31	6.00	1.1908
	32	5.80	1.1840
	33	5.60	1.1848
	34	5.40	1.1951
	35	5.20	1.2068
	36	5.00	1.2180
	37	4.80	1.2287
	38	4.60	1.2383
	39	4.40	1.2470
	40	4.20	1.2546
	41	4.00	1.2609
	42	3.80	1.2661
	43	3.60	1.2698
	44	3.40	1.2734
	45	3.20	1.2819
	46	3.00	1.2905
	47	2.80	1.3072
	48	2.60	1.3353
	49	2.40	1.3624
	50	2.20	1.3892
	51	2.00	1.4156
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 5
RAOC W(Z) at 4000 MWD/MTU

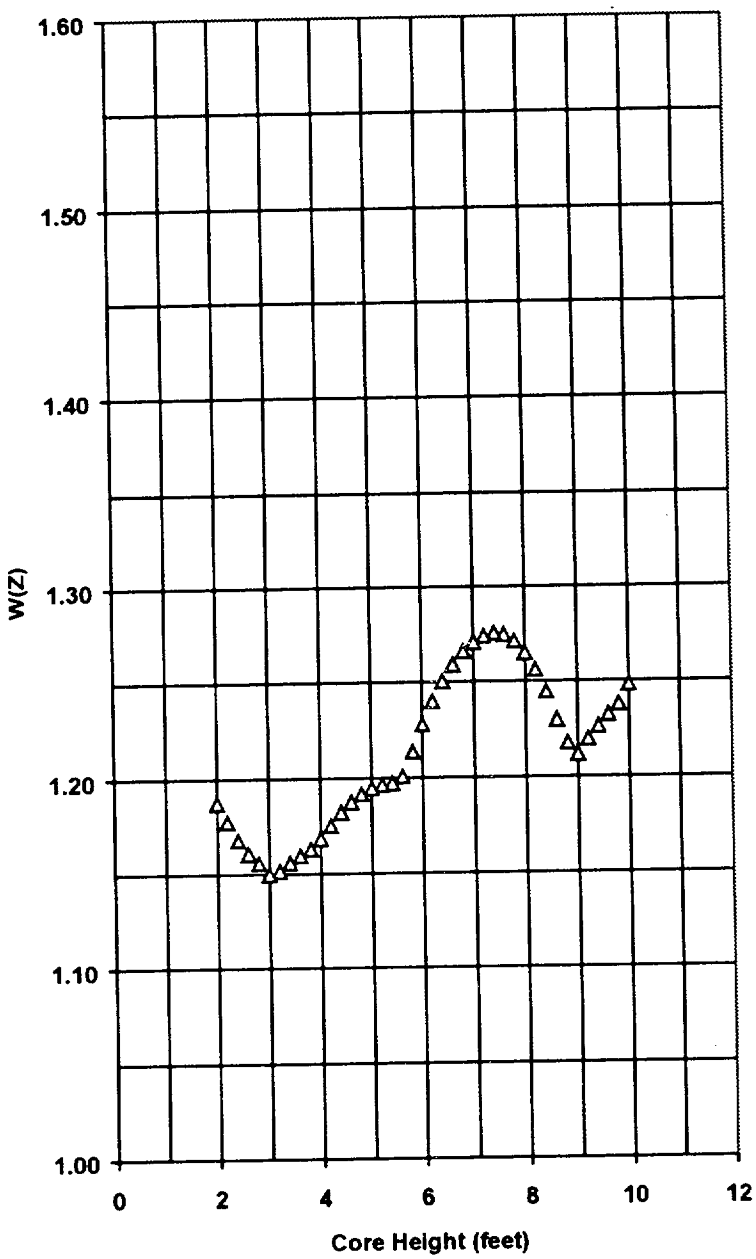


	Axial Point	Elevation (feet)	MOL-1 W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2512
	12	9.80	1.2412
	13	9.60	1.2303
	14	9.40	1.2185
	15	9.20	1.2058
	16	9.00	1.1931
	17	8.80	1.1928
	18	8.60	1.2049
	19	8.40	1.2139
	20	8.20	1.2205
	21	8.00	1.2248
	22	7.80	1.2268
	23	7.60	1.2266
	24	7.40	1.2243
	25	7.20	1.2208
	26	7.00	1.2158
	27	6.80	1.2093
	28	6.60	1.2013
	29	6.40	1.1921
	30	6.20	1.1817
	31	6.00	1.1708
	32	5.80	1.1574
	33	5.60	1.1479
	34	5.40	1.1502
	35	5.20	1.1569
	36	5.00	1.1632
	37	4.80	1.1687
	38	4.60	1.1687
	39	4.40	1.1775
	40	4.20	1.1804
	41	4.00	1.1826
	42	3.80	1.1835
	43	3.60	1.1842
	44	3.40	1.1861
	45	3.20	1.1866
	46	3.00	1.1955
	47	2.80	1.2119
	48	2.60	1.2334
	49	2.40	1.2557
	50	2.20	1.2779
	51	2.00	1.2997
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 6
RAOC W(Z) at 10000 MWD/MTU

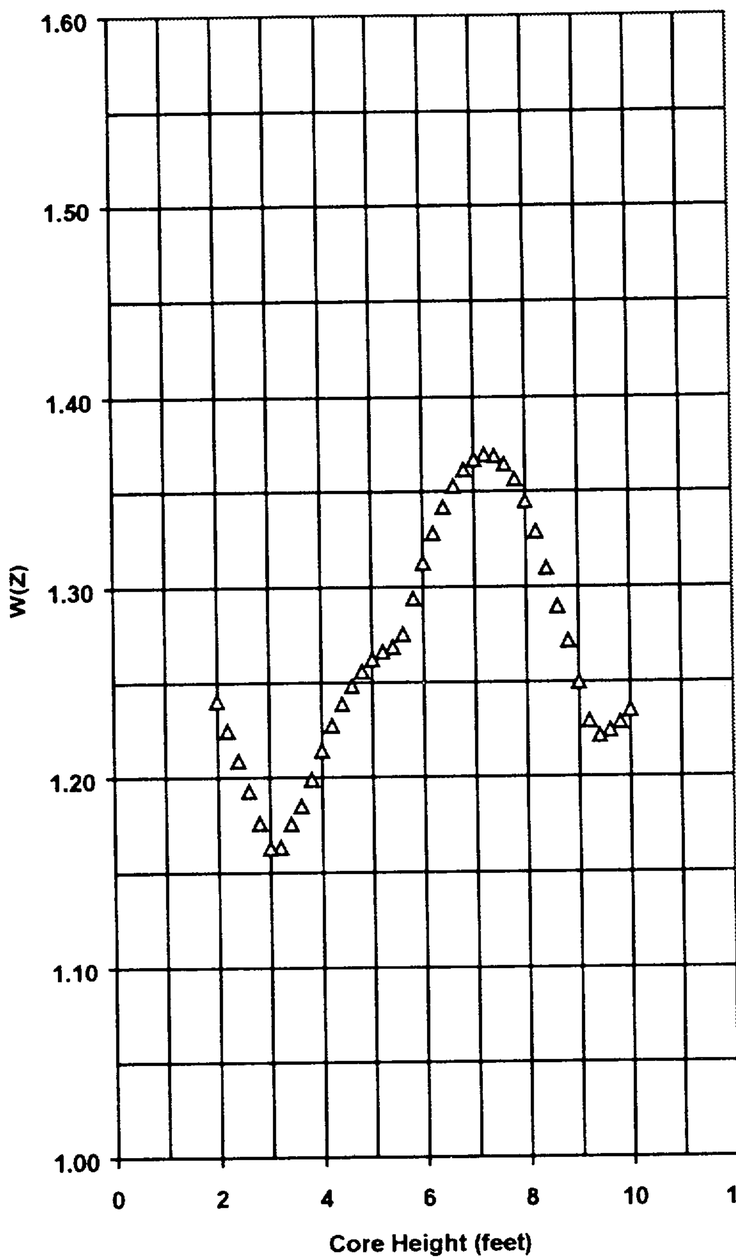


	Axial Point	Elevation (feet)	MOL-2 W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2482
	12	9.80	1.2377
	13	9.60	1.2323
	14	9.40	1.2262
	15	9.20	1.2196
	16	9.00	1.2116
	17	8.80	1.2175
	18	8.60	1.2297
	19	8.40	1.2444
	20	8.20	1.2561
	21	8.00	1.2650
	22	7.80	1.2712
	23	7.60	1.2747
	24	7.40	1.2755
	25	7.20	1.2738
	26	7.00	1.2707
	27	6.80	1.2659
	28	6.60	1.2590
	29	6.40	1.2502
	30	6.20	1.2397
	31	6.00	1.2280
	32	5.80	1.2136
	33	5.60	1.2007
	34	5.40	1.1972
	35	5.20	1.1965
	36	5.00	1.1948
	37	4.80	1.1918
	38	4.60	1.1875
	39	4.40	1.1821
	40	4.20	1.1753
	41	4.00	1.1685
	42	3.80	1.1629
	43	3.60	1.1595
	44	3.40	1.1559
	45	3.20	1.1517
	46	3.00	1.1499
	47	2.80	1.1559
	48	2.60	1.1607
	49	2.40	1.1680
	50	2.20	1.1777
	51	2.00	1.1876
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 7
RAOC W(Z) at 18000 MWD/MTU

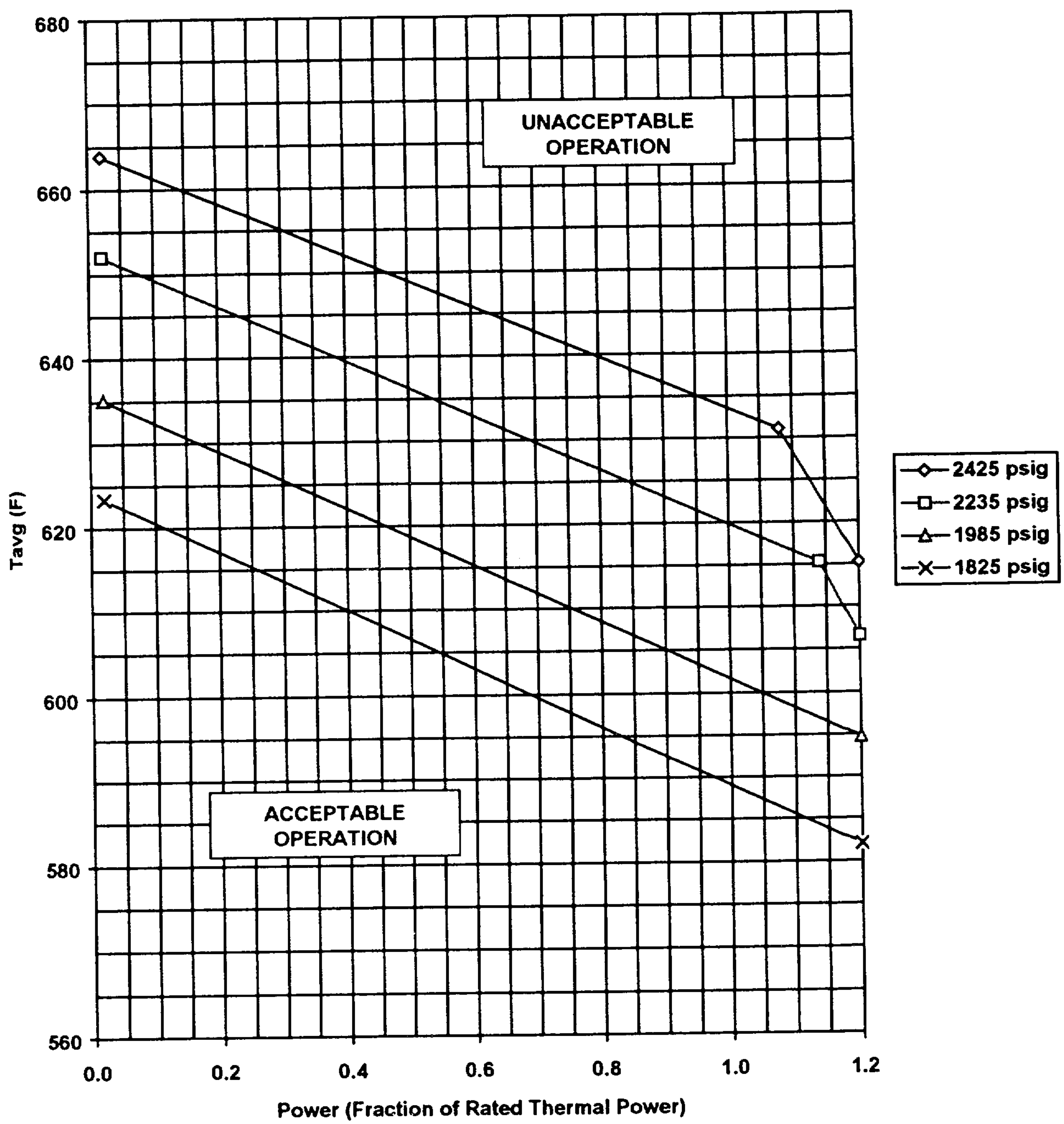


	Axial Point	Elevation (feet)	EOL W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2352
	12	9.80	1.2287
	13	9.60	1.2245
	14	9.40	1.2216
	15	9.20	1.2294
	16	9.00	1.2499
	17	8.80	1.2717
	18	8.60	1.2897
	19	8.40	1.3097
	20	8.20	1.3289
	21	8.00	1.3447
	22	7.80	1.3564
	23	7.60	1.3645
	24	7.40	1.3688
	25	7.20	1.3696
	26	7.00	1.3670
	27	6.80	1.3614
	28	6.60	1.3528
	29	6.40	1.3417
	30	6.20	1.3280
	31	6.00	1.3125
	32	5.80	1.2939
	33	5.60	1.2754
	34	5.40	1.2686
	35	5.20	1.2662
	36	5.00	1.2621
	37	4.80	1.2559
	38	4.60	1.2480
	39	4.40	1.2387
	40	4.20	1.2274
	41	4.00	1.2143
	42	3.80	1.1991
	43	3.60	1.1850
	44	3.40	1.1754
	45	3.20	1.1632
	46	3.00	1.1628
	47	2.80	1.1760
	48	2.60	1.1930
	49	2.40	1.2091
	50	2.20	1.2248
	51	2.00	1.2405
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 8
Reactor Core Safety Limits



JOSEPH M. FARLEY NUCLEAR PLANT

CORE OPERATING LIMITS REPORT

UNIT 2 - CYCLE 15

DECEMBER 2001

REVISION 1

APPROVED FOR ISSUE:

CDCollins 1.3.02

OPERATIONS MANAGER / DATE

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for FNP UNIT 2 CYCLE 15 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The Technical Requirement affected by this report is listed below:

13.1.1 SHUTDOWN MARGIN - MODES 1 and 2 (with $k_{eff} \geq 1$)

The Technical Specifications affected by this report are listed below:

2.1.1 Reactor Core Safety Limits for THERMAL POWER
3.1.1 SHUTDOWN MARGIN - MODES 2 (with $k_{eff} < 1$), 3, 4 and 5
3.1.3 Moderator Temperature Coefficient
3.1.5 Shutdown Bank Insertion Limits
3.1.6 Control Bank Insertion Limits
3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$
3.2.2 Nuclear Enthalpy Rise Hot Channnel Factor - $F_{\Delta H}^N$
3.2.3 Axial Flux Difference
3.3.1 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Setpoint Parameter Values for Table 3.3.1-1
3.4.1 RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature, and RCS Total Flow Rate
3.9.1 Boron Concentration

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using NRC-approved methodologies, including those specified in Technical Specification 5.6.5.

2.1 SHUTDOWN MARGIN - MODES 1 AND 2 (with $k_{eff} \geq 1.0$) (Technical Requirement 13.1.1)

2.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77 percent Δk/k.

2.2 SHUTDOWN MARGIN - MODES 2 (with $k_{eff} < 1.0$), 3, 4 and 5 (Specification 3.1.1)

2.2.1 Modes 2 ($k_{eff} < 1.0$), 3 and 4 - The SHUTDOWN MARGIN shall be greater than or equal to 1.77 percent Δk/k.

2.2.2 Mode 5 - The SHUTDOWN MARGIN shall be greater than or equal to 1.0 percent Δk/k.

2.3 Moderator Temperature Coefficient (Specification 3.1.3)

2.3.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less than or equal to $+0.7 \times 10^{-4}$ Δk/k/°F for power levels up to 70 percent RTP with a linear ramp to 0 Δk/k/°F at 100 percent RTP.

The EOL/ARO/RTP-MTC shall be less negative than -4.3×10^{-4} Δk/k/°F.

2.3.2 The MTC Surveillance limits are:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to -3.65×10^{-4} Δk/k/°F.

The 100 ppm/ARO/RTP-MTC should be less negative than -4.0×10^{-4} Δk/k/°F.

where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero THERMAL POWER
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER

2.4 Shutdown Bank Insertion Limits (Specification 3.1.5)

2.4.1 The shutdown banks shall be withdrawn to a position greater than or equal to 225 steps.

2.5 Control Bank Insertion Limits (Specification 3.1.6)

2.5.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.6 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3.2.1)

2.6.1 $F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z)$ for P > 0.5

$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z)$ for P ≤ 0.5

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

2.6.2 $F_Q^{RTP} = 2.50$

2.6.3 K(Z) is provided in Figure 2.

2.6.4 $F_Q(Z) \leq \frac{F_Q^{RTP} * K(Z)}{P * W(Z)}$ for P > 0.5

$F_Q(Z) \leq \frac{F_Q^{RTP} * K(Z)}{0.5 * W(Z)}$ for P ≤ 0.5

2.6.5 W(Z) values are provided in Figures 4 through 7.

2.6.6 The $F_Q(Z)$ penalty factors are provided in Table 1.

2.7 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^{N}$ (Specification 3.2.2)

2.7.1 $F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} * \left(1 + PF_{\Delta H} * (1 - P)\right)$

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

2.7.2 $F_{\Delta H}^{RTP} = 1.70$

2.7.3 $PF_{\Delta H} = 0.3$

2.8 Axial Flux Difference (Specification 3.2.3)

2.8.1 The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 3.

2.9 Boron Concentration (Specification 3.9.1)

2.9.1 The boron concentration shall be greater than or equal to 2000 ppm.[1]

2.10 Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)

2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the safety limits specified in Figure 8.

2.11 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Setpoint Parameter Values for Table 3.3.1-1 (Specification 3.3.1)

2.11.1 The Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for TS Table 3.3.1-1 are listed in COLR Tables 2 and 3.

2.12 RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature, and RCS Total Flow Rate (Specification 3.4.1)

2.12.1 RCS DNB parameters for pressurizer pressure, RCS average temperature and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2209 psig;

b. RCS average temperature ≤ 580.3°F; and

c. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method and ≥ 264,200 GPM when using the elbow tap method.

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1 This concentration bounds the condition of $k_{eff} \leq 0.95$ (all rods in less the most reactive rod) and subcriticality (all rods out) over the entire cycle. This concentration includes additional boron to address uncertainties and B^{10} depletion.

Table 1

$F_Q(Z)$ PENALTY FACTOR

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor
30	1.029
354	1.031
557	1.031
761	1.029
965	1.026
1372	1.020
4428	1.020
4632	1.021
4836	1.029
5040	1.038
5243	1.036
6262	1.020

Notes:

1. The Penalty Factor, to be applied to $F_Q(Z)$ in accordance with SR 3.2.1.2, is the maximum factor by which $F_Q(Z)$ is expected to increase over a 39 EFPD interval (surveillance interval of 31 EFPD plus the maximum allowable extension not to exceed 25% of the surveillance interval per SR 3.0.2) starting from the burnup at which the $F_Q(Z)$ was determined.
2. Linear interpolation is adequate for intermediate cycle burnups.
3. For all cycle burnups outside the range of the table, a penalty factor of 1.020 shall be used.

Table 2

Reactor Trip System Instrumentation - Overtemperature ΔT (OTΔT) Setpoint Parameter Limits

$T' \leq 577.2°F$	$P' = 2235$ psig	
$K_1 = 1.17$	$K_2 = 0.017/°F$	$K_3 = 0.000825/psi$
$\tau_1 \geq 30$ sec	$\tau_2 \leq 4$ sec	
$\tau_4 = 0$ sec	$\tau_5 \leq 6$ sec	$\tau_6 \leq 6$ sec

$f_1(\Delta I) =$	$-2.48\ \{23 + (q_t - q_b)\}$	when $(q_t - q_b) \leq -23\%$ RTP
	0% of RTP	when -23% RTP $< (q_t - q_b) \leq 15\%$ RTP
	$2.05\ \{(q_t - q_b) - 15\}$	when $(q_t - q_b) > 15\%$ RTP

Table 3

Reactor Trip System Instrumentation - Overpower ΔT (OPΔT)
Setpoint Parameter Values

$T'' \leq 577.2°F$

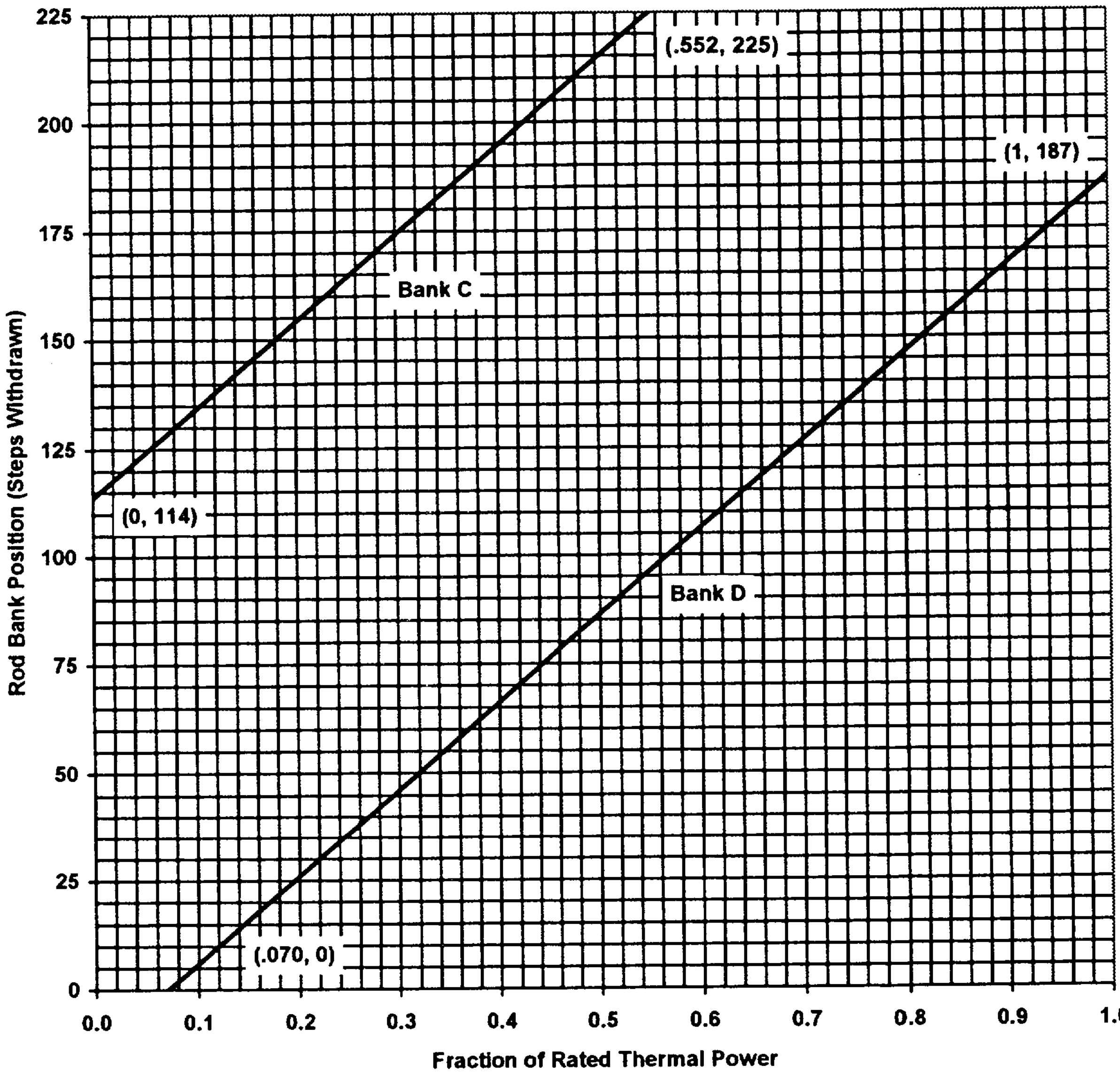
$K_4 = 1.10$	$K_5 = 0.02/°F$ for increasing T_{avg} $K_5 = 0/°F$ for decreasing T_{avg}	$K_6 = 0.00109/°F$ when $T > T''$ $K_6 = 0/°F$ when $T \leq T''$

$\tau_3 \geq 10$ sec
$\tau_4 = 0$ sec
$\tau_5 \leq 6$ sec
$\tau_6 \leq 6$ sec

$f_2(\Delta I) = 0\%$ RTP for all ΔI

Figure 1
Rod Bank Insertion Limits versus Rated Thermal Power

Fully Withdrawn – 225 to 231 steps, inclusive



Fully Withdrawn shall be the condition where control rods are at a position within the interval ≥ 225 and ≤ 231 steps withdrawn.

Note: The Rod Bank Insertion Limits are based on the control bank withdrawal sequence A, B, C, D and a control bank tip-to-tip distance of 128 steps.

Figure 2
K(Z) – Normalized $F_Q(Z)$ as a Function of Core Height

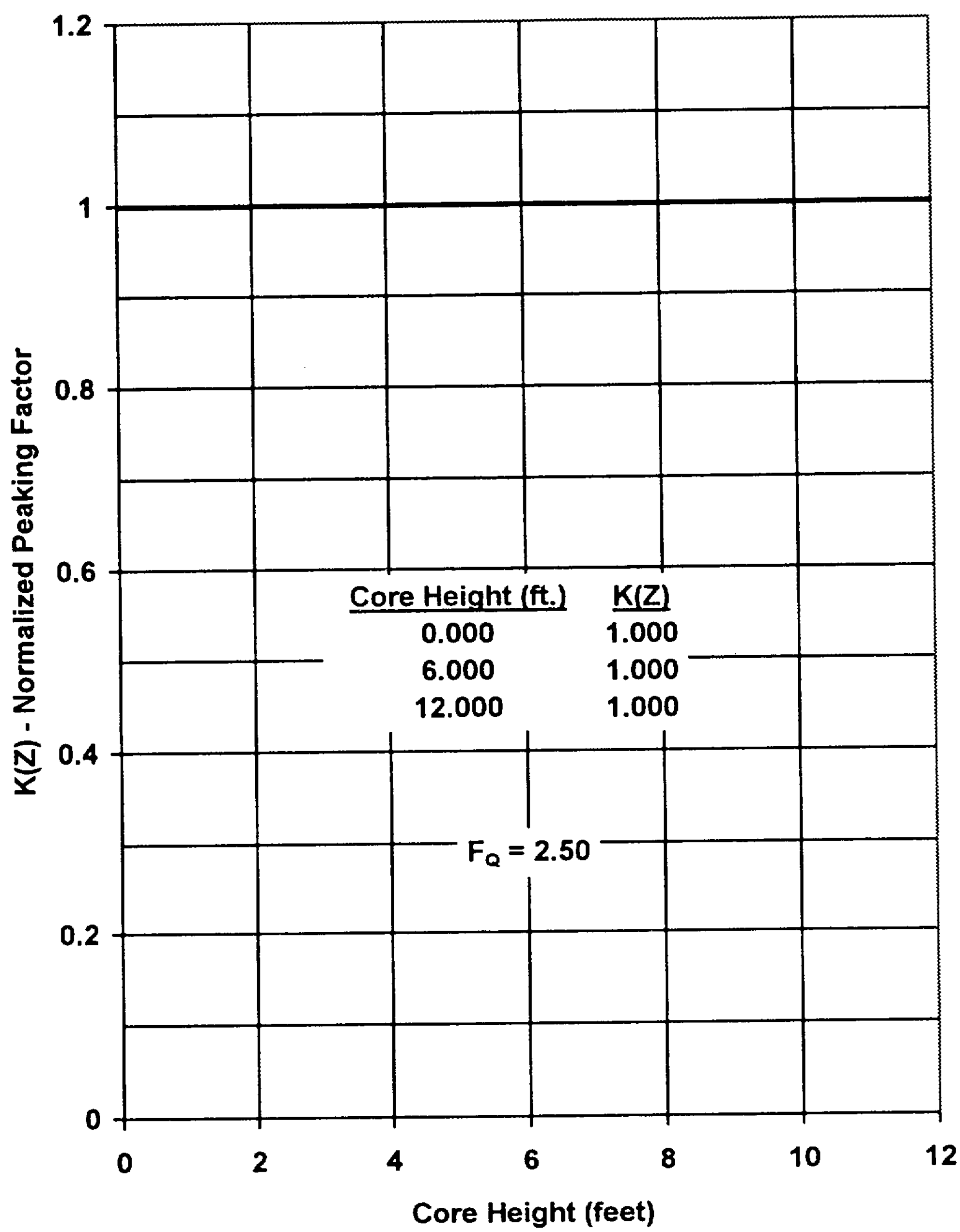


Figure 3
Axial Flux Difference Limits as a Function of Rated Thermal Power for RAOC

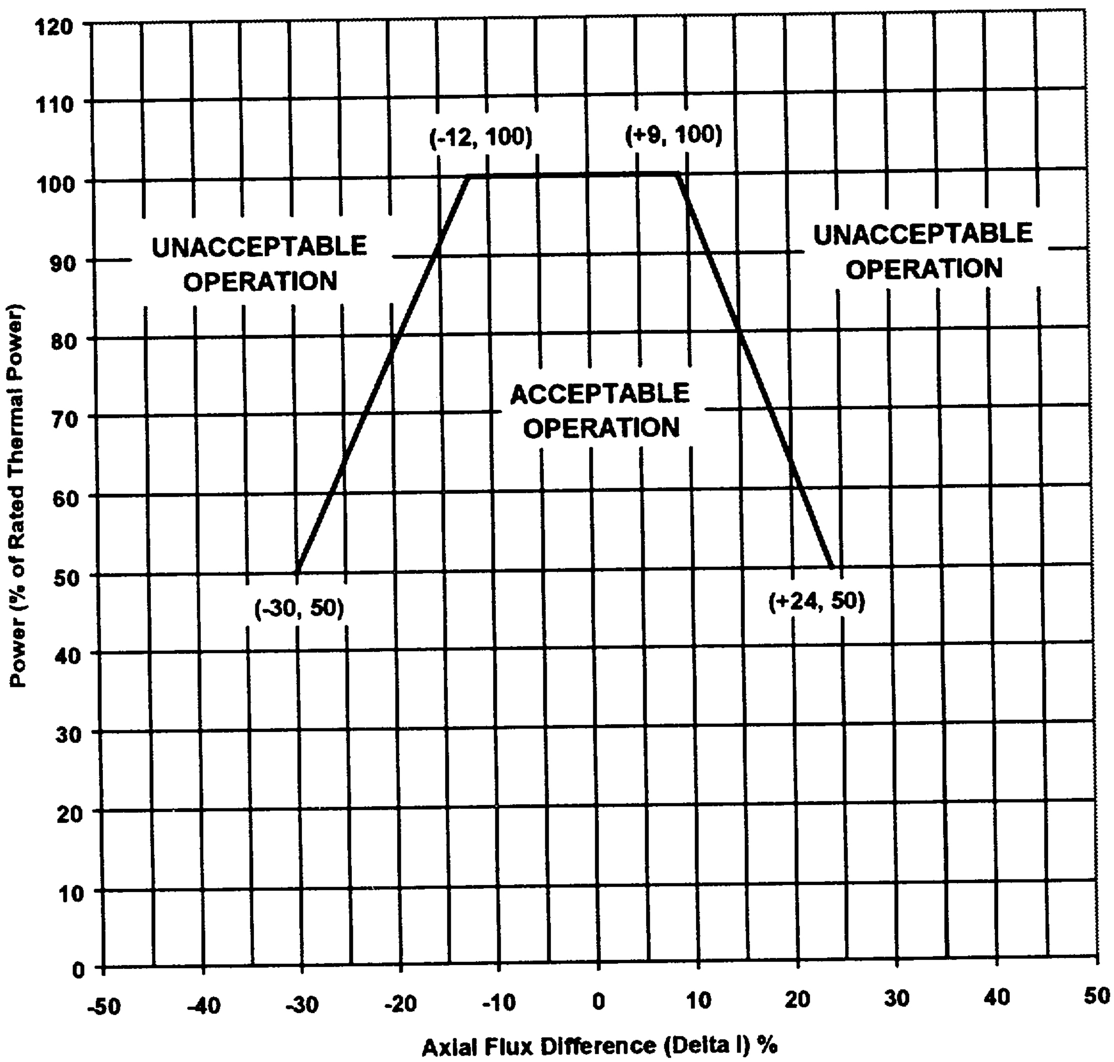
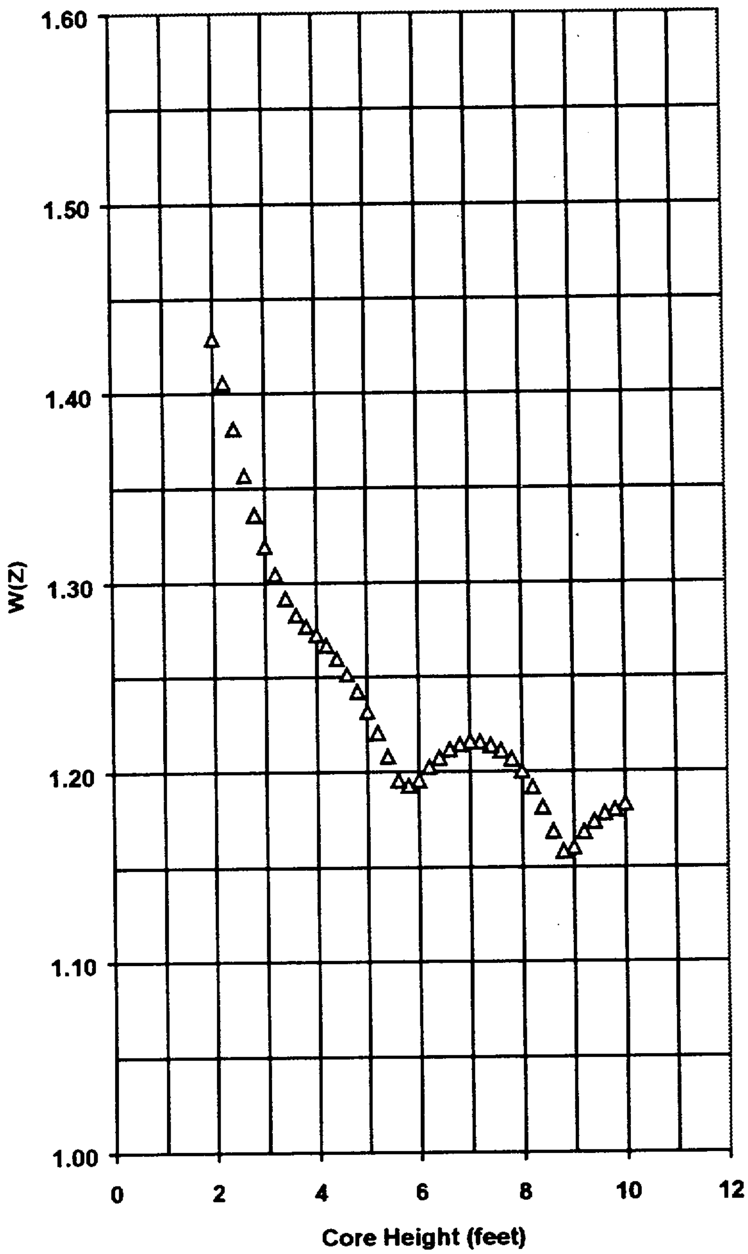


Figure 4
RAOC W(Z) at 150 MWD/MTU

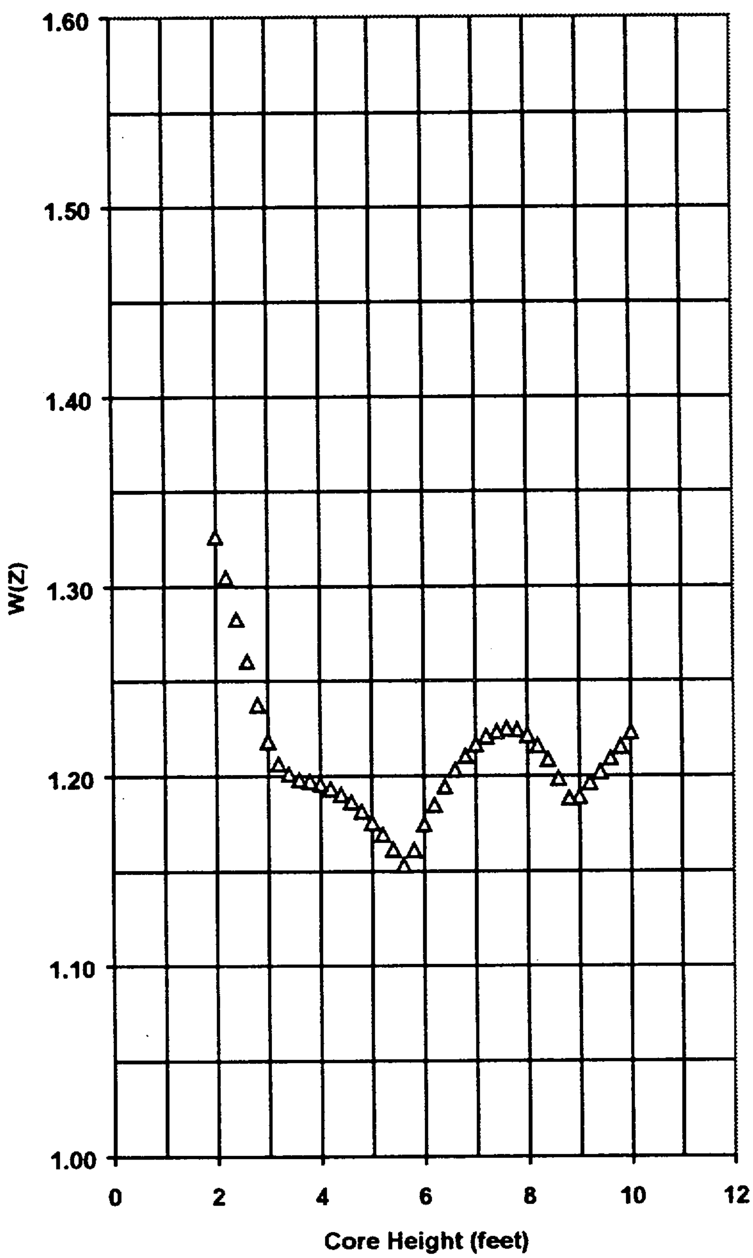


	Axial Point	Elevation (feet)	BOL W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.1830
	12	9.80	1.1798
	13	9.60	1.1780
	14	9.40	1.1736
	15	9.20	1.1682
	16	9.00	1.1600
	17	8.80	1.1582
	18	8.60	1.1686
	19	8.40	1.1813
	20	8.20	1.1919
	21	8.00	1.2003
	22	7.80	1.2067
	23	7.60	1.2112
	24	7.40	1.2138
	25	7.20	1.2156
	26	7.00	1.2158
	27	6.80	1.2143
	28	6.60	1.2116
	29	6.40	1.2075
	30	6.20	1.2026
	31	6.00	1.1957
	32	5.80	1.1929
	33	5.60	1.1955
	34	5.40	1.2082
	35	5.20	1.2205
	36	5.00	1.2318
	37	4.80	1.2423
	38	4.60	1.2516
	39	4.40	1.2600
	40	4.20	1.2669
	41	4.00	1.2726
	42	3.80	1.2770
	43	3.60	1.2830
	44	3.40	1.2920
	45	3.20	1.3049
	46	3.00	1.3197
	47	2.80	1.3362
	48	2.60	1.3568
	49	2.40	1.3813
	50	2.20	1.4056
	51	2.00	1.4290
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 5
RAOC W(Z) at 4000 MWD/MTU

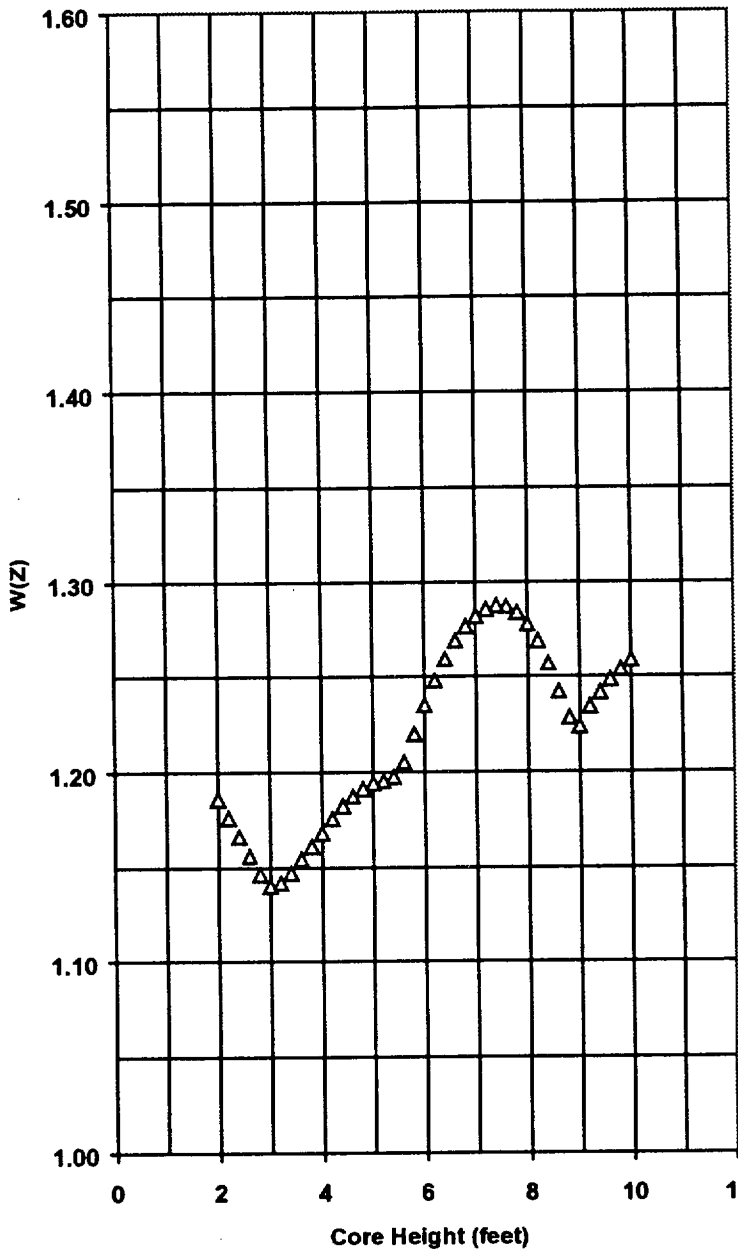


	Axial Point	Elevation (feet)	MOL-1 W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2231
	12	9.80	1.2155
	13	9.60	1.2093
	14	9.40	1.2020
	15	9.20	1.1964
	16	9.00	1.1890
	17	8.80	1.1881
	18	8.60	1.1987
	19	8.40	1.2087
	20	8.20	1.2162
	21	8.00	1.2215
	22	7.80	1.2243
	23	7.60	1.2249
	24	7.40	1.2234
	25	7.20	1.2208
	26	7.00	1.2166
	27	6.80	1.2106
	28	6.60	1.2033
	29	6.40	1.1946
	30	6.20	1.1846
	31	6.00	1.1744
	32	5.80	1.1608
	33	5.60	1.1530
	34	5.40	1.1611
	35	5.20	1.1690
	36	5.00	1.1752
	37	4.80	1.1813
	38	4.60	1.1863
	39	4.40	1.1903
	40	4.20	1.1934
	41	4.00	1.1958
	42	3.80	1.1973
	43	3.60	1.1982
	44	3.40	1.2012
	45	3.20	1.2066
	46	3.00	1.2185
	47	2.80	1.2377
	48	2.60	1.2606
	49	2.40	1.2828
	50	2.20	1.3048
	51	2.00	1.3263
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 6
RAOC W(Z) at 10000 MWD/MTU

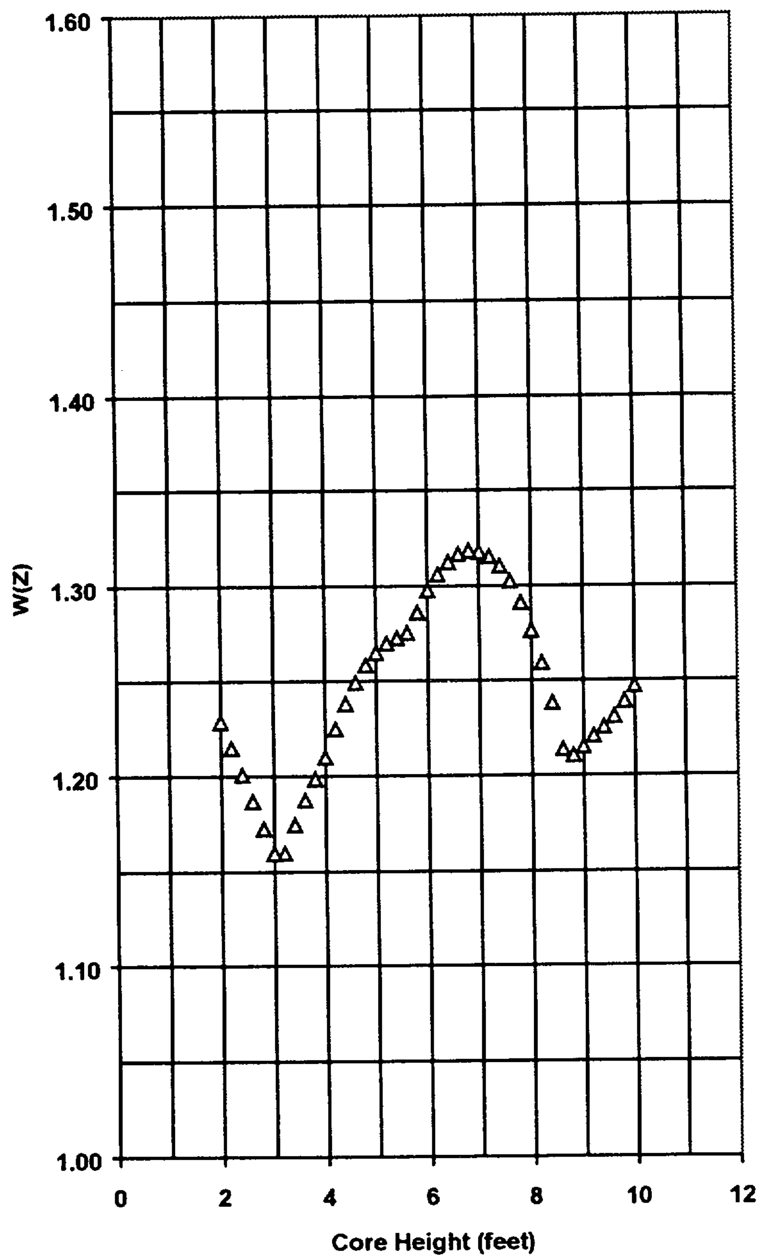


	Axial Point	Elevation (feet)	MOL-2 W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2585
	12	9.80	1.2537
	13	9.60	1.2482
	14	9.40	1.2415
	15	9.20	1.2342
	16	9.00	1.2238
	17	8.80	1.2283
	18	8.60	1.2420
	19	8.40	1.2567
	20	8.20	1.2687
	21	8.00	1.2776
	22	7.80	1.2837
	23	7.60	1.2869
	24	7.40	1.2874
	25	7.20	1.2854
	26	7.00	1.2818
	27	6.80	1.2764
	28	6.60	1.2688
	29	6.40	1.2591
	30	6.20	1.2477
	31	6.00	1.2349
	32	5.80	1.2199
	33	5.60	1.2052
	34	5.40	1.1974
	35	5.20	1.1955
	36	5.00	1.1942
	37	4.80	1.1913
	38	4.60	1.1873
	39	4.40	1.1821
	40	4.20	1.1756
	41	4.00	1.1681
	42	3.80	1.1611
	43	3.60	1.1545
	44	3.40	1.1473
	45	3.20	1.1422
	46	3.00	1.1404
	47	2.80	1.1467
	48	2.60	1.1562
	49	2.40	1.1662
	50	2.20	1.1761
	51	2.00	1.1859
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 7
RAOC W(Z) at 17000 MWD/MTU



	Axial Point	Elevation (feet)	EOL W(Z)
*	1	12.00	1.0000
*	2	11.80	1.0000
*	3	11.60	1.0000
*	4	11.40	1.0000
*	5	11.20	1.0000
*	6	11.00	1.0000
*	7	10.80	1.0000
*	8	10.60	1.0000
*	9	10.40	1.0000
*	10	10.20	1.0000
	11	10.00	1.2470
	12	9.80	1.2392
	13	9.60	1.2311
	14	9.40	1.2256
	15	9.20	1.2210
	16	9.00	1.2152
	17	8.80	1.2107
	18	8.60	1.2139
	19	8.40	1.2383
	20	8.20	1.2593
	21	8.00	1.2764
	22	7.80	1.2910
	23	7.60	1.3022
	24	7.40	1.3103
	25	7.20	1.3153
	26	7.00	1.3178
	27	6.80	1.3184
	28	6.60	1.3166
	29	6.40	1.3124
	30	6.20	1.3059
	31	6.00	1.2976
	32	5.80	1.2861
	33	5.60	1.2753
	34	5.40	1.2725
	35	5.20	1.2698
	36	5.00	1.2650
	37	4.80	1.2582
	38	4.60	1.2493
	39	4.40	1.2382
	40	4.20	1.2250
	41	4.00	1.2100
	42	3.80	1.1988
	43	3.60	1.1875
	44	3.40	1.1745
	45	3.20	1.1598
	46	3.00	1.1596
	47	2.80	1.1724
	48	2.60	1.1870
	49	2.40	1.2009
	50	2.20	1.2149
	51	2.00	1.2288
*	52	1.80	1.0000
*	53	1.60	1.0000
*	54	1.40	1.0000
*	55	1.20	1.0000
*	56	1.00	1.0000
*	57	0.80	1.0000
*	58	0.60	1.0000
*	59	0.40	1.0000
*	60	0.20	1.0000
*	61	0.00	1.0000

* Top and Bottom 15% Excluded per Technical Specification B3.2.1.

This figure is referred to by Technical Specification B3.2.1.

Figure 8
Reactor Core Safety Limits