ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-91-08)

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DEFINITIONS

SECTION

1.0 DEFINITIONS

DEFINITIONS

SECTION		PAGE
1.0 DEFINITIONS (Continued)		

Amendment No. 71
May 18, 1988

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions. R145

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

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

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

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMIT REPORT

1.10 (INSERT ATTACHMENT B)

ATTACHMENT B

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification ~~[6.9.X.X]~~. Unit operation within these operating limits is addressed in individual specifications.

6.9.1.14

DEFINITIONS

DOSE EQUIVALENT I-131

1.11 ~~1.10~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." | R

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 ~~1.11~~ $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant. | R

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.13 ~~1.12~~ The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. | R

Insert

FREQUENCY NOTATION

1.14 ~~1.13~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2. | R

GASEOUS RADWASTE TREATMENT SYSTEM

1.15 ~~1.14~~ A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment. | R

IDENTIFIED LEAKAGE

1.16 ~~1.15~~ IDENTIFIED LEAKAGE shall be: | R

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

1.17 ~~1.16~~ MEMBERS OF THE PUBLIC shall include all individuals who are not occupationally associated with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category does not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials. R75

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 ~~1.17~~ The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.5 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8. R1

OPERABLE - OPERABILITY

1.19 ~~1.18~~ A system, subsystem, train, or component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). R75

OPERATIONAL MODE - MODE

1.20 ~~1.19~~ An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1. R75

PHYSICS TESTS

1.21 ~~1.20~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission. R75

PRESSURE BOUNDARY LEAKAGE

1.22 ~~1.21~~ PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 ~~1.22~~ The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71; State regulations; and other requirements governing the disposal of solid radioactive wastes. R R

PURGE - PURGING

1.24 ~~1.23~~ PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 ~~1.24~~ QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER (RTP) R145

1.26 ~~1.25~~ RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 ~~1.26~~ The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.28 ~~1.27~~ A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

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SHIELD BUILDING INTEGRITY

1.29 ~~1.28~~ SHIELD BUILDING INTEGRITY shall exist when: R

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

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

SHUTDOWN MARGIN

1.30 ~~1.29~~ SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn. R75

SITE BOUNDARY

1.31 ~~1.30~~ The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see Figure 5.1-1). R75

SOLIDIFICATION

1.32 ~~1.31~~ Deleted R

SOURCE CHECK

1.33 ~~1.32~~ Deleted R

STAGGERED TEST BASIS

1.34 ~~1.33~~ A STAGGERED TEST BASIS shall consist of: R75

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 ~~1.34~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. R75

UNIDENTIFIED LEAKAGE

1.36 ~~1.35~~ UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE. R75

UNRESTRICTED AREA

1.37 ~~1.36~~ An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commerical, institutional, and/or recreational purposes. R75

VENTILATION EXHAUST TREATMENT SYSTEM

1.38 ~~1.37~~ A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components. R75

VENTING

1.39 ~~1.38~~ VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process. R75

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be: WITHIN THE LIMITS SPECIFIED IN THE COLR. THE MAXIMUM UPPER LIMIT SHALL BE LESS THAN 0 DELTA K/K/°F

~~a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.~~

~~b. Less negative than -4.0×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

APPLICABILITY: BEGINNING OF CYCLE LIFE (BOL) LIMIT ~~Specification 3.1.1.3.a~~ - MODES 1 and 2* only#
END OF CYCLE LIFE (EOL) LIMIT ~~Specification 3.1.1.3.b~~ - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit SPECIFIED IN THE COLR ~~of 3.1.1.3.a above~~ operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than ~~0 delta k/k/°F~~ THE BOL LIMIT SPECIFIED IN THE COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.6.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. R40

b. With the MTC more negative than the EOL limit SPECIFIED IN THE COLR ~~of 3.1.1.3.b above~~, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

SPECIFIED IN THE COLR

a. The MTC shall be measured and compared to the BOL limit ~~of Specification 3.1.1.3.a, above~~, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

THE 300 PPM SURVEILLANCE LIMIT SPECIFIED IN THE COLR

b. The MTC shall be measured at any THERMAL POWER and compared to ~~-3.1×10^{-4} delta k/k/°F~~ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates that MTC is more negative than ~~-3.1×10^{-4} delta k/k/°F~~, the MTC shall be remeasured, and compared to the EOL MTC limit ~~of Specification 3.1.1.3.b~~ SPECIFIED IN THE COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

THE 300 PPM SURVEILLANCE LIMIT SPECIFIED IN THE COLR

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements,

   2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limit of ~~Figure 3.1-1;~~ Specification 3.1.3.6. R118
   T~~t~~he THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

LIMITED IN PHYSICAL INSERTION AS SPECIFIED IN THE COLR R112

3.1.3.5 All shutdown rods shall be ~~fully withdrawn.**~~

APPLICABILITY: MODES 1* and 2*#

ACTION:

INSERTED BEYOND THE INSERTION LIMIT SPECIFIED IN THE COLR

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

RESTORE THE ROD TO WITHIN THE INSERTION LIMIT SPECIFIED IN THE COLR, OR

a. ~~fully withdraw the rod, or~~

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

SURVEILLANCE REQUIREMENTS

WITHIN THE INSERTION LIMIT SPECIFIED IN THE COLR

4.1.3.5 Each shutdown rod shall be determined to be ~~fully withdrawn:~~

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

~~**Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.~~ R112

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as ~~shown in Figure 3.1-1.~~ SPECIFIED IN THE COLR. R45

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the ~~above~~ insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the ~~above figure, or~~ INSERTION LIMITS SPECIFIED IN THE COLR, OR R118

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

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*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

DELETE

R112

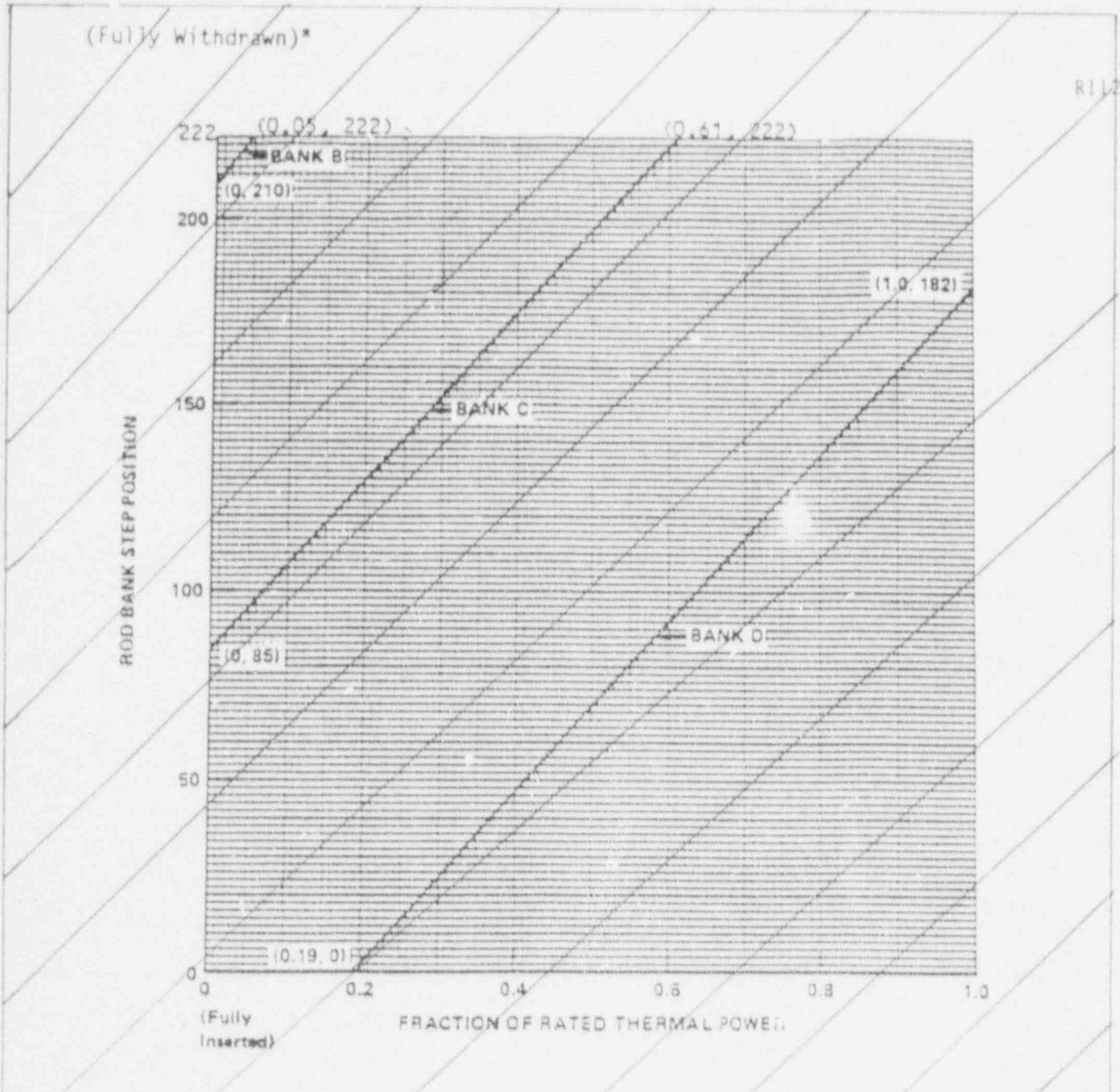

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

*See page 3/4 1-23.

DELETE

REACTIVITY CONTROL SYSTEMS

FIGURE 3.1-1 NOTATION

Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of $\geq$ 222 and $\leq$ 231 steps withdrawn, inclusive.

There are no rod insertion limits when the shutdown and control banks are at a position within the interval $\geq$ 222 and $\leq$ 231 steps withdrawn, inclusive. The fully withdrawn position shall be specified in a reload safety evaluation for each cycle of operation and, once specified, shall not be changed unless such a change is specifically evaluated. R112

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the ~~allowed operational space defined by Figure 3.2-1.~~ LIMITS SPECIFIED IN THE COLR. R2

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ~~Figure 3.2-1~~ limits SPECIFIED IN THE COLR;

   1. Either restore the indicated AFD to within the ~~Figure 3.2-1~~ limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the ~~Figure 3.2-1~~ limits SPECIFIED IN THE COLR.

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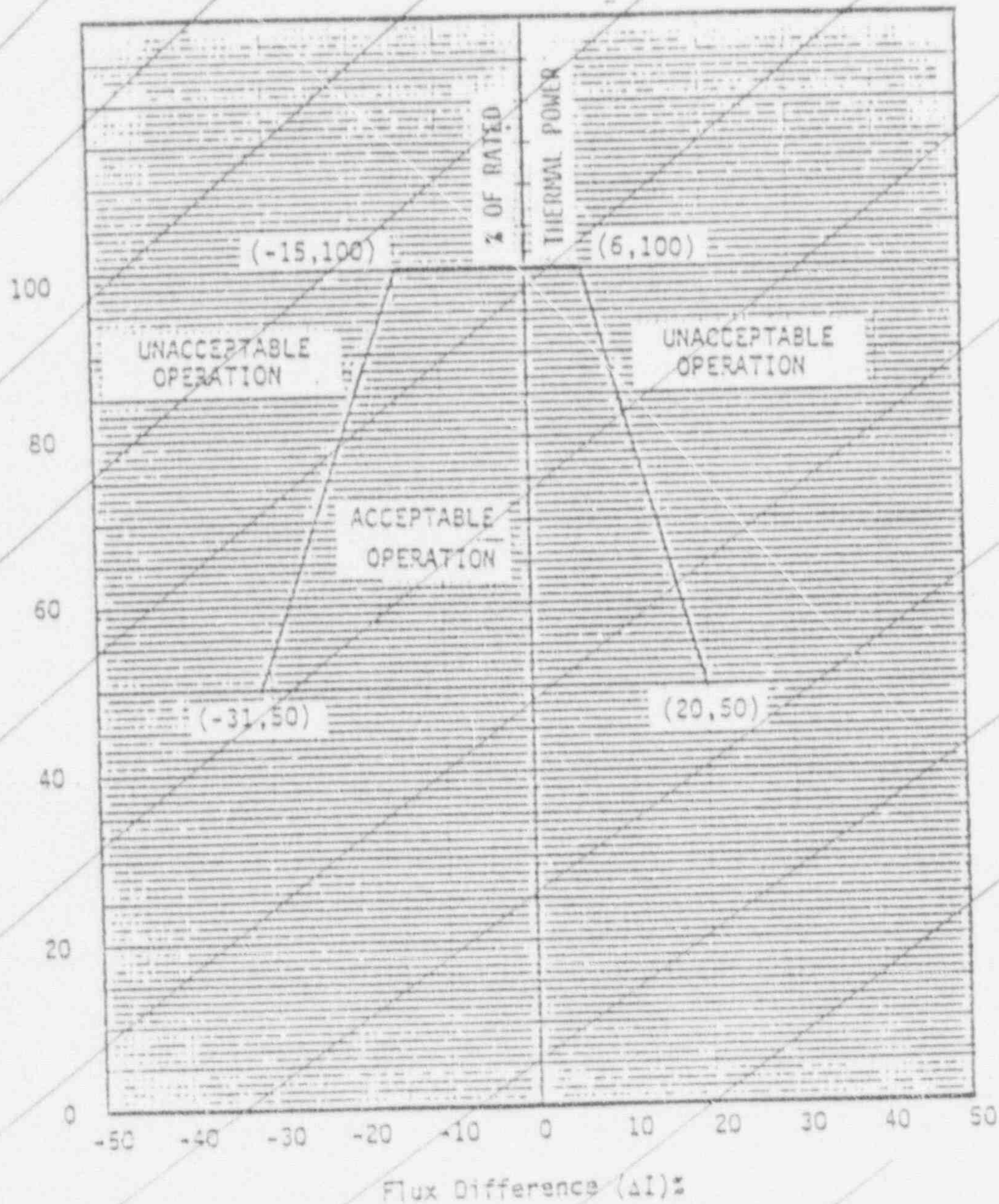

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \le \cancel{\frac{[2.32]}{P}\,[K(Z)]}$ for P > 0.5 $\frac{[F_Q^{RTP}][K(Z)]}{P}$ R144

$F_Q(Z) \le \cancel{\frac{[2.32]}{0.5}\,[K(Z)]}$ for P $\le$ 0.5 $\frac{[F_Q^{RTP}][K(Z)]}{0.5}$ R144

WHERE F_Q^{RTP} = THE F_Q LIMIT AT RATED THERMAL POWER (RTP) SPECIFIED IN THE COLR,

~~where~~ P = $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, AND

K(Z) = THE NORMALIZED $F_Q(Z)$ AS A FUNCTION OF ~~and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.~~ CORE HEIGHT SPECIFIED IN THE COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

R144

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_{Q(z)}$ component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{\cancel{2.32}\ F_Q^{RTP} \times K(z)}{P \times W(z)} \quad \text{for } P > 0.5$$ R144

$$F_Q^M(z) \leq \frac{\cancel{2.32}\ F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \quad \text{for } P \leq 0.5$$ R144

K(Z) IS THE NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT,

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, ~~F_Q limit~~ F_Q^{RTP} is the F_Q limit, ~~K(z) is given in Figure 3.2-2~~, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. ~~This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.14.~~ F_Q^{RTP}, K(Z), AND W(Z) ARE SPECIFIED IN THE COLR AS PER SPECIFICATION 6.9.1.14.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 effective full power days, whichever occurs first.

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*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

R144

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SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right]$$

has increased since the previous determinatin of $F_Q{}^M(z)$ either of the following actions shall be taken:

1. $F_Q{}^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.2.c, or

2. $F_Q{}^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right] \text{ is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.32}\ F_Q^{RTP}}{P} \times K(z)}\right]\right) -1\right\} \times 100 \qquad \text{for } P \geq 0.5$$ R144

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.32}\ F_Q^{RTP}}{0.5} \times K(z)}\right]\right) -1\right\} \times 100 \qquad \text{for } P < 0.5$$ R144

2. Either of the following actions shall be taken:

   a. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of ~~Figure 3.2-1~~ SPECIFICATION 3 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

R144

DELETE

Figure 3.2-2

K(Z) AS A FUNCTION OF CORE HEIGHT

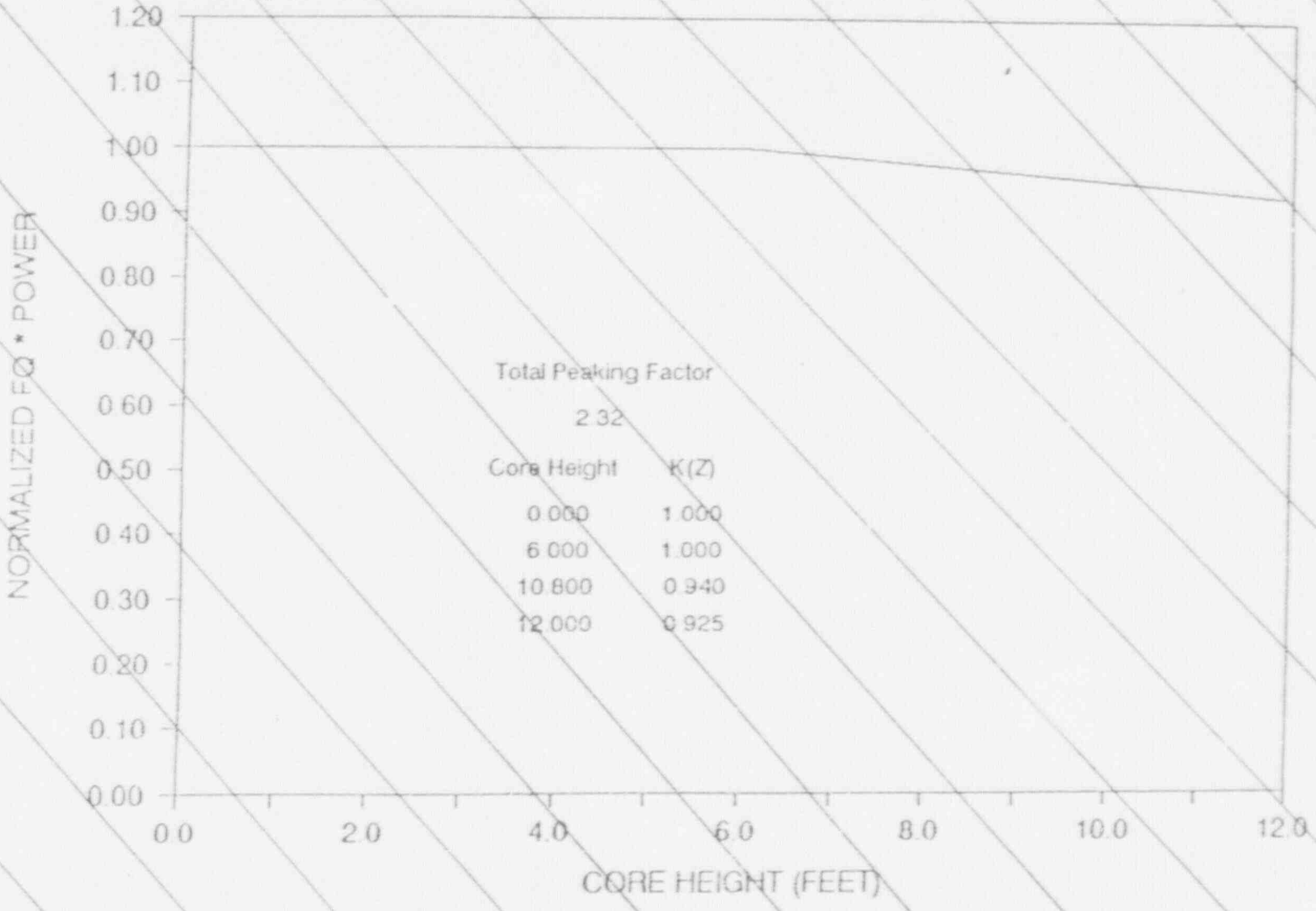

Figure 3.2-2
K(Z) - Normalized Fq(Z) as a Function of Core Height

R144

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3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The Nuclear Enthalpy Hot Channel Factor, $F^N_{\Delta H}$, shall be limited by the following relationship: R14

~~Where:~~

~~a.~~ $F^N_{\Delta H} \le \overset{F^{RTP}_{\Delta H}}{\cancel{1.55}}\,[1.0 + \overset{PF_{\Delta H}}{\cancel{0.3}}\,(1.0-P)]$

WHERE ~~b.~~ $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F^{RTP}_{\Delta H}$ = THE $F^N_{\Delta H}$ LIMIT AT RATED THERMAL POWER (RTP) SPECIFIED IN THE COLR, AND

$PF_{\Delta H}$ = THE POWER FACTOR MULTIPLIER FOR $F^N_{\Delta H}$ SPECIFIED IN THE COLR.

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\le$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and R14

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting END OF CYCLE LIFE (EOL) MTC value. ~~-4.0×10^{-4} delta k/k/°F.~~ The 300 PPM SURVEILLANCE LIMIT MTC value ~~of -3.1×10^{-4} delta k/k/°F~~ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL ~~MDC value -4.0×10^{-4} delta k/k/°F.~~ MTC VALUE.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°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 P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in a OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 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) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

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. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires

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 calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

THE F_Q LIMIT SPECIFIED IN THE COLR

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of ~~2.32~~ times the normalized axial peaking factor is not exceeded R144
during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS R142

The limits on the heat flux hot channel factor and the nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

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Each of these hot channel factors is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 13 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The $F^N_{\Delta H}$ limit as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. The 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance. R142

When an $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq$ $F^{RTP}_{\Delta H}$ ~~1.55~~/1.08. The 8% allowance is based on the following considerations.

a. abnormal perturbatios in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

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Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty.

The applicable value ~~on~~ OF rod bow penalty is referenced in the FSAR. R142

Margin in excess of the rod bow penalty is available for plant design flexibility.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor, W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function ~~for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.14.~~ IS SPECIFIED IN THE COLR.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio 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 two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

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 initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R142

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.

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MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or 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. R76

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PORC.

CORE OPERATING LIMITS REPORT
~~RADIAL PEAKING FACTOR LIMIT REPORT~~

DELETE AND REPLACE WITH INSERT C

~~6.9.1.14 The W(z) function for normal operation shall be provided at least 60 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.~~ R76

~~Any information needed to suport W(z) will be by request from the NRC and need not be included in this report.~~

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4. R76

6.9.2.2 Diesel Generator Reliability Improvement Program

As a minimum the Reliability Improvement Program report for NRC audit, required by LCO 3.8.1.1, Table 4.8-1, shall include: R56

(a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

(b) analysis of failures and determination of root causes of failures

(c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

(d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

(e) the schedule for implementation of each action from d) above

(f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

INSERT C

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference limits for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, K(Z), and W(Z) for Specification 3/4.2.2, and

6. Nuclea· Enthalpy Hot Channel Factor and Power Factor Multiplier for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", JUNE 1983 (W Proprietary).
   (Methodology for Specification 3.2.1 - Axial Flux difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

3. WCAP-10266-P-A Rev.2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

CORE OPERATING LIMITS REPORT (continued)

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

DEFINITIONS

DEFINITIONS

R63

Amendment No. 63
May 18, 1988

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be: R63

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions. R132

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when: R63

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

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

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3, R117

d. The containment leakage rates are within the limits of Specification 3.6.[illegible] and

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

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals. R63

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position. R63

CORE OPERATING LIMITS REPORT
1.10 (INSERT ATTACHMENT B)

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ATTACHMENT B

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.x.x. Unit operation within these operating limits is addressed in individual specifications.

6.9.1.14

DOSE EQUIVALENT I-131

1.11 ~~1.10~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." R6

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 ~~1.11~~ $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant. R6

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.13 ~~1.12~~ The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. R6

FREQUENCY NOTATION

1.14 ~~1.13~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2. R6

GASEOUS RADWASTE TREATMENT SYSTEM

1.15 ~~1.14~~ A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment. R6

IDENTIFIED LEAKAGE

1.16 ~~1.15~~ IDENTIFIED LEAKAGE shall be: R63

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBERS OF THE PUBLIC

1.17 ~~1.16~~ MEMBERS OF THE PUBLIC shall include all individuals who are not occupationally associated with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category does not include non-employees such a vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials. R63

OFFSITE DOSE CALCULATION MANUAL

1.18 ~~1.17~~ The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.5 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8. R134 R134

OPERABLE - OPERABILITY

1.19 ~~1.18~~ A system, subsystem, train, or component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). R63

OPERATIONAL MODE - MODE

1.20 ~~1.19~~ An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1. R63

PHYSICS TESTS

1.21 ~~1.20~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission. R63

PRESSURE BOUNDARY LEAKAGE

1.22 ~~1.21~~ PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall. R63

PROCESS CONTROL PROGRAM (PCP)

1.23 ~~1.22~~ The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71; State regulations; and other requirements governing the disposal of solid radioactive wastes. R134

PURGE - PURGING

1.24 ~~1.23~~ PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. R63

QUADRANT POWER TILT RATIO

1.25 ~~1.24~~ QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average. R63

RATED THERMAL POWER (RTP)

1.26 ~~1.25~~ RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt. R63

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 ~~1.26~~ The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. R63

REPORTABLE EVENT

1.28 ~~1.27~~ A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50. R63

SHIELD BUILDING INTEGRITY

1.29 ~~1.28~~ SHIELD BUILDING INTEGRITY shall exist when: R63

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

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

SHUTDOWN MARGIN

1.30 ~~1.29~~ SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn. R63

SITE BOUNDARY

1.31 ~~1.30~~ The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1). R63

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SOLIDIFICATION

1.32 ~~1.31~~ Deleted. R134

SOURCE CHECK

1.33 ~~1.32~~ Deleted. R134

STAGGERED TEST BASIS

1.34 ~~1.33~~ A STAGGERED TEST BASIS shall consist of: R63

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 ~~1.34~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. R63

UNIDENTIFIED LEAKAGE

1.36 ~~1.35~~ UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE. R63

UNRESTRICTED AREA

1.37 ~~1.36~~ An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional, and/or recreational purposes. R63

VENTILATION EXHAUST TREATMENT SYSTEM

1.38 ~~1.37~~ A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components. R6

VENTING

1.39 ~~1.38~~ VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process. R6

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be: WITHIN THE LIMITS SPECIFIED IN THE COLR. THE MAXIMUM UPPER LIMIT SHALL BE LESS THAN 0 DELTA K/K/°F

~~a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.~~

~~b. Less negative than -4 x 10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

APPLICABILITY: BEGINNING OF CYCLE LIFE (BOL) LIMIT ~~Specification 3.1.1.3.a~~ - MODES 1 and 2* only#
END OF CYCLE LIFE (EOL) LIMIT ~~Specification 3.1.1.3.b~~ - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit SPECIFIED IN THE COLR ~~of 3.1.1.3.a above,~~ operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than ~~0 delta k/k/°F~~ THE BOL LIMIT SPECIFIED IN THE COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.6.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. R28

b. With the MTC more negative than the EOL limit SPECIFIED IN THE COLR ~~of 3.1.1.3.b above,~~ be in HOT SHUTDOWN within 12 hours.

---

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit ~~of Specification 3.1.1.3.a, above~~ SPECIFIED IN THE CO, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to ~~-3.1×10^{-4} delta k/k/°F~~ THE 300 PPM SURVEILLANCE LIMIT SPECIFIED IN THE COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-3.1×10^{-4} delta k/k/°F~~ THE 300 PPM SURVEILLANCE LIMIT SPECIFIED IN THE COLR, the MTC shall be remeasured, and compared to the EOL MTC limit ~~of specification 3.1.1.3.b~~ SPECIFIED IN THE COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: Modes 1* and 2*.

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod while maintaining the rod sequence and insertion limit of ~~Figure 3.1-1~~ SPECIFICATION 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or R104

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

*See Special Test Exceptions 3.10.2 and 3.10.3.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be ~~fully withdrawn.**~~ LIMITED IN PHYSICAL INSERTION AS SPECIFIED IN THE COLR. R9

APPLICABILITY: Modes 1* and 2*#.

ACTION:

With a maximum of one shutdown rod ~~not fully withdrawn~~ INSERTED BEYOND THE INSERTION LIMIT SPECIFIED IN THE COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. ~~Fully withdraw the rod, or~~ RESTORE THE ROD TO WITHIN THE INSERTION LIMIT SPECIFIED IN THE COLR, OR

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be ~~fully withdrawn~~ WITHIN THE INSERTION LIMIT SPECIFIED IN THE COLR:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0

~~**Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.~~ R98

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as ~~shown in Figure 3.1-1.~~ SPECIFIED IN THE COLR. R33

APPLICABILITY: Modes 1* and 2*#.

ACTION:
With the control banks inserted beyond the ~~above~~ insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the ~~above figure, or~~ INSERTION LIMITS SPECIFIED IN THE COLR, OR R104

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0.

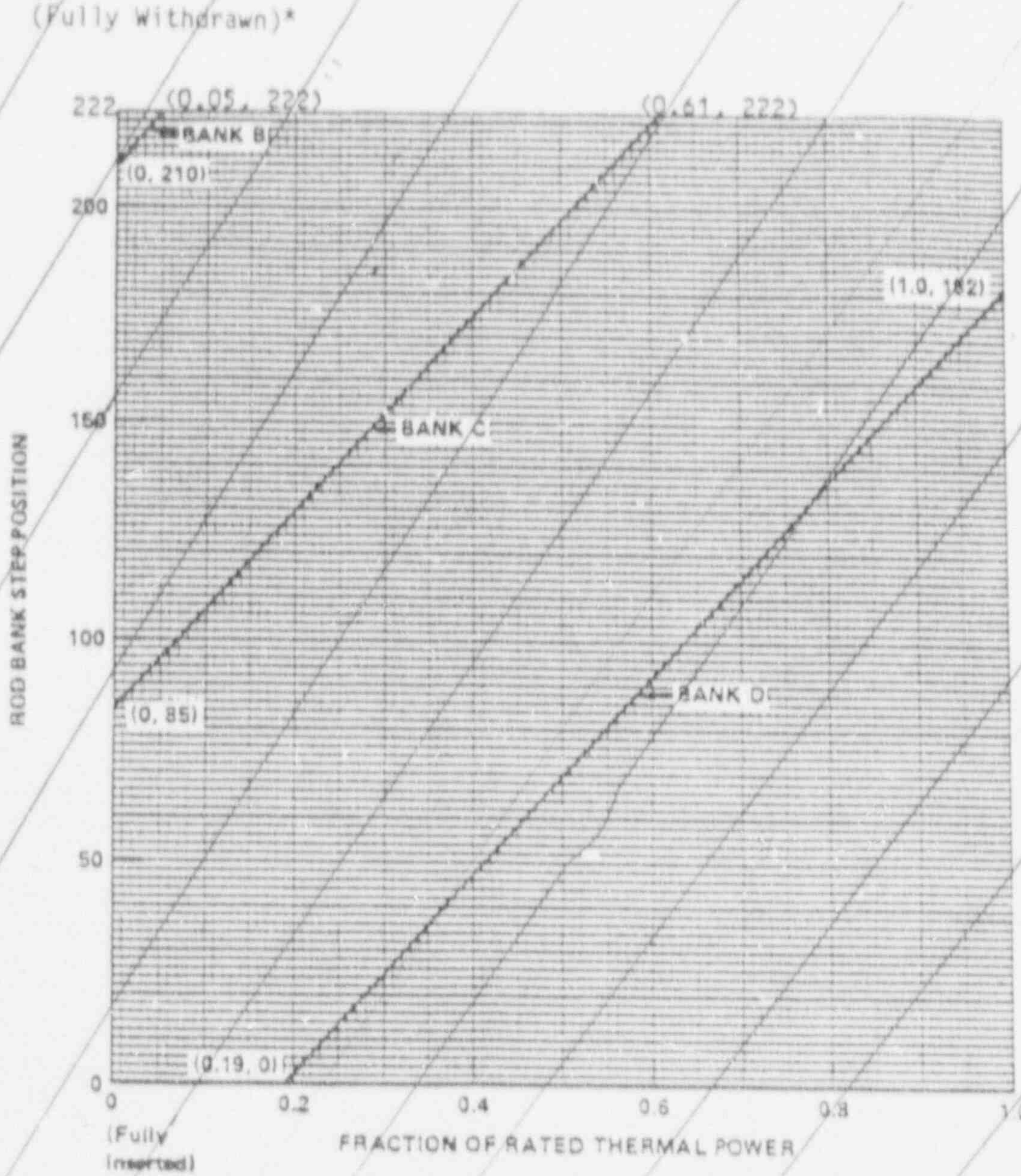

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

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*See page 3/4 1-23

R98

~~FIGURE 3.1-1 NOTATION~~

~~Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.~~ R98

~~There are no rod insertion limits when the shutdown and control banks are at a position within the interval > 222 and < 231 steps withdrawn, inclusive. The fully withdrawn position shall be specified in a reload safety evaluation for each cycle of operation and, once specified, shall not be changed unless such a change is specifically evaluated.~~

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the ~~allowed operational space defined by Figure 3.2-1.~~ LIMITS SPECIFIED IN THE COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ~~Figure 3.2-1~~ limits SPECIFIED IN THE COLR;

1. Either restore the indicated AFD to within the ~~Figure 3.2-1~~ limits within 15 minutes, or R 21

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the ~~Figure 3.2-1~~ limits SPECIFIED IN THE COLR.

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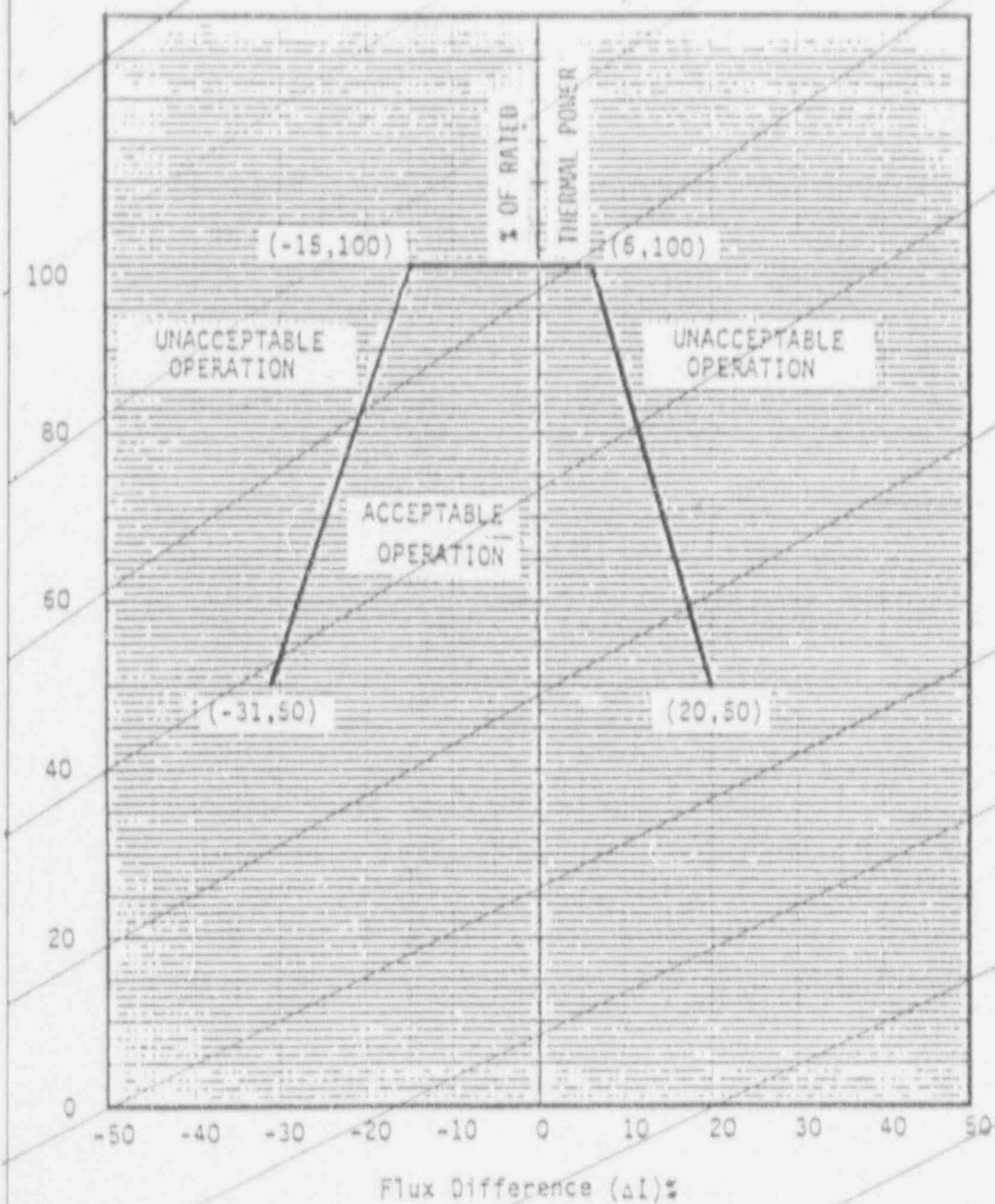

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq \cancel{\frac{[2.32]\,[K(Z)]}{P}}$ for $P > 0.5$ $\frac{[F_Q^{RTP}][K(Z)]}{P}$ R131

$F_Q(Z) \leq \cancel{\frac{[2.32]\,[K(Z)]}{0.5}}$ for $P \leq 0.5$ $\frac{[F_Q^{RTP}][K(Z)]}{0.5}$ R131

WHERE F_Q^{RTP} = THE F_Q LIMIT AT RATED THERMAL POWER (RTP) SPECIFIED IN THE COLR,

~~where~~ $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, AND

$K(Z)$ = THE NORMALIZED $F_Q(Z)$ AS A FUNCTION OF ~~and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.~~ CORE HEIGHT SPECIFIED IN THE COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit: R21

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

R131

OCT 2 [illegible]

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. R21

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{\boxed{\cancel{2.32}}^{\,F_Q^{RTP}}}{P \times W(z)} \times K(z) \quad \text{for } P > 0.5$$ R13

$$F_Q^M(z) \leq \frac{\boxed{\cancel{2.32}}^{\,F_Q^{RTP}}}{W(z) \times 0.5} \times K(z) \quad \text{for } P \leq 0.5$$ R13

K(z) IS THE NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

F_Q^{RTP}

where $F_Q^M(z)$ is measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, $\boxed{\cancel{F_Q \text{ limit}}}$ is the F_Q limit, ~~K(z) is given in Figure 3.2-2~~, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. ~~This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.14.~~ F_Q^{RTP}, K(z), AND W(z) ARE SPECIFIED IN THE COLR AS PER SPECIFICATION 6.9.1.14.

d. Measuring $F_Q^M(z)$ according to the following schedule: R21

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 effective full power days, whichever occurs first.

---

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

R13

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)} \right]$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. $F_Q^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that R21

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)} \right] \quad \text{is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.22}\; F_Q^{RTP}}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P \geq 0.5$$ R131

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.22}\; F_Q^{RTP}}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P < 0.5$$ R131

2. Either of the following actions shall be taken:

   a. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of SPECIFICATION 3.2.1 ~~Figure 3.2-1~~ are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or R21

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above. R131

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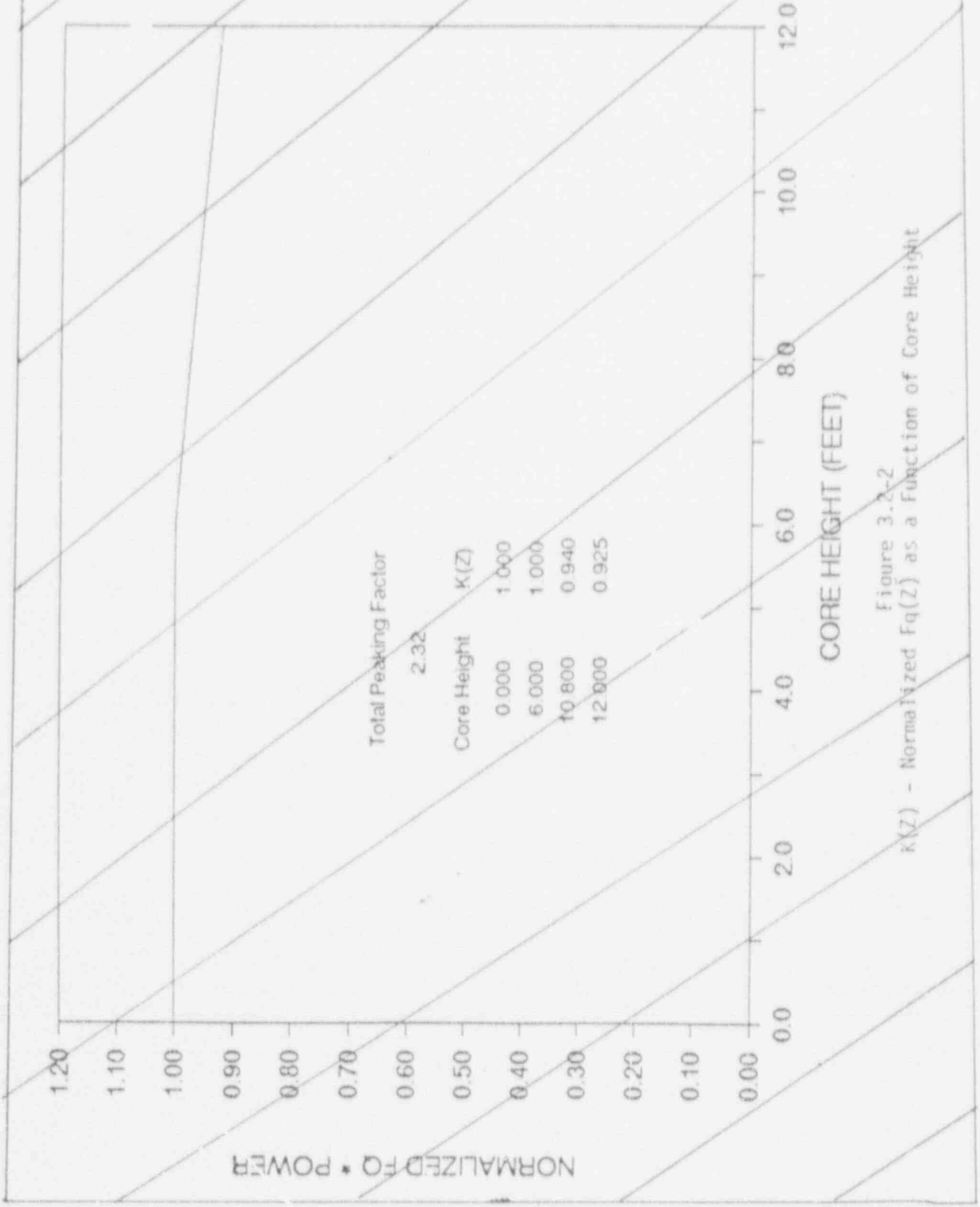
Figure 3.2-2

K(Z) AS A FUNCTION OF CORE HEIGHT



R1

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3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The Nuclear Enthalpy Hot Channel Factor, $F^N_{\Delta H}$, shall be limited by the following relationship:

~~where:~~

~~a.~~ $F^N_{\Delta H} \le \cancel{1.55}\ F^{RTP}_{\Delta H} [1.0 + \cancel{0.3}\ PF_{\Delta H} (1.0 - P)]$

WHERE ~~b.~~ $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F^{RTP}_{\Delta H}$ = THE $F^N_{\Delta H}$ LIMIT AT RATED THERMAL POWER (RTP) SPECIFIED IN THE COLR, AND

$PF_{\Delta H}$ = THE POWER FACTOR MULTIPLIER FOR $F^N_{\Delta H}$ SPECIFIED IN THE COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\le$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting END OF CYCLE LIFE (EOL) MTC value ~~-4.0×10^{-4} delta k/k/°F~~. The 300 PPM SURVEILLANCE LIMIT MTC value ~~of -3.1×10^{-4} delta k/k/°F~~ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value. ~~of -4.0×10^{-4} k/k/°F.~~

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within it analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in a OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 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) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection system are provided to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to

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 calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows: R21

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

THE F_Q LIMIT SPECIFIED IN THE COLR

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.32~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. R13

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The compuer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS R130

The limits on heat flux hot channel factor and nuclear enthalpy hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. R21

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BASES

Each of these hot channel factors is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 13 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The $F^N_{\Delta H}$ limit as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. The 5% is the appropriate allowance for a full core map taken with the in-core detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the in-core detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq$ ~~1.55~~ $F^{RTP}_{\Delta H}$/1.08. The 8% allowance is based on the following considerations.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty. R130

The applicable value of rod bow penalty is referenced in the FSAR.

Margin in excess of the rod bow penalty is available for plant design flexibility.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function ~~for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.14.~~ IS SPECIFIED IN THE COLR. R21

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio 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 two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

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 initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R130

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.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or 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. R64

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PORC.

CORE OPERATING LIMITS REPORT
~~RADIAL PEAKING FACTOR LIMIT REPORT~~

DELETE AND REPLACE WITH INSERT C

~~6.9.1.14 The W(z) function for normal operation shall be provided at least 60 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.~~ R64

~~Any information needed to suport W(z) will be by request from the NRC and need not be included in this report.~~

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4. R64

6.9.2.2 Diesel Generator Reliability Improvement Program

As a minimum the Reliability Improvement Program report for NRC audit, required by LCO 3.8.1.1, Table 4.8-1, shall include: R44

(a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

(b) analysis of failures and determination of root causes of failures

(c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

(d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

(e) the schedule for implementation of each action from d) above

(f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

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INSERT C

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference limits for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, K(Z), and W(Z) for Specification 3/4.2.2, and

6. Nuclear Enthalpy Hot Channel Factor and Power Factor Multiplier for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", JUNE 1983 (W Proprietary).
   (Methodology for Specification 3.2.1 - Axial Flux difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

3. WCAP-10266-P-A Rev.2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

CORE OPERATING LIMITS REPORT (continued)

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-91-08)

DESCRIPTION AND JUSTIFICATION FOR
CREATING THE CORE OPERATING LIMITS REPORT

Description of Change

The proposed technical specification (TS) changes concern the relocation of several cycle-specific core operating limits for Sequoyah Nuclear Plant from the TSs to the Core Operating Limits Report (COLR). The impacted TSs will be amended to note that the limit has been relocated to the COLR, and a COLR paragraph will be added to the Administrative Controls Section to replace the Radial Peaking Factor Report. The COLR will be required to be submitted to NRC within 30 days after cycle start-up (Mode 2) or upon issuance of any midcycle revision to allow for continued trending of the cycle-specific parameters.

The proposed changes will reference the COLR for specific parameters and will ensure that cycle-specific parameters are maintained within the limits of the COLR. The cycle-specific parameter limits proposed for relocation to the COLR as part of this TS change include:

1. Moderator Temperature Coefficient
2. Shutdown Rod Insertion Limits
3. Control Rod Insertion Limits
4. Axial Flux Difference
5. Heat Flux Hot Channel Factor
6. Nuclear Enthalpy Hot Channel Factor

Note: A listing of each revised TS is provided in Attachment A.

Reason for Change

In Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," NRC encouraged licensees to remove certain cycle-specific parameters from the TS provided that these parameters are determined by NRC-approved methodologies. Presently, the parameters described above can change from cycle to cycle and would require a TS revision each cycle. By removing these certain parameters from the TS and creating a separate report (COLR) that contains these specific values, TS revisions are no longer required. The COLR will replace the Radial Peaking Factor Limit Report required by TS 6.9.1.14. This change will result in a resource savings for both NRC and SQN.

Justification for Change

The current TS method of controlling the above reactor physics parameters to ensure conformance to 10 CFR 50.36 (which requires the lowest functional levels acceptable for continued safe operation) is to specify the values determined to be within the acceptance criteria using an NRC-approved calculation methodology. The methodologies for calculating these parameters have been approved by NRC.

The removal of cycle-dependent variables from the TS has no impact upon plant operation or safety. No safety-related equipment, safety function, or plant operations will be altered as a result of this proposed change. Since applicable Updated Final Safety Analysis Report limits will be maintained, and the TSs will continue to require operation within the core operating limits calculated by the approved methodologies, this proposed change is administrative in nature and does not affect the purpose of the TS involved. Appropriate actions to be taken if the limits are violated will remain in the TSs.

This proposed change will control the cycle-specific parameters within the acceptance criteria and ensure conformance to 10 CFR 50.36 by using the approved methodology instead of specifying TS values. The COLR will document the specific parameter limits resulting from NRC-approved calculations, including midcycle or other revisions to parameter values. Therefore, the proposed change is in conformance with the requirements of 10 CFR 50.36.

Any changes to the COLR will be made in accordance with the requirements of 10 CFR 50.59, with a copy of the revised COLR sent to NRC as required in Section 6.9.1.14 of the TSs. From cycle to cycle, the COLR will be revised such that the appropriate core operating limits for the applicable unit and cycle will apply. Therefore, the need to continually revise TSs for every reload is eliminated.

Environmental Impact Evaluation

The proposed change request does not involve an unreviewed environmental question because operation of SQN Units 1 and 2 in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the Staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or decisions of the Atomic Safety and Licensing Board.

2. Result in a significant change in effluents or power levels.

3. Result in matters not previously reviewed in the licensing basis for SQN that may have a significant environmental impact.

ATTACHMENT A

TECHNICAL SPECIFICATIONS AFFECTED BY PROPOSED AMENDMENT AND BRIEF DESCRIPTION OF CHANGE

PAGE*	TECHNICAL SPECIFICATION	CHANGE DESCRIPTION
3/4 1-4	3.1.1.3 Moderator Temperature Coefficient (MTC)	Relocates MTC limits to the Core Operating Limit Report (COLR) and corrects references.
3/4 1-5	4.1.1.3 MTC Surveillance Requirements	Relocates MTC limits to the COLR and corrects references.
B3/4 1-2	3/4.1.1.3 Moderator Temperature Coefficient Bases	Change deletes specific MTC values.
3/4 1-14	3.1.3.1.c.2 Movable Control Assemblies-Group Height	Change replaces Figure 3.1-1 references with Specification 3.1.3.6.
3/4 1-20	3.1.3.5 Shutdown Rod Insertion Limit	Change clarifies that the fully withdrawn position for shutdown banks is specified in the COLR.
3/4 1-21	3.1.3.6 Control Rod Insertion Limit	Change removes reference to Figure 3.1-1 and relocates to COLR.
3/4 1-22	Figure 3.1-1 Rod Bank Insertion Limits Versus Thermal Power Four Loop Operation	Relocates figure to COLR.
3/4 1-23	Figure 3.1-1 Notation	Revised and relocated with figure to COLR.
3/4 2-1	3.2.1 Axial Flux Difference	Replaces references to Figure 3.2-1 that are moved to the COLR.
3/4 2-4	Figure 3.2-1 Axial Flux Difference Limits as a Function of Rated Thermal Power	Figure moved to the COLR.
B3/4 2-1	3/4.2.1 Axial Flux Difference Bases	The F_Q limit is removed and reference to the COLR added.
B3/4 2-2	3/4.2.2 and 3/4.2.3 Heat Flux and Nuclear Enthalpy Hot Channel Factors Bases	Relocates the numerical value of $F_{\Delta H}$ to the COLR.

TECHNICAL SPECIFICATIONS AFFECTED BY PROPOSED AMENDMENT AND BRIEF DESCRIPTION OF CHANGE

PAGE*	TECHNICAL SPECIFICATION	CHANGE DESCRIPTION
3/4 2-5	3.2.2 Heat Flux Hot Channel Factor	Relocates the F_Q and K(Z) to the COLR. The numerical F_Q limit is replaced with a function F_Q^{RTP}, that is to be specified in the COLR.
3/4 2-6&7	4.2.2.2 Heat Flux Hot Channel Factor Surveillance Requirements	Same as 3.2.2 above.
3/4 2-9	Figure 3.2-2 K(Z)-Normalized $F_Q(Z)$ as a Function of Core Height	Figure moved to the COLR.
3/4 2-10	3.2.3 Nuclear Enthalpy Hot Channel Factor	Relocates the numerical value of $F_{\Delta H}^{RTP}$ and defines $PF_{\Delta H}$ in the COLR.
B3/4 2-4	3/4.2.2 and 3/4.2.3 Heat Flux and Nuclear Enthalpy Hot Channel Factor Bases	The Radial Peaking Factor Limits Report is replaced by the COLR.
6-21	6.9.1.14 Radial Peaking Factor Limit Report	Replace the Radial Peaking Factor Report with the COLR.

*Unit 1 pages listed only, Unit 2 will be similar.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-91-05)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Significant Hazards Evaluation

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The removal of cycle-specific core operating limits from the SQN TSs has no influence or impact on the probability or consequences of any accident previously evaluated. Although not in the TSs, the core operating limits will be followed in the operation of SQN. The proposed amendment does not affect the actions to be taken when or if limits are exceeded. Each accident analysis addressed in the SQN Updated Final Safety Analysis Report will be examined with respect to changes in cycle-dependent parameters, which are obtained from the use of NRC-approved reload design methodologies. This will ensure that the transient evaluation of new reloads is bounded by previously accepted analysis. This examination, which will be performed in accordance with the requirements of 10 CFR 50.59, ensures that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   Operating SQN in accordance with the proposed change will not create the possibility of a new or different kind of accident from any previously analyzed. The removal of the specific core operating limits from the TSs does not modify safety-related equipment or systems, nor does it change any safety-related setpoints used to prevent or mitigate previously analyzed accidents. The core operating limits will be defined in a separate document (COLR) from the TS and will be adhered to during plant operation. Also, the limiting condition of operation requirements remain in effect and appropriate actions will be taken if any limits are exceeded. Therefore, the proposed amendment does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The margin of safety is not affected by the removal of cycle-specific core operating limits from the TSs. The margin of safety presently provided by current TSs remains unchanged. Appropriate measures exist to control the values of these cycle-specific limits. The proposed amendment continues to require operation within the core limits as obtained from the NRC-approved reload design methodologies and appropriate actions to be taken when or if limits are violated remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC-approved documentation. In addition, each future reload will involve a 10 CFR 50.59 safety review to ensure that operation of the unit within the cycle-specific limits will not involve a significant reduction in a margin of safety.

Therefore, the proposed changes will only move the pertinent parameters from one document to another and do not impact the operation of SQN in a manner that involves a reduction in the margin of safety.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-91-08)

SAMPLE CORE OPERATING LIMITS REPORT (COLR)

SAMPLE SEQUOYAH NUCLEAR PLANT

CORE OPERATING LIMITS REPORT

REVISION A

APRIL 27, 1991

Not To Be Used For Operation. For Illustration Only

Reviewed:

_______________________________ / _______________

Reactor Engineering Supervisor Date

Approved:

_______________________________ / _______________

Technical Support Manager Date

_______________________________ / _______________

PORC Chairman Date

SAMPLE COLR FOR SEQUOYAH UNIT [ ] CYCLE [ ]

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Sequoyah Unit [ ] Cycle [ ] has been prepared in accordance with the requirements of Technical Specification (TS) 6.9.1.14.

The TSs affected by this report are listed below:

3/4.1.1.3	Moderator Temperature Coefficient
3/4.1.3.5	Shutdown Rod Insertion Limit
3/4.1.3.6	Control Rod Insertion Limits
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Nuclear Enthalpy Hot Channel Factor

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 the NRC-approved methodologies specified in TS 6.9.1.14.

2.1 Moderator Temperature Coefficient (Specification 3/4.1.1.3) [3/4.1.1.3]

2.1.1 The moderator temperature coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less positive than $-0.5\ \Delta k/k/°F$.

The EOL/ARO/RTP-MTC shall be less negative than $-4.0\times10^{-4}\ \Delta k/k/°F$.

2.1.2 The 300 ppm surveillance limit is:

The measured 300 ppm/ARO/RTP-MTC should be less negative than or equal to $-3.1\times10^{-4}\ \Delta k/k/°F$.

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

2.2 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)
[3/4.1.3.5]

2.2.1 The shutdown rods shall be withdrawn to a position as defined below:

Cycle Burnup (MWD/MTU)	Steps Withdrawn
$\leq$ 2,000	$\geq$ 226 to $\leq$ 231
> 2,000 to < 14,000	$\geq$ 222 to $\leq$ 231
$\geq$ 14,000	$\geq$ 226 to $\leq$ 231

2.3 Control Rod Insertion Limit (Specification 3/4.1.3.6)
[3/4.1.3.6]

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

2.4 Axial Flux Difference (Specification 3/4.2.1)
[3/4.2.1]

2.4.1 The axial flux difference (AFD) limits are provided in Figure 2.

2.5 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3/4.2.2)
[3/4.2.2]

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

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

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

2.5.1 $F_Q^{RTP} = 2.32$

2.5.2 K(Z) is provided in Figure 3.

2.5.3 Note that the W(Z) values required by TS SR 4.2.2.2 are provided in Figures 4 through 7.

2.6 Nuclear Enthalpy Rise Hot Channel Factor - $F^{N}_{\Delta H}$ (Specification 3/4.2.3) [3/4.2.3]

$$F^{N}_{\Delta H} \leq F^{RTP}_{\Delta H} * (1+PF_{\Delta H} * [1-P])$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

2.6.1 $F^{RTP}_{\Delta H} = 1.55$

2.6.2 $PF_{\Delta H} = 0.3$

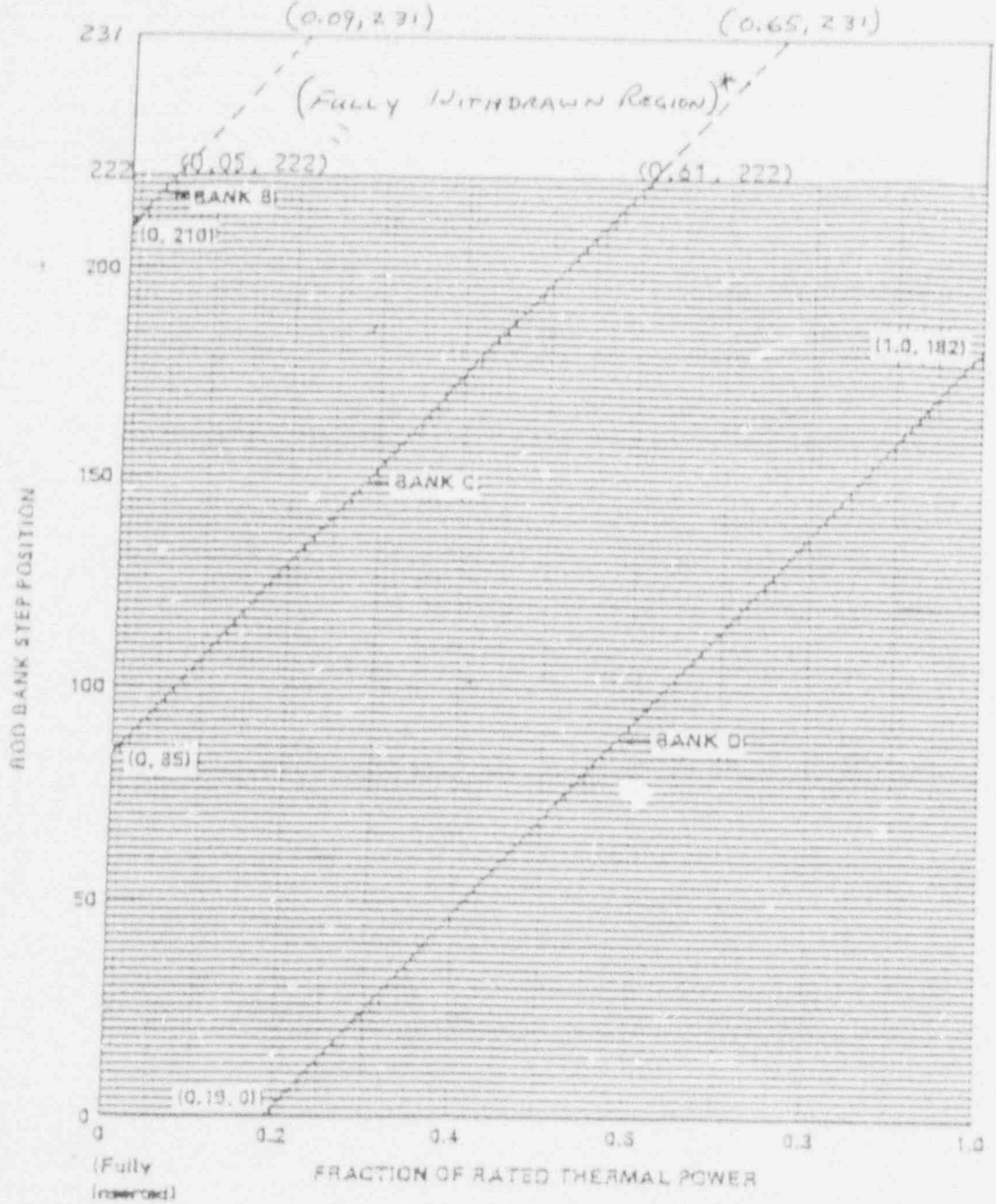


FIGURE 1 – ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

* Fully withdrawn REGION shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive.

FULLY WITHDRAWN SHALL BE THE POSITION AS DEFINED BELOW,

CYCLE BURNUP (MWD/MTU)	STEP WITHDRAWN
≤ 2000	≥ 226 TO ≤ 231
> 2000 TO < 14,000	≥ 222 TO ≤ 231
≥ 14,000	≥ 226 TO ≤ 231

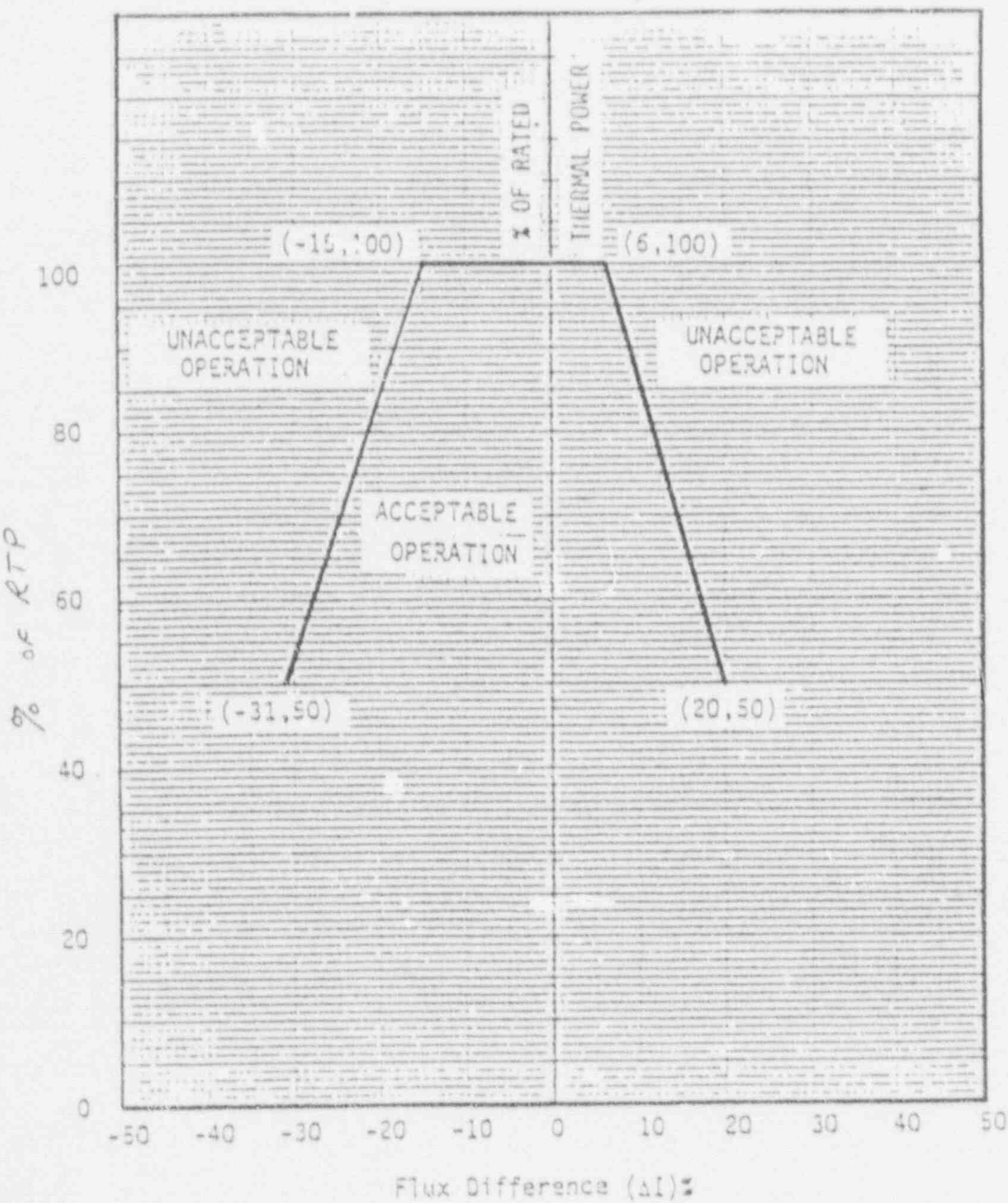


FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

K(Z) AS A FUNCTION OF CORE HEIGHT

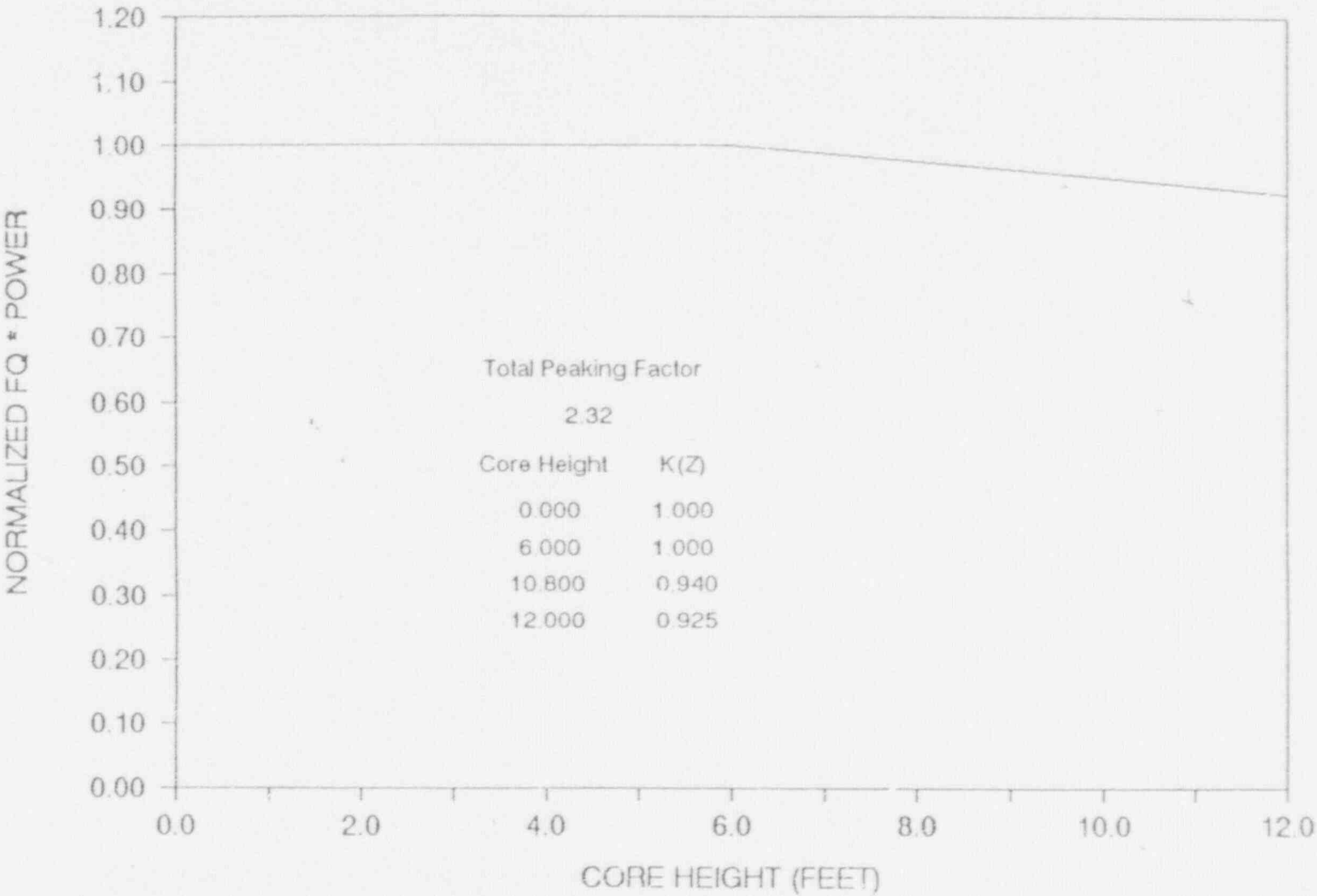
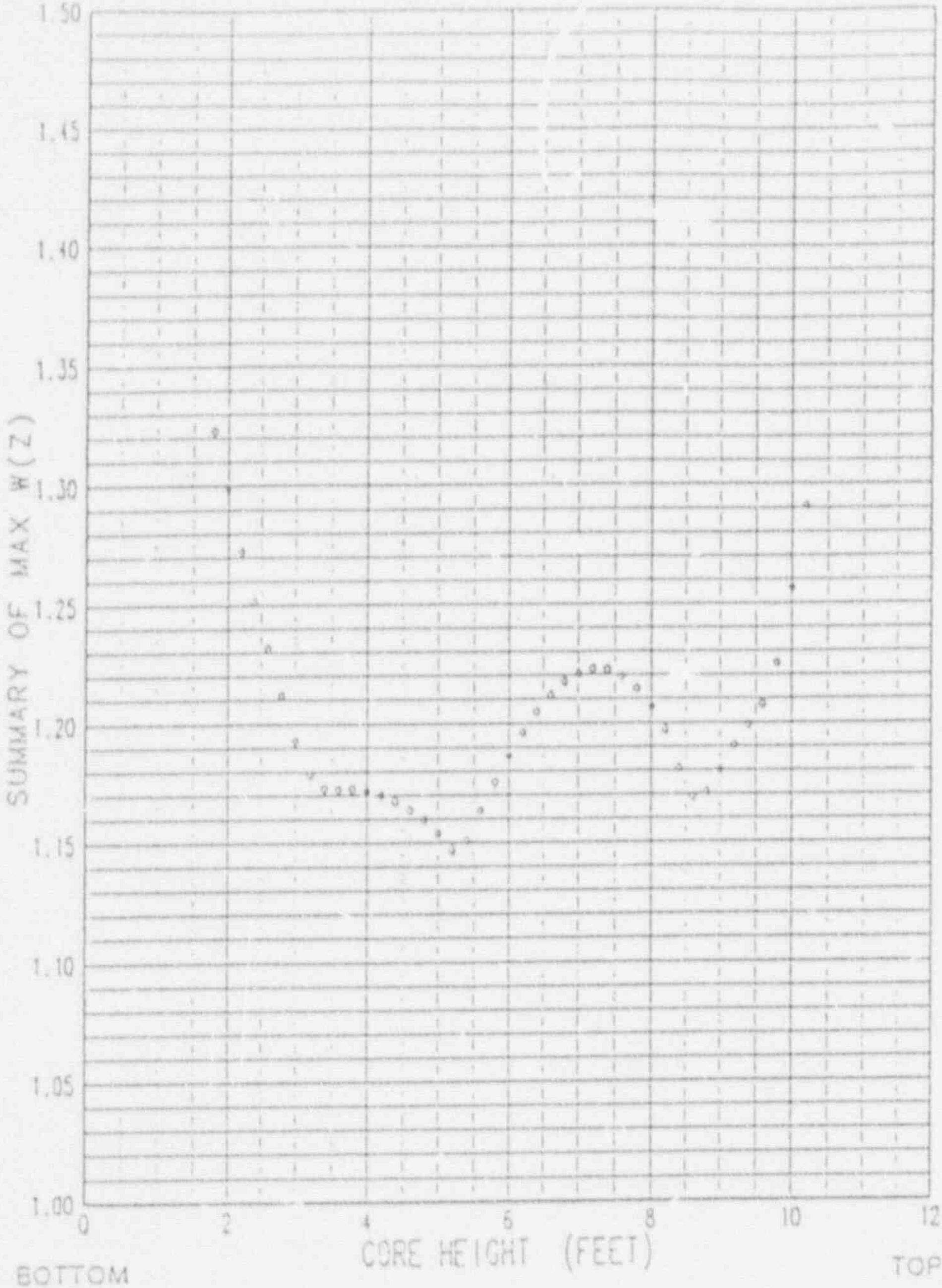


FIGURE 3

K(Z) - Normalized Fq(Z) as a Function of Core Height



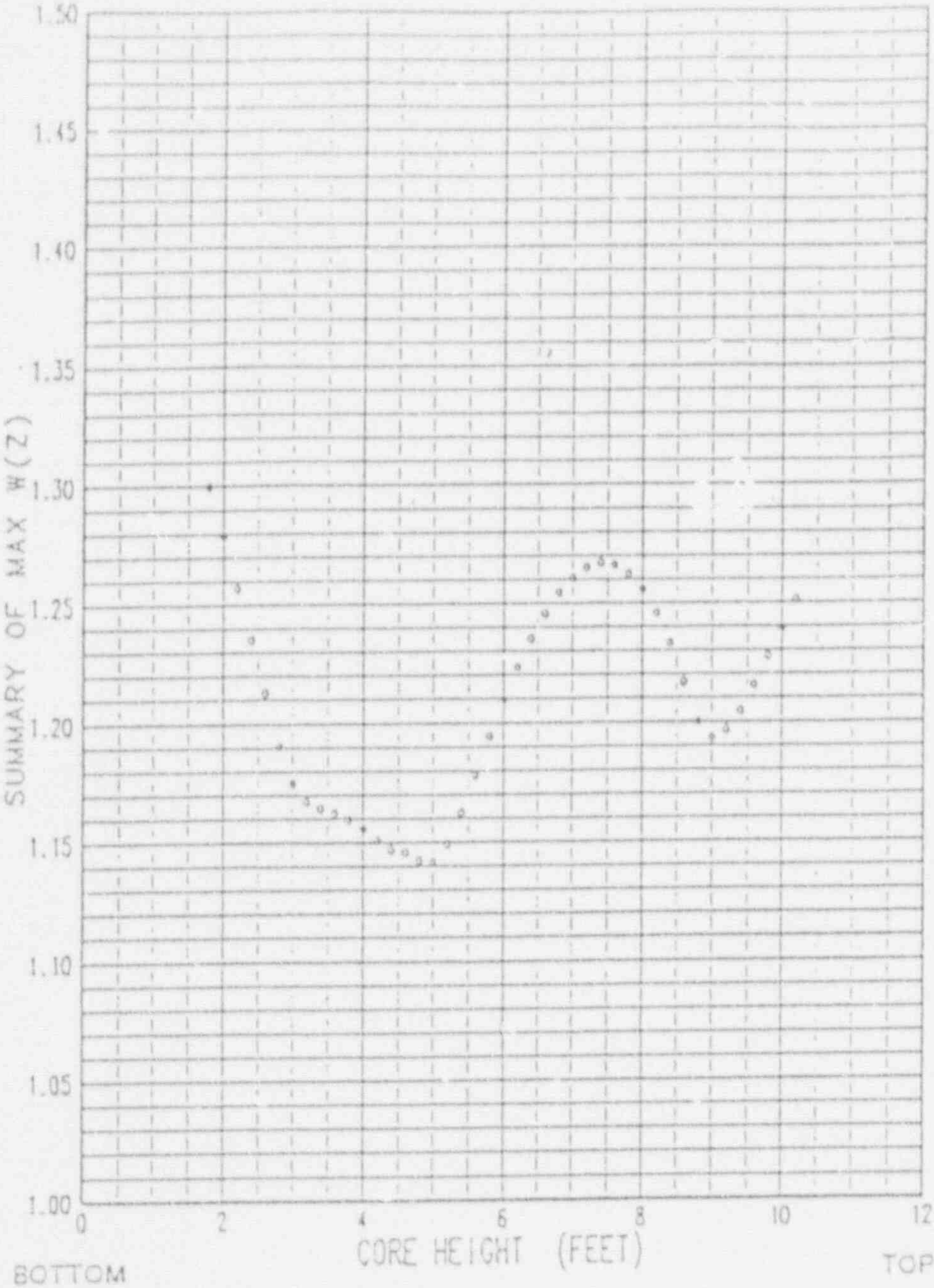
BU 150

	Height (Feet)	MAX W(Z)
*	0.0000	1.0000
*	0.0000	1.0000
*	0.2000	1.0000
*	0.4000	1.0000
*	0.6000	1.0000
*	0.8000	1.0000
*	1.0000	1.0000
*	1.2000	1.0000
*	1.4000	1.0000
*	1.6000	1.0000
	1.8000	1.3231
	2.0000	1.2980
	2.2000	1.2724
	2.4000	1.2511
	2.6000	1.2321
	2.8000	1.2121
	3.0000	1.1927
	3.2000	1.1785
	3.4000	1.1729
	3.6000	1.1724
	3.8000	1.1728
	4.0000	1.1719
	4.2000	1.1700
	4.4000	1.1673
	4.6000	1.1637
	4.8000	1.1598
	5.0000	1.15[illegible]
	5.2000	1.1467
	5.4000	1.1510
	5.6000	1.1634
	5.8000	1.1754
	6.0000	1.1863
	6.2000	1.1962
	6.4000	1.2047
	6.6000	1.2117
	6.8000	1.2172
	7.0000	1.2209
	7.2000	1.2227
	7.4000	1.2224
	7.6000	1.2197
	7.8000	1.2144
	8.0000	1.2070
	8.2000	1.1968
	8.4000	1.1809
	8.6000	1.1688
	8.8000	1.1709
	9.0000	1.1801
	9.2000	1.1905
	9.4000	1.1992
	9.6000	1.2078
	9.8000	1.2249
	10.000	1.2567
	10.200	1.2909
*	10.400	1.0000
*	10.600	1.0000
*	10.800	1.0000
*	11.000	1.0000
*	11.200	1.0000
*	11.400	1.0000
*	11.600	1.0000
*	11.800	1.0000
*	12.000	1.0000

FIGURE 4

RAOC SUMMARY OF MAX W(Z) AT 150 MWD/MTU

* TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G



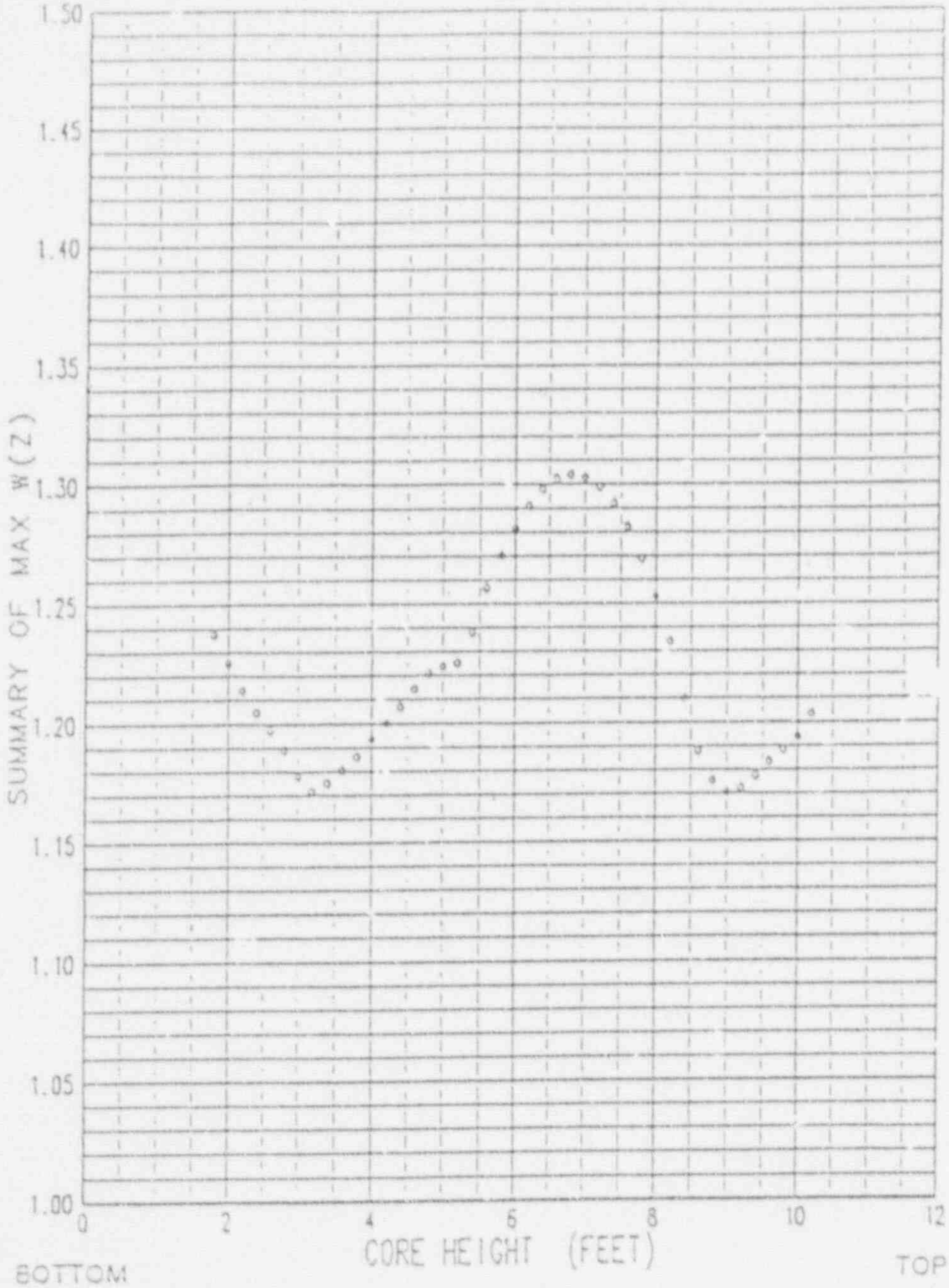
BU 4000

Height (Feet)	MAX W(Z)
* 0.0000	1.0000
* 0.2000	1.0000
* 0.4000	1.0000
* 0.6000	1.0000
* 0.8000	1.0000
* 1.0000	1.0000
* 1.2000	1.0000
* 1.4000	1.0000
* 1.6000	1.0000
1.8000	1.2999
2.0000	1.2787
2.2000	1.2570
2.4000	1.2355
2.6000	1.2130
2.8000	1.1906
3.0000	1.1750
3.2000	1.1672
3.4000	1.1643
3.6000	1.1623
3.8000	1.1596
4.0000	1.1559
4.2000	1.1508
4.4000	1.1469
4.6000	1.1458
4.8000	1.1426
5.0000	1.1418
5.2000	1.1491
5.4000	1.1623
5.6000	1.1782
5.8000	1.1945
6.0000	1.2096
6.2000	1.2232
6.4000	1.2353
6.6000	1.2457
6.8000	1.2543
7.0000	1.2608
7.2000	1.2651
7.4000	1.2669
7.6000	1.2661
7.8000	1.2624
8.0000	1.2558
8.2000	1.2460
8.4000	1.2332
8.6000	1.2169
8.8000	1.1998
9.0000	1.1932
9.2000	1.1965
9.4000	1.2044
9.6000	1.2154
9.8000	1.2276
10.000	1.2393
10.200	1.2512
* 10.400	1.0000
* 10.600	1.0000
* 10.800	1.0000
* 11.000	1.0000
* 11.200	1.0000
* 11.400	1.0000
* 11.600	1.0000
* 11.800	1.0000
* 12.000	1.0000

FIGURE 5

RAOC SUMMARY OF MAX W(Z) AT 4000 MWD/MTU

* TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G



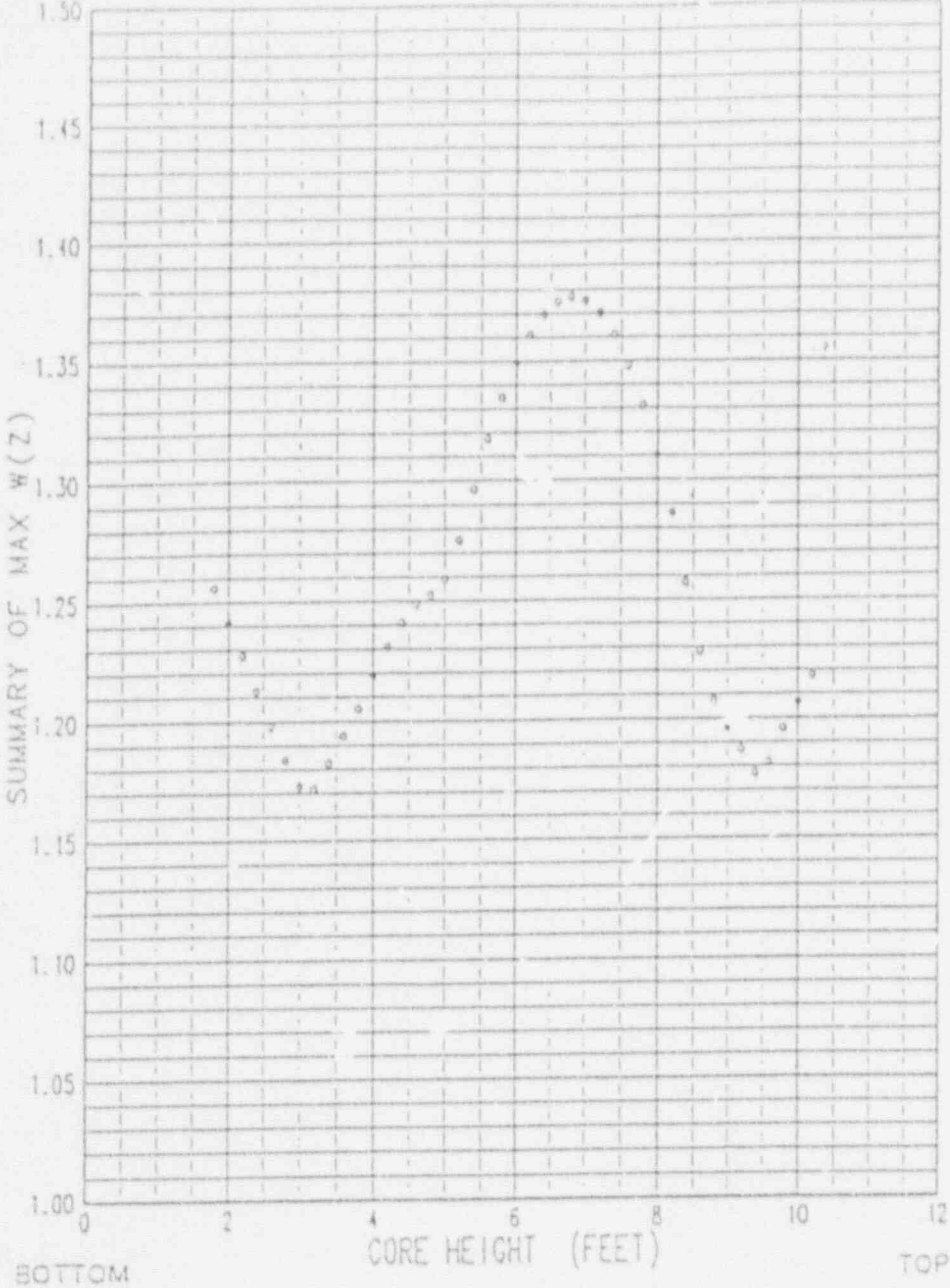
BU 10000

	Height (Feet)	MAX W(Z)
*	0.0000	1.0000
*	0.2000	1.0000
*	0.4000	1.0000
*	0.6000	1.0000
*	0.8000	1.0000
*	1.0000	1.0000
*	1.2000	1.0000
*	1.4000	1.0000
*	1.6000	1.0000
	1.8000	1.2376
	2.0000	1.2255
	2.2000	1.2140
	2.4000	1.2049
	2.6000	1.1973
	2.8000	1.1839
	3.0000	1.1710
	3.2000	1.1717
	3.4000	1.1750
	3.6000	1.1804
	3.8000	1.1862
	4.0000	1.1933
	4.2000	1.2001
	4.4000	1.2070
	4.6000	1.2145
	4.8000	1.2209
	5.0000	1.2233
	5.2000	1.2254
	5.4000	1.2381
	5.6000	1.2566
	5.8000	1.2703
	6.0000	1.2816
	6.2000	1.2911
	6.4000	1.2979
	6.6000	1.3023
	6.8000	1.3040
	7.0000	1.3029
	7.2000	1.2990
	7.4000	1.2920
	7.6000	1.2820
	7.8000	1.2688
	8.0000	1.2528
	8.2000	1.2339
	8.4000	1.2101
	8.6000	1.1880
	8.8000	1.1753
	9.0000	1.1706
	9.2000	1.1721
	9.4000	1.1774
	9.6000	1.1832
	9.8000	1.1882
	10.000	1.1938
	10.200	1.2034
*	10.400	1.0000
*	10.600	1.0000
*	10.800	1.0000
*	11.000	1.0000
*	11.200	1.0000
*	11.400	1.0000
*	11.600	1.0000
*	11.800	1.0000
*	12.000	1.0000

FIGURE 6

RAOC SUMMARY OF MAX W(Z) AT 10000 MWD/MTU

* TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G



BU 14000

Height (Feet)	MAX W(Z)
* 0.0000	1.0000
* 0.2000	1.0000
* 0.4000	1.0000
* 0.6000	1.0000
* 0.8000	1.0000
* 1.0000	1.0000
* 1.2000	1.0000
* 1.4000	1.0000
* 1.6000	1.0000
1.8000	1.2561
2.0000	1.2419
2.2000	1.2275
2.4000	1.2129
2.6000	1.1980
2.8000	1.1837
3.0000	1.1728
3.2000	1.1717
3.4000	1.1825
3.6000	1.1938
3.8000	1.2[illegible]2
4.0000	1.[illegible]1
4.2000	1.2311
4.4000	1.2409
4.6000	1.2488
4.8000	1.2526
5.0000	1.2590
5.2000	1.2752
5.4000	1.2965
5.6000	1.3175
5.8000	1.3349
6.0000	1.3494
6.2000	1.3612
6.4000	1.3697
6.6000	1.3750
6.8000	1.3769
7.0000	1.3753
7.2000	1.3700
7.4000	1.3609
7.6000	1.3480
7.8000	1.3310
8.0000	1.3105
8.2000	1.2864
8.4000	1.2571
8.6000	1.2284
8.8000	1.2076
9.0000	1.1961
9.2000	1.1869
9.4000	1.1769
9.6000	1.1815
9.8000	1.1956
10.000	1.2068
10.200	1.2180
* 10.400	1.0000
* 10.600	1.0000
* 10.800	1.0000
* 11.000	1.0000
* 11.200	1.0000
* 11.400	1.0000
* 11.600	1.0000
* 11.800	1.0000
* 12.000	1.0000

FIGURE 7

RAOC SUMMARY OF MAX W(Z) AT 14000 MWD/MTU

* TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G