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50	09/27/82	TS 6-2a, TS 6-5 thru TS 6-8, TS 6-11, TS 6-12, TS 6-17, TS 6-26 thru TS 6-28				

physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components:

- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility".

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: Part 20, Section 30.34 of Part 30 Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensees are authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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- (3) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1, 3.1.2 and 3.1.4 through 3.1.28 of the Fire Protection Safety Evaluation Report. These modifications shall be completed by the dates specified in Table 3.1. Dates for resolution of items are specified in Table 3.2. In the event that these dates for completion cannot be met, the licensee shall submit a report explaining the circumstances and propose a revised schedule.

(4) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). These approved documents consist of information withheld from public disclosures pursuant to 10 CFR 2.790 (d).

- a) "Industrial Security Manual" dated May 25, 1977, January 9, 1978, December 18, 1978, January 30, 1979, March 7, 1979 and March 27, 1979.
- b) Kewaunee Nuclear Power Plant Safeguards Contingency Plan, as originally submitted by letter of March 27, 1979, and subsequently revised and re-submitted by letter of February 20, 1981, pursuant to 10 CFR 73.40. The Safeguards Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b) within 30 days of this approval by the Commission.

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5. Pressurizer Power Operated Relief Valves (PORV) and PORV Block Valves.

- a. Two PORV's and their associated block valves shall be operable during hot standby and operating modes.
 - 1. If a pressurizer PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed and maintained closed by administrative procedures to prevent inadvertent opening.
 - 2. If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the valve; otherwise the unit shall be placed in the hot shutdown condition using normal operating procedures.

6. Pressurizer Heaters

- A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is greater than 350°F.

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- F.1. Isolation valves 8806A (SI9A), 8801A (SI11A) and 8801B (SI11B) in the discharge of the high head SIS and block valve 8809C (SI13) are in the open position with their power breaker locked out.
2. Accumulator isolation valves 8800A (SI20A) and 8800B (SI20B) shall be opened with their power breaker locked out before reactor coolant system pressure exceeds 1000 psig.
- G. Automatic valves, instrumentation, piping, and interlocks associated with the above components and required to function during accident and/or post-accident conditions are operable. |47
- H. During the Monthly Valve Operation Surveillance Testing of the Safety Injection System it is permissible to close the hand operated valve isolating the Concentrated Boric Acid Tanks from the Safety Injection Pump Suction. During this short test period an operator shall stand by the valve to open it if Safety Injection is required. He will have headset communication with the Control Room. At completion of the test he will verify the valve is returned to open, and this will be checked by at least one additional person.
2. During power operation or recovery from inadvertent trip, any one of the following conditions of inoperability may exist during the time intervals specified. The reactor shall be placed in the hot shutdown condition if operability is not restored within the time specified, and it shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.
- A. ONE safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. The other safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.
- B. ONE residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours. The other residual heat removal pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.
- C. ONE residual heat exchanger may be out of service for a period of no more than 48 hours.

3.5 INSTRUMENTATION SYSTEM

Applicability

Applies to reactor protection and engineered safety features instrumentation systems.

Objective

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded, and to delineate the conditions of the reactor protection instrumentation and engineered safety features circuits necessary to ensure reactor safety.

Specification

- a. Setting limits for instrumentation which initiate operation of the engineered safety features shall be as stated in Table TS 3.5-1.
- b. For on-line testing or in the event of failure of a sub-system instrumentation channel, plant operation shall be permitted to continue at rated power in accordance with Tables TS 3.5-2 through TS 3.5-5. | 42
- c. If for Tables TS 3.5-2 through TS 3.5-5 the number of channels of a particular sub-system in service falls below the limits given in Column Three, or if the values in Column Four cannot be achieved, operation shall be limited according to the requirement shown in Column 6, as soon as practicable. | 42
- d. In the event of sub-system instrumentation channel failure permitted by Specification 3.5.b, Tables TS 3.5-2 through TS 3.5-5 need not be observed during the short period of time (approximately 4 hours) the operable sub-system channels are tested, where the failed channel must be blocked to prevent unnecessary reactor trip. | 42
- e. The instrumentation in Table 3.5-5 shall be operable. In the event the limits given in column 1 and 2 cannot be maintained operator action will be in accordance with the respective notes. | 38

6. The set points and associated ranges for the undervoltage relays have been established to always maintain motor voltages at or above 80% of their nameplate rating and to prevent prolonged operation of motors below 90% of their nameplate rating. All safeguard motors were designed to accelerate their loads to operating speed with 80% nameplate voltage, but not necessarily within their design temperature rise. Prolonged operation below 90% of nameplate voltage may result in shortening of motor insulation life, but short term operation below 90% of nameplate voltage will not result in unacceptable effects due to the service factor provided in the motors and the conservative insulation system used on the motors.

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The primary safeguard buses undervoltage trip (87.5% of nominal bus voltage) is designed to protect against a loss of voltage to the safeguard bus and assures that safeguard protection action will proceed as assumed in the FSAR. The associated time delay feature prevents inadvertent actuation of the undervoltage relays from voltage dips, while assuring that the diesel generators will reach full capacity before the safety injection pump loads are sequenced on.

The safeguard buses second level undervoltage trip (95% of nominal bus voltage) is designed to protect against prolonged operation below 90% of nameplate voltage of safeguard pumps. The time delay of less than 5 minutes allows the operator time to restore voltage by minimizing or balancing loads on the safeguard buses while maintaining the preferred source of power. Up to 5 minutes of operation of safeguard pumps between 80% and 90% of nameplate voltage is acceptable due to the service factor and conservative insulation designed into the motors.

Each relay in the undervoltage protection channels will fail safe and is alarmed to alert the operator to the failure.

A blackout signal which occurs during the sequence loading following a safety injection signal will result in a reinitiation of the sequence loading logic at time step 0 as long as the Safety Injection signal has not been re-set. The Kewaunee Emergency Procedures warn the operators that a Blackout Signal occurring after reset of Safety Injection will not actuate the sequence loading and instructs to re-initiate Safety Injection if needed.

Instrument Operating Conditions

During plant operations, the complete protective instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection Systems, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in another channel.

The operability of the instrumentation noted in Table 3.5-5 assures that sufficient information is available on these selected plant parameters to aid the operator in identification of an accident and assessment of plant conditions during and following an accident. In the event the instrumentation noted in Table 3.5-5 is not operable, the operator is given instruction on compensatory actions.

References:

- (1) FSAR Section 7.5
- (2) FSAR Section 14.3
- (3) FSAR Section 14.2.5

TS 3.5-6

Amendment No. 47
11/29/82

Specification 3.9.a.7 limits the amount of radioactivity that may be inadvertently released to the environment.

b. Airborne Effluents

1. The release rate of gross gaseous activity, except for halogens and particulates with half-lives longer than eight days, shall be limited to $3.6 \times 10^{-12} \frac{\text{sec}}{\text{cc}} \left\{ \sum \frac{O_i}{\text{MPC}_i} \right\} \leq 1$ where O_i is the release rate in $\mu\text{Ci}/\text{sec}$ for isotope i , and MPC_i is the maximum permissible concentration of isotope i as defined in Appendix B, table II, Column 1, 10 CFR 20. The 3.6×10^{-12} sec/cc value includes the conversion factor of m^3 to cc.
2. The release rate of halogens and particulates with half-lives greater than eight days released to the environs as part of airborne effluents, shall be controlled such that the release rate over any one hour period does not exceed $5.1 \times 10^{-1} \mu\text{Ci}/\text{sec}$.
3.
 - a. The release rates of gross gaseous activity shall not exceed 16 percent of the value specified in 3.9.b.1 above, when averaged over any calendar quarter.
 - b. The release rates of halogens and particulates with half-lives greater than eight days shall not exceed 12 percent of the value specified in 3.9.b.2 above, when averaged over any calendar quarter.
4. During release of gaseous wastes, the following conditions shall be met:
 - a. The gross activity monitor, the iodine activity sampler and particulate activity sampler located in the release path shall be operable. |47
 - b. Automatic isolation devices capable of terminating the gaseous release shall be operable.

c. For effluent streams having continuous monitoring capability, the gross, halogen, and particulate activity and flowrate shall be monitored and recorded.

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For effluent streams without continuous monitoring capability the gross, halogen, and particulate activity along with the release volume shall be monitored and recorded.

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5. Radioactive gaseous wastes collected in the gas decay tanks shall be held up a minimum of 45 days, except for those gaseous wastes resulting from purge and fill operations associated with refueling and reactor startup. Releases of radioactive gaseous wastes at less than 1/100 the limits specified by 3.9.h.1 and 3.9.b.2 are permitted at any time as required for operational flexibility.
6. Reactor containment building purge shall be filtered through the purge filter (HEPA - charcoal) whenever the concentration of iodine and particulate isotopes exceeds the occupational MPC inside the reactor building.
7. The maximum activity to be contained in one gas decay tank shall not exceed 43,500 curies. (Equivalent to Xe-133).
8. Gaseous waste from the condenser air ejector shall be filtered through HEPA filters provided in the Auxiliary Building Vent System.
9. When the annual projected release rate of radioactive materials in gaseous wastes, averaged over a calendar quarter exceeds twice the annual objectives, the licensee shall notify the Director, Directorate

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS.

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objective

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the hot shutdown margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron, or part length rod position.

b. Power Distribution Limits

1. At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

A. $F_Q^N(Z)$ Limits:

(i) Westinghouse Electric Corporation Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.22/P) \times K(Z) \text{ for } P > .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.44) \times K(Z) \text{ for } P \leq .5$$

(ii) Exxon Nuclear Company Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq F_Q^T(E_j)/P \times K(Z) \text{ for } P \geq .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.42) \times K(Z) \text{ for } P \leq .5$$

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where:

P is the fraction of full power at which the core is operating

K(Z) is the function given in Figure TS 3.10-2

Z is the core height location for the F_Q of interest

F_Q^T(E_j) is the function given in Figure TS 3.10-6

E_j is exposure of the fuel rod for the F_Q of interest

B. F_{ΔH}^N Limits For All Fuel

F_{ΔH}^N x 1.04 ≤ 1.55 (1 + 0.2(1 - P)) For 0 to 24,000 MWD/MTU burnup fuel

F_{ΔH}^N x 1.04 ≤ 1.52 (1 + 0.2(1 - P)) For greater than 24,000 MWD/MTU burnup fuel

where:

P is the fraction of full power at which the core is operating

2. If, for any measured hot channel factor, the relationships specified in 3.10.b.1 are not true, reactor power shall be reduced by a fractional amount of the design power to a value for which the relationships are true, and the high neutron flux trip setpoint shall be reduced by the same fractional amount. If subsequent incore mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.
3. Following initial loading and at regular effective full power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of specification 3.10.b.1 are satisfied.
4. The measured F_Q^{EQ}(Z) hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core:

A. Westinghouse Electric Corporation Fuel

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.22/P) \times K(Z)$$

B. Exxon Nuclear Company Fuel

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq F_Q^T(E_j)/P \times K(Z)$$

where:

P is the fraction of full power at which the core is operating

$V(Z)$ is defined in Figure TS 3.10.-7.

$F_Q^{EQ}(Z)$ is a measured FQ distribution obtained during the target flux determination

5. Power distribution maps using the movable detector system shall be made to confirm the relationship of specification 3.10.b.4 according to the following schedules with allowances for a 25% grace period:
 - A. During the target flux difference determination or once per effective full power monthly interval whichever occurs first.
 - B. Upon achieving equilibrium conditions after reaching a thermal power level more than 10% higher than the power level at which the last power distribution measurement was performed in accordance with 3.10.b.5.A above.
 - C. If a power distribution map indicates an increase in peak pin power, $F_{\Delta H}^N$, of 2% or more, due to exposure, when compared to the last power distribution map either of the following actions shall be taken:
 - i. $F_Q^{EQ}(Z)$ shall be increased by an additional 2% for comparison to the relationship specified in 3.10.b.4 OR
 - ii. $F_Q^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every 7 effective full power days until a power distribution map indicates that the peak pin power, $F_{\Delta H}^N$, is not increasing with exposure when compared to the last power distribution map.
6. If, for a measured F_Q^{EQ} , the relationships of 3.10.b.4 are not satisfied and the relationships of 3.10.b.1 are satisfied, within 12 hours take one of the following actions:

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- A. Take corrective actions to improve the power distribution and upon achieving equilibrium conditions measure the target flux difference and verify that the relationships specified in 3.10.b.4 are satisfied, OR
- B. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the left hand sides of the relationships specified in 3.10.b.4 exceed the limits specified in the right hand sides.
7. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per full power month.
8. The indicated axial flux difference shall be considered outside of the limits of sections 3.10.b.9 through 3.10.b.12 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.
9. Except during physics tests, during excore detector calibration and except as modified by 3.10.b.10 through 3.10.b.12 below, the indicated axial flux difference shall be maintained within a $\pm 5\%$ band about the target flux difference.
10. At a power level greater than 90 percent of rated power if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent of rated power.
11. At power levels greater than 50 percent and less than or equal to 90 percent of rated power:
- A. The indicated axial flux difference may deviate from its $\pm 5\%$ target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by

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-10 percent and +10 percent from the target axial flux difference at 90% rated power and increasing by -1% and +1% from the target axial flux difference for each 2.7% decrease in rated power below 90% and above 50%. If the cumulative time exceeds one hour, then the reactor power shall be reduced immediately to less than or equal to 50% power and the high neutron flux setpoint reduced to less than or equal to 55% of rated power.

B. A power increase to a level greater than 90% of rated power is contingent upon the indicated axial flux difference being within its target band.

12. At a power level no greater than 50% of rated power:

A. The indicated axial flux difference may deviate from its target band.

B. A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) of the preceding 24 hour period.

One half of the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90% of rated power.

13. Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.b.10 or the flux difference time requirement of 3.10.b.11A. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

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c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio exceeds 1.02, one of the following actions shall be taken within two hours:
 - A. Eliminate the tilt.
 - B. Restrict maximum core power level two percent for every one percent of indicated power tilt ratio exceeding 1.0.
2. If the tilt condition is not eliminated after 24 hours, reduce power to 50 percent or lower.
3. Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:
 - A. Restrict maximum core power level by 2 percent of rated values for every one percent of indicated power tilt ratio exceeding 1.0.
 - B. If the tilt condition is not eliminated within 12 hours, the reactor shall be brought to a minimum load condition (≤ 30 Mwe).
4. If the indicated quadrant tilt exceeds 1.09 and there is no simultaneous indication of rod misalignment, the reactor shall immediately be brought to a No Load condition ($\leq 5\%$ reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.
2. The control banks shall be limited in physical insertion; insertion limit is shown in Figure TS 3.10-3.
3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure TS 3.10-1 must be maintained except for the low power physics test

to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted and the part length rods fully withdrawn.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In specifications 3.10.e.1 and 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

1. When reactor power is greater than or equal to 85% of rating the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 12 steps when reactor power is greater than or equal to 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and specification 3.10.b applied. If peaking factors are not determined within 4 hours, the reactor power shall be reduced to less than 85% of rating.
2. When reactor power is less than 85% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is less than 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and specification 3-10.b applied.
3. And, in addition to 3.10.e.1 and 3.10.e.2 above, if the misaligned rod cluster control assembly is not realigned within 8 hours, the rod shall be declared inoperable.

f. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service, then:
 - A. For operation between 50 percent and 100 percent of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.
 - B. During operation below 50 percent of rating, no special monitoring is required.
2. Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.
3. If a rod cluster control assembly having a rod position indicator channel out of service is found to be misaligned from 3.10.f.1.(A) above, then specification 3.10.e will be applied.

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g. Inoperable Rod Limitations

1. An inoperable rod is a rod which does not trip or which is declared inoperable under specification 3.10.e or 3.10.h.

BASIS

SHUTDOWN REACTIVITY

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. To maintain the required trip reactivity, the rod insertion limits of Figure TS 3.10-3 must be observed. In addition, for hot shutdown conditions, the shutdown margin of Figure TS 3.10-1 must be provided for protection against the steamline break accident which requires more shutdown reactivity at end of core life (due to a more negative moderator temperature coefficient at end-of-life boron concentrations).

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. The available control rod reactivity or excess beyond needs, decreases with decreasing boron concentration, because the negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low.

The exception to the rod insertion limits in Specification 3.10.d.3 is to allow the measurement of the worth of all rods less the worth of the worst case of an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest, such as end-of-life cooldown or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

POWER DISTRIBUTION CONTROL

Criteria

Criteria have been chosen for Condition I and II events as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First the peak value of linear power density must not exceed the value assumed in the accident analysis.^{1, 3} Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.²

In addition to conditions imposed for Condition I and II events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F.

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local neutron flux in the core at core elevation Z divided by the core averaged neutron flux, assuming nominal fuel and rod dimensions.

$F_Q^{EQ}(Z)$ is the measured F_Q^N distribution obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for F_Q^N defined by specification 3.10.b.1 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope indicate that peak clad temperatures remain below the 2200°F limit. | 41

The $F_Q^N(Z)$ limits of specification 3.10.b.1.A include consideration of enhanced fission gas release at high burnup, off-gassing (release of absorbed gases), and other effects in fuel supplied by Exxon Nuclear Company; this results in an additional penalty in the form of the function $F_Q^T(Ej)$, as shown in Figure TS 3.10-6, which is applied to Exxon fuel. References 7 and 8 discuss these phenomena. | 41

When a F_Q^N measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. | 41

In specification 3.10.b.1 and 3.10.b.4 F_Q^N is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests). | 41

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor

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

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

In the specified limit of $F_{\Delta H}^N$ there is an 8% allowance for uncertainties¹ which means that normal operation of the core is expected to result in $F_{\Delta H}^N < 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q^N , (b) the operator has a direct influence on F_Q^N through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^N$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available. When a measurement of $F_{\Delta H}^N$ is taken, experimental error must be allowed for and 4% is the appropriate allowance.

The use of $F_{\Delta H}^N$ in specification 3.10.b.5 is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

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The $F_{\Delta H}^N$ limits of specification 3.10.b.1 include consideration of fuel rod bow effects. Since the effects of rod bow are dependent on fuel burnup an additional penalty is incorporated in a decrease in the $F_{\Delta H}^N$ limit of 2% for 0-15000 MWD/MTU fuel burnup, 4% for 15000-24000 MWD/MTU fuel burnup, and 6% for greater than 24000 MWD/MTU fuel burnup. These penalties are counter-balanced by credits for increased Reactor Coolant flow and lower Core inlet temperature. The Reactor Coolant System flow has been determined to exceed design by greater than 8%. Since the flow channel protective trips are set on a percentage of full flow, significant margin to DNB is provided. One half of the additional flow is taken as a DNB credit to offset 2% of the $F_{\Delta H}^N$ penalty. The existence of 4% additional reactor coolant flow will be verified after each refueling at power prior to exceeding 95% power. If the reactor coolant flow measured per loop averages less than 92560 gpm, the $F_{\Delta H}^N$ limit shall be reduced at the rate of 1% for every 1.8% of reactor coolant design flow (89000 gpm design flow rate) for fuel with greater than 15000 MWD/MTU burnup. Uncertainties in reactor coolant flow have already been accounted for in the flow channel protective trips for design flow. The assumed T_{inlet} for DNB analysis was 540°F while the normal T_{inlet} at 100% power is approximately 532°F. The reduction of maximum allowed T_{inlet} at 100% power to 536°F as addressed in specification 3.10.k provides an additional 2% credit to offset the rod bow penalty. The combination of the penalties and offsets results in a required 2% reduction of allowed $F_{\Delta H}^N$ for high burnup fuel, 24000 MWD/MTU. The permitted relaxation in $F_{\Delta H}^N$ allows radial power shape changes with rod insertion to the insertion limits.

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Surveillance

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Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would, otherwise, affect these bases.

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For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is $\geq 85\%$, or an indicated 24 steps when reactor power is $< 85\%$.
2. Control rod banks are sequenced with overlapping banks as shown in Figure TS 3.10-3.
3. The control bank insertion limits are not violated.
4. Axial power distribution control specifications which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

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The specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.⁹

Conformance with specification 3.10.b.9 through 3.10.b.12 ensures the F_Q^N upper bound envelope is not exceeded and xenon distributions will not develop which at a later time would cause greater local power peaking.

At the beginning of cycle, power escalation may proceed without the constraints of section 3.10.b.5 since the startup test program provides adequate surveillance to ensure peaking factor limits. Target flux difference surveillance is initiated after achieving equilibrium conditions for sustained operation. 41

The target (or reference) value of flux difference is determined as follows.

At any time that equilibrium xenon conditions have been established, the indicated flux difference is determined from the nuclear instrumentation. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviations of $\pm 5\%$ flux difference are permitted from the indicated reference value. Figure TS 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure TS 3.10-4 shows the typical variation of the full power value with burnup. 41

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10% increment in peaking factor for flux difference in the range +10% to -10% from the target flux increasing by 1% from the target axial flux difference for each 2.7% decrease in rated power below 90% and above 50%. Therefore, while the deviation exists the power level is limited to 90% or lower depending on the indicated flux difference without additional core monitoring. If, for any reason, flux difference is not controlled within the

+5% band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50% is required to protect against potentially more severe consequences of some accidents unless incore monitoring is initiated.

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As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with the specifications is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

QUADRANT POWER TILT LIMITS

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt

alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This symmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level. This monitoring is required by Technical Specifications, Section 4.1.

The two hour time interval in specification 3.10.c is considered ample to identify a dropped or misaligned rod. In the event that the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the movable detector system. For a tilt condition ≤ 1.09 an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment. In the event a tilt condition of ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for flux mapping and turbine synchronization.

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a low power condition for investigation by flux

mapping. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2% for each 1% the tilt ratio exceeds 1.0) for the 8 hour period necessary to correct the rod misalignment.

ROD MISALIGNMENT LIMITATIONS

During normal power operation it is desirable to maintain the rods in alignment with their respective banks to provide consistency with the assumption of the safety analyses, to maintain symmetric neutron flux and power distribution profiles, to provide assurance that peaking factors are within acceptable limits and to assure adequate shutdown margin.

Analyses have been performed which indicate that the above objectives will be met if the rods are aligned within the limits of Specification 3.10.e. A relaxation in those limits for power levels below 85% is allowable because of the increased margin in peaking factors and available shutdown margin obtained while operating at lower power levels. This increased flexibility is desirable to account for the non-linearity inherent in the rod position indication system and for the effects of temperature and power as seen on the rod position indication system.

Rod position measurement is performed through the effects of the rod drive shaft metal on the output voltage of a series of vertically stacked coils located above the head of the reactor pressure vessel. The rod position can be determined by the analog individual rod position indicators, the plant process computer which receives a voltage input from the conditioning module, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

The plant process computer converts the output voltage signal from each IRPI conditioning module to an equivalent position (in steps) through a curve fitting process, which may include the latest actual voltage-to-position rod calibration curve.

The rod position as determined by any of these methods can then be compared to the bank demand position which is indicated on the group step counters to determine the existence and magnitude of a rod misalignment. This comparison is performed automatically by the plant process computer. The rod deviation monitor on the annunciator panel is activated (or re-activated) if the two position signals for any rod as detected by the process computer deviate by more than a predetermined value. The value of this setpoint is set to warn the operator when the technical specification limits are exceeded.

The rod position indicator system is calibrated once per refueling cycle and forms the basis of the correlation of rod position vs. voltage. This calibration is typically performed at hot shutdown conditions prior to initial operations for that cycle. Upon reaching full power conditions and verifying that the rods are aligned with their respective banks the rod position indication may be adjusted to compensate for the effects of the power ascension. After this adjustment is performed, the calibration of the rod position indicator channel is checked at an intermediate and low level to confirm that the calibration is not adversely affected by the adjustment.

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INOPERABLE ROD POSITION INDICATOR CHANNELS

The rod position indicator channel is sufficiently accurate to detect a rod ± 7.5 inches away from its demand position. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

INOPERABLE ROD LIMITATIONS

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30 day period is provided for the re-analysis of all accidents sensitive to the changed initial condition.

ROD DROP TIME

The required drop time to dashpot entry is consistent with safety analysis.

DNB PARAMETERS

The DNB related accident analysis assumed as initial conditions that the T_{inlet} was $4^{\circ}F$ above nominal design or T_{avg} was $4^{\circ}F$ above nominal design. The Reactor Coolant System pressure was assumed to be 30 psi below nominal design.

REFERENCES

- (1) FSAR Section 4.3
- (2) FSAR Section 4.4
- (3) FSAR Section 14
- (4) "Rod Misalignment Analysis," July 27, 1981, submitted to NRC with proposed Technical Specification Amendment 46 by letter from E. R. Mathews (WPSC) to D. G. Eisenhut (NRC) dated August 7, 1981.

- (5) Letter from E. R. Mathews, (WPSC), to D. G. Eisenhut, (NRC), dated January 8, 1980, submitting information on Clad Swelling and Fuel Blockage Models.
- (6) Letter from E. R. Mathews, (WPSC), to A. Schwencer, (NRC), dated December 14, 1979, submitting the ECCS Re-analysis properly accounting for the zirconium/water reaction.
- (7) George C. Cooke, Philip J. Valentine: "Exposure Sensitivity Study for ENC XN-1 Reload Fuel at Kewaunee Using the ENC-WREM-IIA PWR Evaluation Model, WN-NF-79-72," Exxon Nuclear Company, October, 1979.
- (8) Letter from L. C. O'Malley, (Exxon Nuclear Company) to E. D. Novak, (WPSC), providing FQ exposure dependence as a function of rod burnup. February 25, 1981
- (9) XN-NF-77-57 Exxon Nuclear Power Distribution Control for Pressurized Water Reactor, Phase II, January, 1978.

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TABLE TS 3.5-1 (Page 1 of 2)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

<u>NO.</u>	<u>FUNCTIONAL UNIT</u>	<u>CHANNEL</u>	<u>SETTING LIMIT</u>
1	High Containment Pressure (Hi)	Safety Injection ⁽¹⁾	≤ 4 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray	≤ 23 psig
		b. Steam Line Isolation of Both Lines	≤ 17 psig
3	Pressurizer Low Pressure	Safety Injection ⁽¹⁾	≥ 1815 psig
4	Low Steam Line Pressure	Safety Injection ⁽¹⁾	≥ 500 psig
		Lead Time Constant	≥ 12 seconds
		Lag Time Constant	≤ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{avg}	Steam Line Isolation Affected Line ⁽²⁾	\leq d/p corresponding to 0.745×10^6 lb/hr at 1005 psig $\geq 540^\circ$ F
6	High-High Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line ⁽²⁾	\leq d/p corresponding to 4.5×10^6 lb/hr at 735 psig
7	Forebay Level	Trip circ. water pumps	
8	Containment Purge and Vent System Radiation Particulate Detector Radioactive Gas Detector	Containment Ventilation Isolation	\leq value of Radiation Levels in exhaust duct as defined in Note ⁽³⁾

Table TS 3.5-1 (Page 1 of 2)

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TABLE TS 3.5-1 (Page 2 of 2)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

<u>NO.</u>	<u>FUNCTIONAL UNIT</u>	<u>CHANNEL</u>	<u>SETTING LIMIT</u>	
9	Safeguards Bus Undervoltage (4)	Loss of Power	87.5% \pm 2% nominal bus voltage \leq 2.5 second time delay	42
10	Safeguards Bus Second Level (5) Undervoltage	Degraded Grid Voltage	95% \pm 2% of nominal bus voltage \leq 5 minutes time delay	42

- (1) Initiates containment isolation, feedwater line isolation, shield building ventilation, auxiliary building special vent, and starting of all containment fans. In addition, the signal overrides any bypass on the accumulator valves.
- (2) Confirm main steam isolation valves closure within 5 seconds when tested.
d/p = differential pressure
- (3) The setting limits for max radiation levels are derived from the technical specification allowable release rates found in Technical Specification 3.9.b.
- (4) This undervoltage protection channel ensures ESF equipment will perform as assumed in the FSAR.
- (5) This undervoltage protection channel protects ESF equipment from long term low voltage operation.

TABLE TS 3.5-5 (1 of 2)
INSTRUMENTATION OPERATING CONDITIONS FOR INDICATION

<u>NO.</u>	<u>FUNCTIONAL UNIT</u>	<u>REQUIRED TOTAL NO. OF CHANNELS*</u>	<u>MINIMUM CHANNELS OPERABLE**</u>
1	Auxiliary Feedwater Flow to Steam Generators (Narrow Range Level Indication already required operable by Tech Spec Table TS 3.5-2 Item 12).	1/steam gen	1/steam gen
2	Reactor Coolant System Subcooling Margin	2	1
3	Pressurizer Power Operated Relief Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve	1/valve
4	Pressurizer Power Operated Relief Block Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve	1/valve
5	Pressurizer Safety Valve Position (One Channel Temperature, and one Acoustic Sensor per valve)	2/valve	1/valve

* With the number of Operable monitoring instrumentation channels less than the Required Total Number of Channels shown, either restore the inoperable channels to Operable status within fourteen days, or be in at least Hot Shutdown within the next 12 hours.

** With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirements, either restore the minimum number of channels to Operable status within 72 hours or be in at least Hot Shutdown within the next 12 hours.

NOTE: Technical Specification 6.9.b.2 applies only when MINIMUM CHANNELS OPERABLE are less than shown.

TABLE TS 3.5-5 (1 of 2)

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TABLE TS 3.5-5 (2 of 2)

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INSTRUMENT OPERATION CONDITIONS FOR SAFEGUARDS BUS POWER SUPPLY FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
6	Safeguards Bus Undervoltage	2/Bus ⁽¹⁾	1/Bus	1/Bus ⁽²⁾	--		Maintain hot shut-down or operate the diesel generator
7	Safeguards Bus Second Level Undervoltage	1/Bus ⁽³⁾	1/Bus	--	--		When one of the two 6 second time delay relays is out of service, place that relay in the tripped condition.

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- (1) Each channel consists of one instantaneous and one time delayed relay connected in series.
- (2) When one component of a channel is taken out of service, that component shall be in the TRIPPED condition.
- (3) Each channel has 2 time delay relays in parallel which are in series with a third time delay relay.

TABLE TS 3.5-5 (2 of 2)

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TABLE 3.15-1

FIRE DETECTION INSTRUMENTATION

<u>Fire Area</u>	<u>Detectors</u>	<u>Minimum # Required</u>	<u>Required Actions</u>		
AX-21	4160 Switchgear Room	3	2	Establish an hourly fire watch inspection	39
AX-23	Special Vent Filter Housings	9	9	If filter housing is in operation with charcoal filters in service establish an hourly fire watch inspection. If not in service establish a 4-hour inspection frequency.	39
AX-23	Auxiliary Building	4	2	Establish an hourly fire watch inspection	39
AX-24	Fuel Handling Area	3	3	Establish an hourly fire watch inspection	
AX-30	Relay Room	19	6	Establish an hourly fire watch inspection	
AX-32	Cable run area	11	8	Establish an hourly fire watch inspection	
AX-35	Control Room	13	0	Control room is continuously manned	
AX-37	CKD Room	7	4	Establish an hourly fire watch inspection	
SB-65	Shield Building	6	2	Establish a four hour fire watch inspection	
SC-70	Screenhouse	4	2	Establish an hourly fire watch inspection	39
TU-90/91	D/G 1A and day tank room	7	5	Establish an hourly fire watch inspection	
TU-92/93	D/G 1B and day tank room	7	5	Establish an hourly fire watch inspection	
TU 94	Cardox Room	1	1	Establish an hourly fire watch inspection	39
TU 95	Air Compressor & Pump Room	5	4	Establish an hourly fire watch inspection	
TU 97	Battery Room 1A	1	1	Establish an hourly fire watch inspection	
TU 98	Battery Room 1B	1	1	Establish an hourly fire watch inspection	

Table TS 3.15-1

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TABLE TS 3.15-2

FIRE HOSE STATIONS

Location

- | | |
|---|----|
| 1. Adjacent to S/G Blowdown Tank and 4160 V Switchgear Rooms | |
| 2. Adjacent to Main Shop, Tank and Pump Room near Door 78 | 39 |
| 3. Adjacent to Control Room and A/C Equipment Room, 606 elevation near stairs | |
| 4. Screenhouse, north stairway leading to lower level | 39 |
| 5. Adjacent to D/G 1A and D/G 1A day tank rooms | |
| 6. Adjacent to D/G 1B and D/G 1B day tank rooms | |
| 7. Air Compressor and Pump Room near Auxiliary Feedwater Area Panel | 39 |
| 8. Adjacent to Oil Storage Room "B" and SWPT Pressure Filter Assembly | |
| 9. Adjacent to Battery Rooms 1A and 1B | |
| 10. Aux. Building Basement North of Freight elevator (A) | |
| 11. Aux. Building Basement North of Laundry Pumps on south wall of valve gallery. | |
| 12. Aux Building Basement solid radwaste handling area, west of MCC 1-45G | |
| 13. Aux Building Mezz. Southwest of BA Transfer Pumps | 39 |
| 14. Aux Building Mezz. South of S/G Blowdown Tank | |
| 15. Aux Building Operating Floor West of entrance to BA Tank Room | |
| 16. Aux Building Operating Floor East Side of RWST | |
| 17. Stair well at 616 elevation next to "G" wall | |

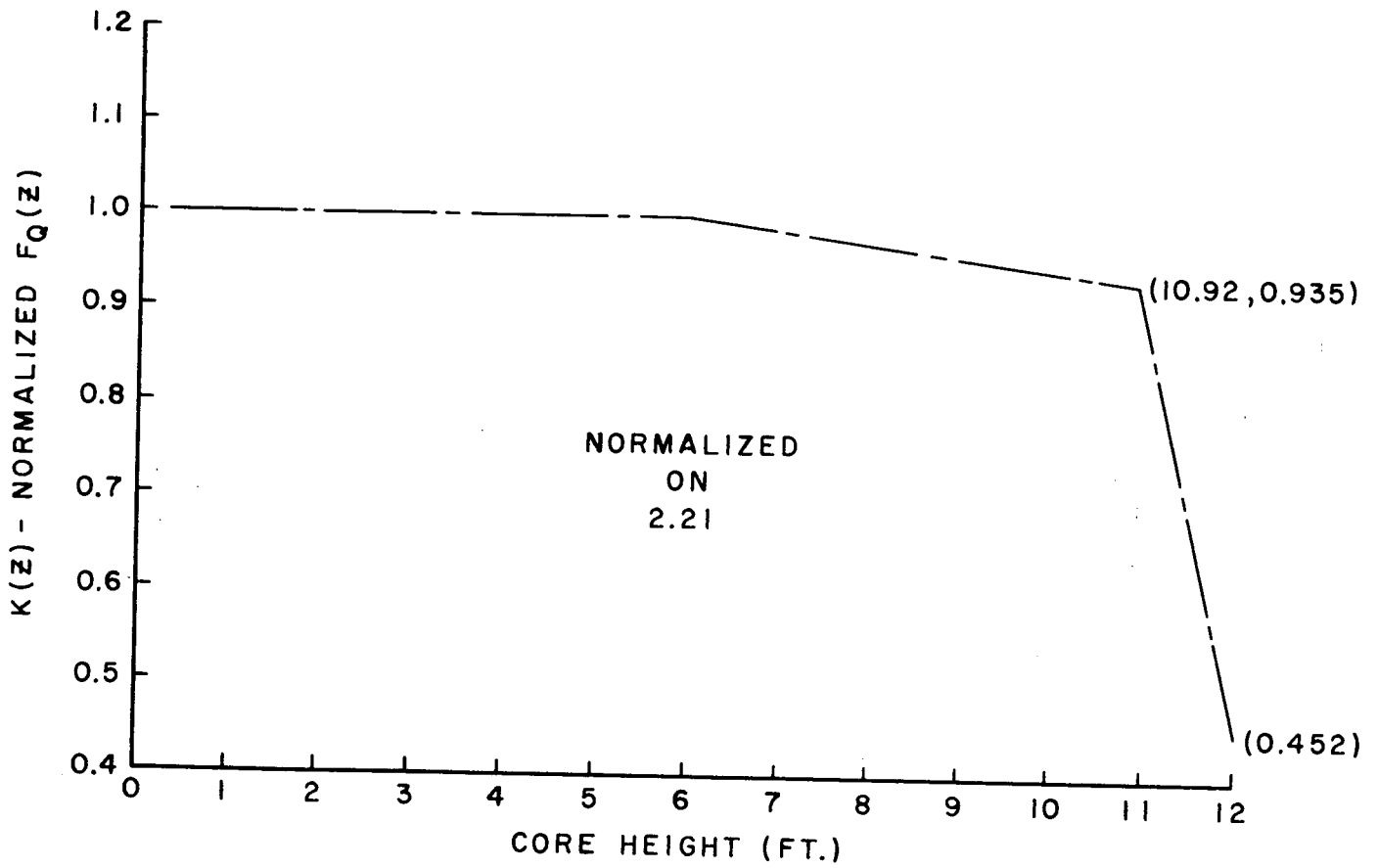
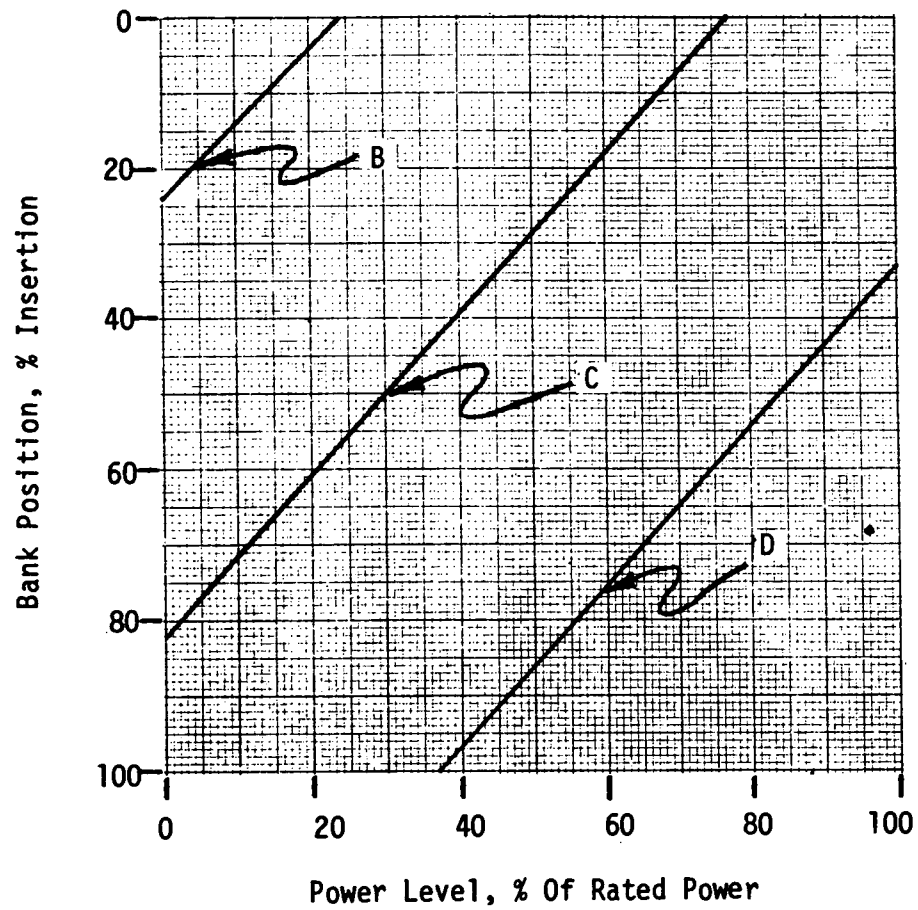


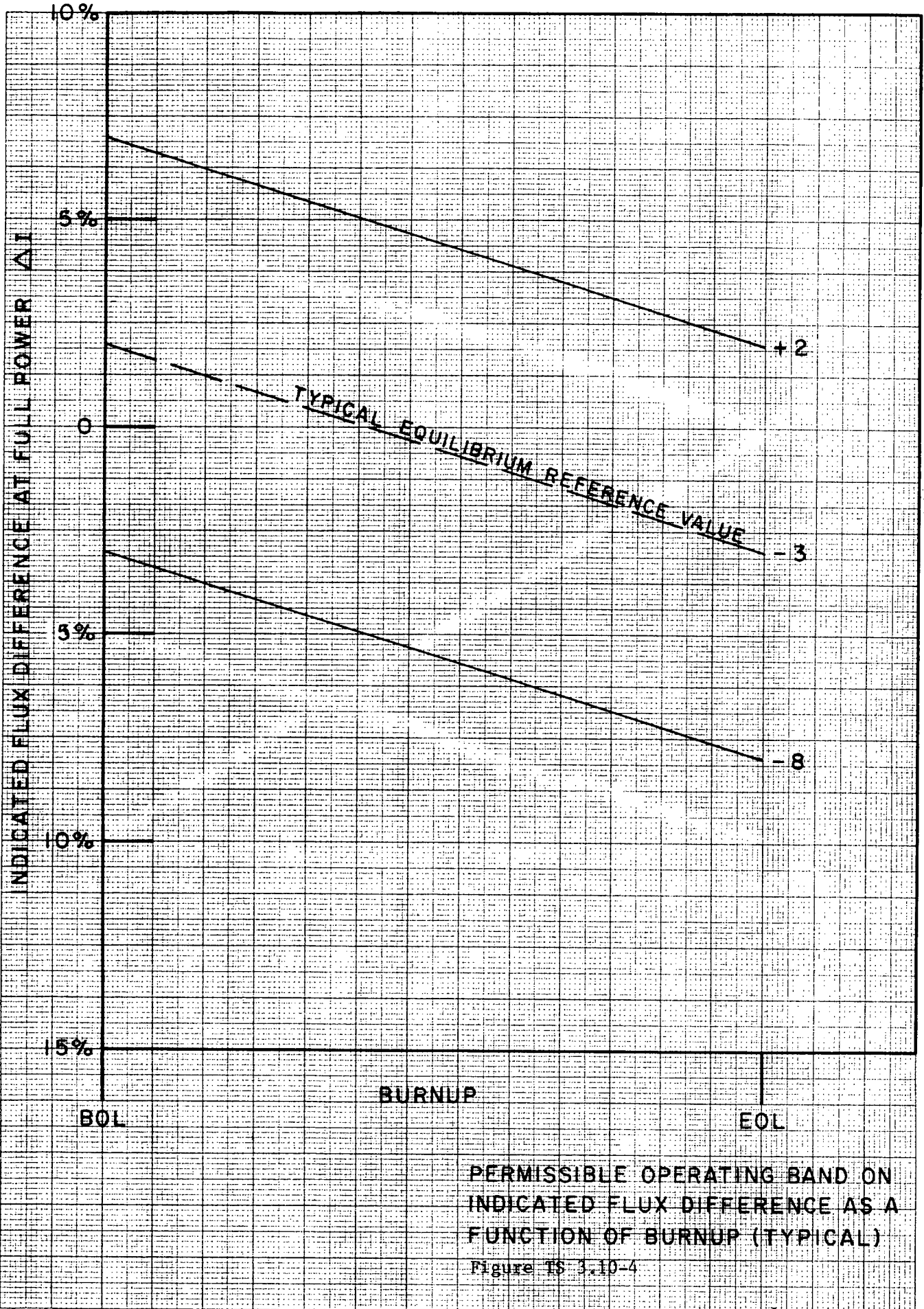
FIGURE TS 3.10-2 HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

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CONTROL BANK INSERTION LIMITS

FIGURE TS3.10-3



PERCENT OF RATED
THERMAL POWER

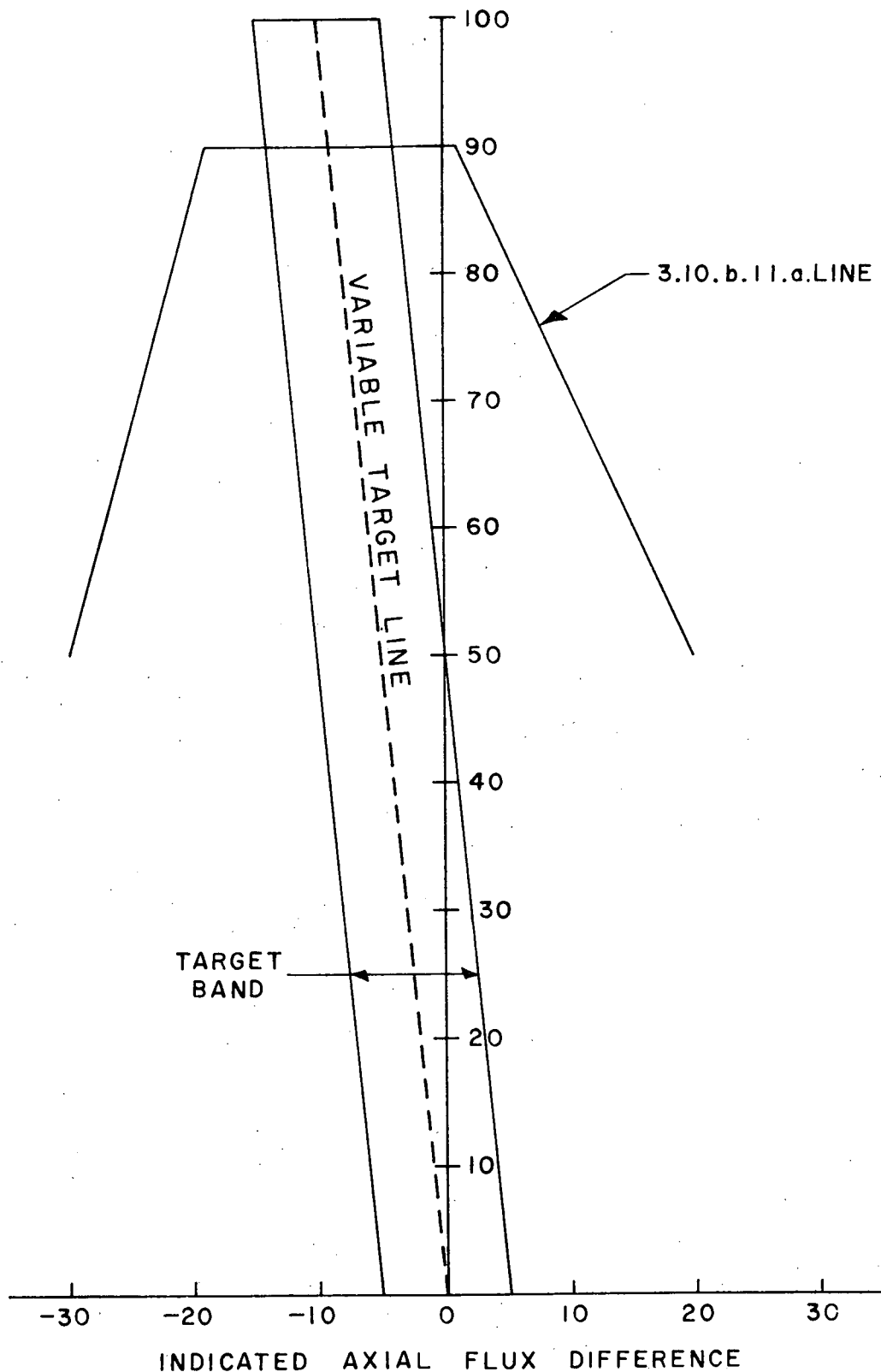
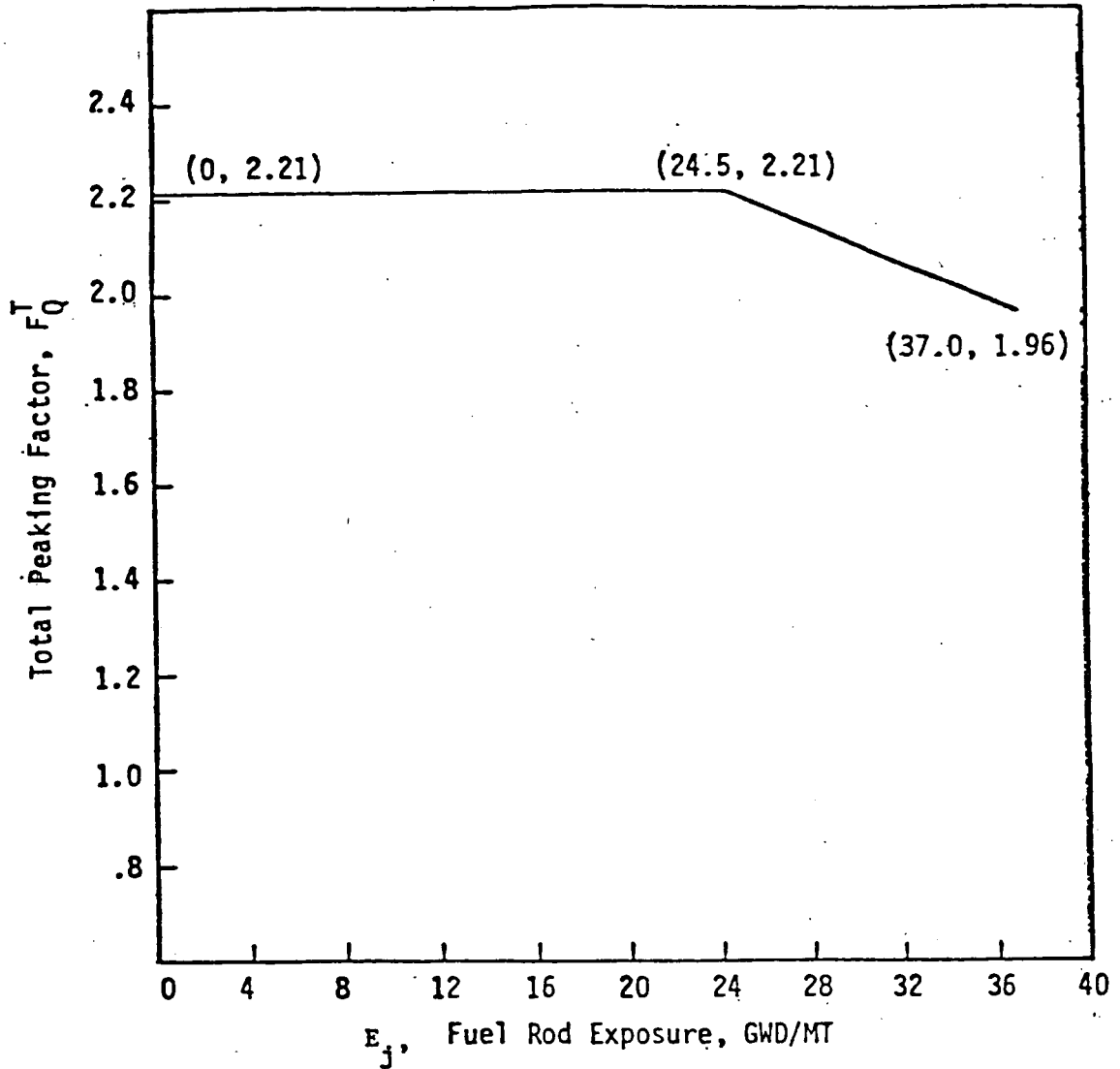


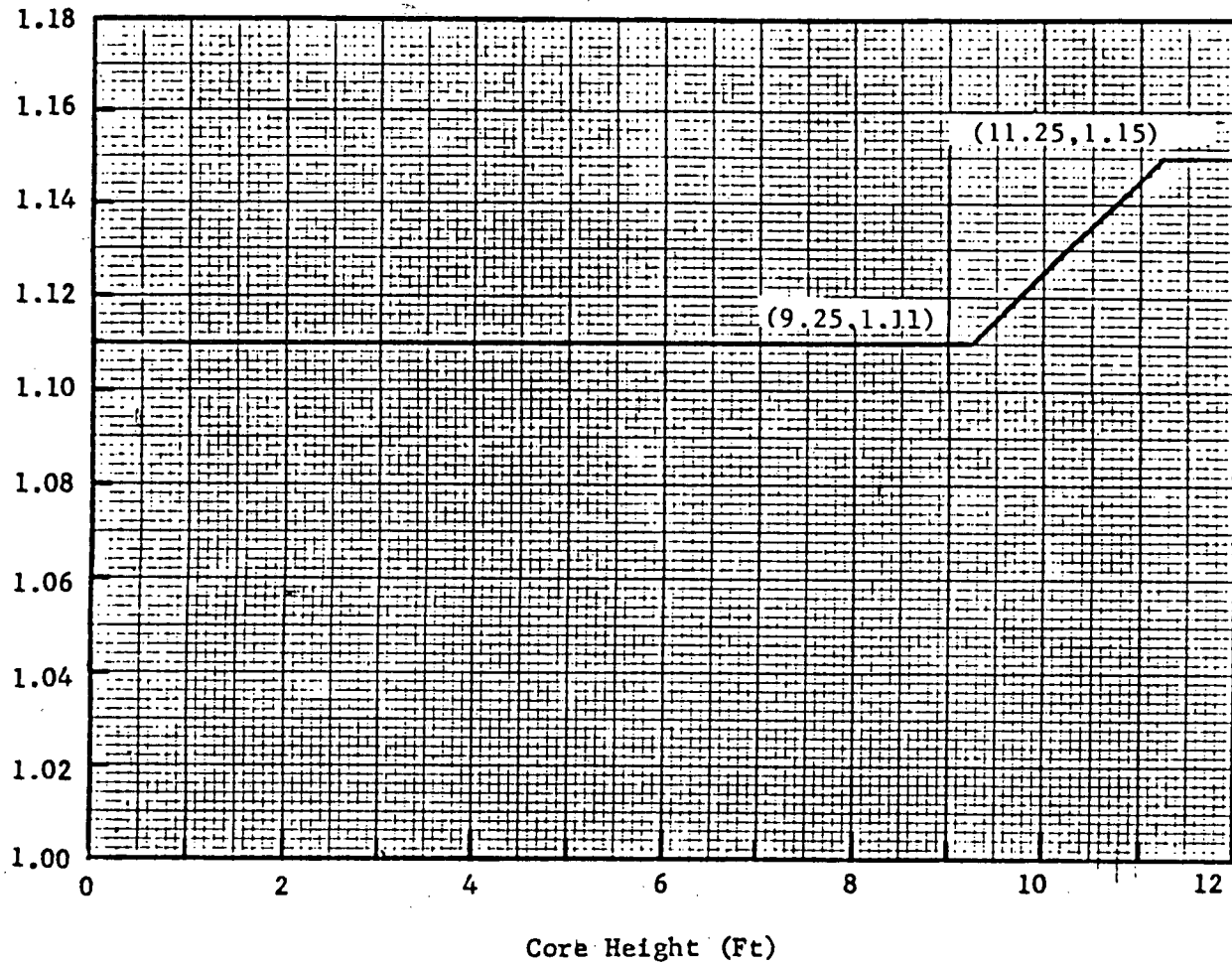
FIGURE TS 3.10-5 TARGET BAND ON
INDICATED FLUX DIFFERENCE AS A
FUNCTION OF OPERATING POWER
LEVEL (TYPICAL)



F_Q^T versus Rod Exposure: $F_Q^T(E_j)$
 (Reference specification 3.10.b.1.a.(ii))

Figure TS 3.10-6

V(Z)



V(Z) as a Function of Core Height

Figure 3.10-7

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5. Safeguard Bus Undervoltage and Safeguard Bus Second Level Undervoltage relays shall be calibrated at least once per refueling cycle (not to exceed 18 months).
6. During each refueling, a checkout of emergency lighting will be performed.

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b. Station Batteries

1. The voltage of each cell shall be measured to the nearest hundredth volt each month. An equalizing charge shall be applied if the lowest cell in the battery falls below 2.13 volts. The temperature and specific gravity of a pilot cell in each battery shall be measured.
2. The following additional measurements shall be made every three months: the specific gravity and height of electrolyte in every cell and the temperature of every fifth cell.
3. All measurements shall be recorded and compared with previous data to detect signs of deterioration.
4. The batteries shall be subjected to a load test during the first refueling and once every five years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

Basis

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and carry rated load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be indicated by an alarm without need for test startup.

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power sources and equipment are operable.

Specification

The following tests and surveillance shall be performed:

a. Diesel Generators

1. Manually-initiated start of each diesel generator, and assumption of load by the diesel generator. This test shall be conducted monthly in accordance with the intent of Paragraph 6.4.1 and 6.4.3 of IEEE 387-1977. | 33
2. Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, all initiated by a simulated loss of all normal a-c station service power supplies together with a simulated safety injection signal. This test will be conducted at each refueling interval to assure that each diesel generator will start and assume required loads to the extent possible within one minute, and operate for \geq 5 minutes while loaded with the emergency loads. | 42
3. Each diesel generator shall be inspected at each major refueling outage.
4. Diesel generator load rejection test in accordance with IEEE 387-1977, section 6.4.5 shall be performed at least once per 18 months. | 33

The less frequent overall system test will demonstrate that the emergency power system and the control system for the engineered safety features equipment will function automatically in the event of loss of all other sources of a-c power, and that the diesel generators will start automatically in the event of a loss-of-coolant accident. This test will demonstrate proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment, to the extent possible, as well as the operability of the diesel generators. A separate test will demonstrate that the emergency lighting⁴² system functions properly.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency will be detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function. Its possible failure to respond is, of course, anticipated by providing two diesel generators, each supplying through an independent bus, a complete and adequate set of engineered safety features equipment. Further, both diesel generators are provided as backup to multiple sources of external power, and this multiplicity of sources should be considered with regard to adequacy of test frequency.

Each diesel generator can start and be ready to accept full load within 10 seconds, and will sequentially start and supply the power requirements for one complete set of engineered safety features equipment in approximately one minute. (1)

Reference:

(1) FSAR Section 8.2

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Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide indication of a cell becoming unserviceable long before it fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

c) Verifying that each high pressure pump auto-start setpoint is ≥ 100 psig.

5. Deleted

c. Spray/Sprinkler Systems

Each of the spray and/or sprinkler systems in Specification 3.15.c shall be demonstrated OPERABLE:

1. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.
2. At least once per 18 months:
 - a) By performing a system functional test which includes simulated automatic actuation of the system, and:
 1. Verifying that the automatic valves in the flow path actuate to their correct positions, and
 2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.
 - b) By visual inspection of the spray headers to verify their integrity, and
 - c) By visual inspection of each nozzle to verify no blockage.
3. At least once per three years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

d. Low Pressure CO₂ Systems

Each of the low pressure CO₂ systems in Specification 3.15.d shall be demonstrated OPERABLE:

1. At least once per 7 days by verifying CO₂ storage tank level and pressure, and
2. At least once per 18 months by verifying:
 - a) The system valves and associated ventilation dampers actuate manually and automatically, upon receipt of a simulated actuation signal, and

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS
(Page 1 of 4)

<u>Channel Description</u>	<u>Check</u>	<u>Calibrate</u>	<u>Test</u>	<u>Remarks</u>
1. Nuclear Power Range	S (1) EFPM (3)****	D (1) EFPQ (3)****	(M) (2)***	1) Heat balance 2) Signal to ΔT ; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial off-set using in-core detectors
2. Nuclear Intermediate Range	*S (1)	N.A.	P (2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trips)
3. Nuclear Source Range	*S (1)	N.A.	P (2)	1) Once/shift when in service 2) Bistable action (alarm, trips)
4. Reactor Coolant Temperature	*S	R	M (1) M (2)	1) Overtemperature ΔT 2) Overpower ΔT
5. Reactor Coolant Flow	S	R**	M	
6. Pressurizer Water Level	S	R**	M	
7. Pressurizer Pressure	S	R**	M	
8A. 4-KV Voltage & Frequency	N.A.	R	M	Reactor protection circuits only
8B. 4-KV Voltage (Loss of Voltage)	N.A.	R	M	Safeguards buses only
8C. 4-KV Voltage (Degraded Grid)	N.A.	R	R	Safeguards buses only

Table TS 4.1-1 (Page 1 of 4)

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TABLE TS 4.10-1 (Page 1 of 6)

Operational Environmental Radiological Surveillance Program

Type of Sample	Location	Sampling Frequency	Type of Analysis	Frequency of Analysis	Reporting Units	Approximate Minimum Detectable Level	Comments
A. Airborne Particulates	K-1f	Weekly	Gross alpha	Weekly	pCi/m ³	4 x 10 ⁻⁴ pCi/m ³ 1 x 10 ⁻³ pCi/m ³	On all samples On all samples Quarterly composite for each station
	K-2		Gross beta	Weekly	pCi/m ³		
	K-7		Gamma Scan	Quarterly	pCi/m ³		
	K-8						
	K-15						
	K-16						
B. Airborne Iodine	Same as A	Bi-weekly	I-131	Bi-weekly	pCi/m ³	1 x 10 ⁻² pCi/m ³	On all samples
C. Ambient ; Beta-Gamma	K-1f	Quarterly Annually	Beta-Gamma	Quarterly	mrems/Q	10 mrem	On all samples On all samples
	K-2		Beta-Gamma	Annually	mrems/A		
	K-3						
	K-4						
	K-5						
	K-6						
	K-7						
	K-8						
	K-15						
	K-16						
TLD (5 chips in each packet)							

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TABLE TS 4.10-1 (Page 5 of 6)

Operational Environmental Radiological Surveillance Program

Type of Sample	Location	Sampling Frequency	Type of Analysis	Frequency of Analysis	Reporting Units	Approximate Minimum Detectable Level	Comments	
M. Bottom Sediments	500' North of discharge (on the beach)	4 times per year	Gross alpha	4/year	pCi/g	Same as Soil	May, July, Sept., Nov.	
	K-1d		Gross beta	4/year	pCi/g		April or May, June August, and October	
	500' South of discharge (on the beach)		Sr-89	4/year	pCi/g		April or May, June, August, and October	
	K-9 K-14		Sr-90	4/year	pCi/g		April or May, June August, and October	
N. Deleted								
O. Periphyton (Slime) and Aquatic Plants	K-1a	Semi-annually	Gross alpha	Semi-annually	pCi/g	0.11 pCi/g wet wt.	2nd and 3rd quarters if available in sufficient quantity	
	K-1b		Gross beta	Semi-annually	pCi/g		0.1 pCi/g wet wt.	2nd and 3rd quarters if available in sufficient quantity
	K-1d		Sr-89	Semi-annually	pCi/g		0.01 pCi/g wet wt.	2nd and 3rd quarters if available in sufficient quantity
	K-1e K-9 K-14		Sr-90	Semi-annually	pCi/g		0.007 pCi/g wet wt.	2nd and 3rd quarters if available in sufficient quantity

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steady state conditions greater than or equal to $1\% \Delta K/K$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 50¢; or occurrence of any unplanned criticality.

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- (5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.
- (6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

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Note: For items 6.9.2.a(5) and 6.9.2.a(6) reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 6.9.2.b(2) and 6.9.2.b(3) below.

- (7) Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.
- (8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications

6.10.2 The following records shall be retained for the duration of the Plant Operating License.

- a. Records of a complete set of as-built drawings for the plant as originally licensed and all print changes showing modifications made to the plant.
- b. Records of new and spent fuel inventory, fuel transfers, and assembly burnup histories.
- c. Records of plant radiation and contamination surveys.
- d. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.
- e. Records of radioactivity in liquid and gaseous wastes released to the environment.
- f. Records of transient or operational cycles for these facility components.
- g. Records of training and qualification for current members of the plant staff.
- h. Records of in-service inspections performed pursuant to these Technical Specifications.
- i. Records of meetings of the NSRAC and PORC.
- j. Records for Environmental Qualification which are covered under the provisions of paragraph 6.14.

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6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

AND BASES

APPENDIX B

<u>Section</u>	<u>Title</u>	<u>Page ES</u>
1.0	Definitions	Deleted
2.0	Environmental Protection Condition	Deleted
2.2	Chemicals	Deleted
2.2.1	Chlorination of Circulating Water Systems	Deleted
2.2.2	Suspended and Dissolved Solids	Deleted
2.2.3	Treatment Chemicals	Deleted
3.0	Monitoring Requirements	Deleted
3.2	Chemicals	Deleted
3.2.1	Chlorination of Circulating Water System	Deleted
3.2.2	Suspended and Dissolved Solids	Deleted
3.2.3	Treatment Chemicals	Deleted
4.0	Environmental Surveillance & Special Studies	Deleted
4.1	Biological	Deleted
4.1.1	Aquatic	Deleted
4.1.2	De-icing Operation	Deleted
5.0	Administrative Controls	
5.1	Organization, Review and Audit	Deleted
5.1.a	Organization	Deleted
5.1.b	Review and Audit	Deleted
5.2	Actions to be Taken in the Event of Violation of the Environmental Technical Specifications	Deleted Deleted
5.3	Operating Procedures	Deleted
5.4	Plant Reporting Requirements	Deleted
5.4.a	Annual Environmental Operating Report	Deleted
5.4.b	Reporting Requirement - 24 Hours and Subsequent Two-Week Followup Report	Deleted Deleted
5.4.c	Reporting Requirement - 30 Days	Deleted
5.4.d	Changes to the Plant or Procedures	Deleted
5.4.e	General Reporting Requirements	Deleted
5.5	Record Retention	
5.5.a	Record Retention - 5 Years	Deleted
5.5.b	Record Rentention - Life of Plant	Deleted

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