3. Pressurizer Safety Valves

   A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power operated relief valves are set for test pressure plus 35 psi and the charging pump has a safety valve to protect the system.

   B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressure Isolation Valves

   A. The pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device when the reactor is in the operating, hot standby or hot shutdown mode of operation. The valves shall be considered operational if the valve leakage as measured from the most recent leakage test is less than the allowable amount indicated in Table TS 3.1-2. 46

Basis

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 1 of the specification requires that both reactor coolant pumps be operating when the reactor is in power operation to provide core cooling in the event that a loss of flow occurs. Planned power operation with one loop out of service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this mode of operation. The flow provided in each case in Part 1 will keep DNBR well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted for any length of time except for tests. Upon loss of one pump below 10% full power the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation will remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at set point. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against over-pressurization.

The Reactor Safety Study (WASH-1400) has identified the potential for an intersystem LOCA which is a significant contributor to the risk of a core melt accident (Event V).

The scenario identified by WASH-1400 describes the failure of in-series check valves or the failure of a check valve in series with a normally open motor operated isolation valve. These failures would result in the over-pressurization of a low pressure system by the Reactor Coolant, causing an intersystem LOCA which bypasses containment. 46

The risk associated with this scenario will be reduced by periodically verifying that valves configured as described by WASH-1400 are performing within acceptable limits.

References:

(1) FSAR Section 7.2.2

TABLE T. S. 3.1-2
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum (a) (b) Allowable Leakage
Reactor Vessel, Core Flooding Line (Upper Plenum Injection)	SI-304A	5.0 Gallons per Minute
	SI-303A	5.0 Gallons per Minute
	SI-304B	5.0 Gallons per Minute
	SI-303B	5.0 Gallons per Minute
Loop B 12" Accumulator Discharge Line	SI-22B	5.0 Gallons per Minute

FOOTNOTES:

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 5.0 gpm are considered unacceptable.

3. Leakage rates greater than 1.0 gpm, but less than or equal to 5.0 gpm, are considered acceptable if: 46

$Lm \leq (.5)(5-Lp) + Lp$

Lm is the latest measured leakage rate in gallons per minute

Lp is the previous measured leakage rate in gallons per minute

4. Leakage rates greater than 1.0 gpm, but less than or equal to 5.0 gpm, are considered unacceptable if: $Lm > (.5)(5-Lp) + Lp$

(b) Minimum test differential pressure shall not be less than 150 psid.

b. HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

Specification

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2 for the service period up to 10 equivalent fullpower years. 46

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated using the methods discussed above. The derivation of the limit curves is consistent with NRC Regulatory Standard Review Plan Directorate of Licensing, Section 5.3.2 "Pressure-Temperature Limits" 1974 in Reference (1).

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. Weld metal Charpy test specimens for Capsule R indicate that the core region weld metal exhibits the largest shift in RT_{NDT} ($235^{o}F$). 46
The capsule experienced equivalent dose of 10 effective fullpower years, as presented in WCAP 9878.

The results of Irradiation Capsules V and R analyses are presented in WCAP 8908 and WCAP 9878, respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 10 effective fullpower years. 46 46

Pressure Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of $100^{\circ}F/hr$ cannot be achieved with the installed equipment.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1972 Addenda, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

2. Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-70, 1975 Book of ASTM Standards, Part 45, pp. 756-763.

3. W. S. Hazelton, S. L. Anderson, and S. E. Yanichko, "Basis for Heatup and Cooldown Limit Curves," WCAP 7924, July 1972.

4. S. E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

5. S. E. Yanichko, et al, "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March, 1981. 46

6. Letter from P. S. VanTeslaar (Westinghouse) to C. W. Giesler (WPS) dated April 30, 1981, transmitting KNPP Heatup and Cooldown curves based on Capsule R results.

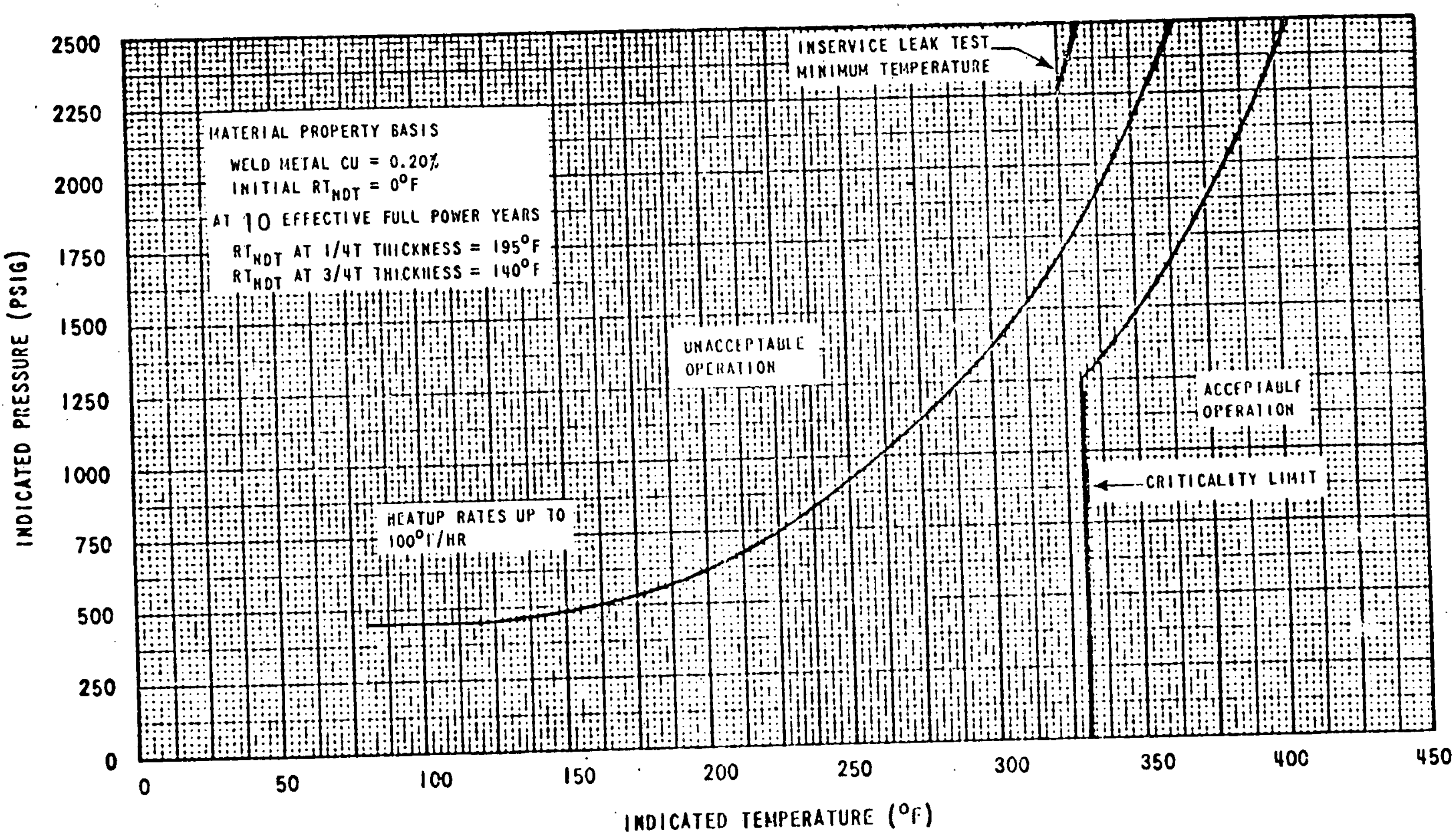

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Figure TS 3.1-1 Kewaunee Reactor Coolant System Heatup Limitations Applicable for Periods up to 10 Effective Full Power Years. Margins of 60 PSIG and 10°F are Include for Possible Instrument Error

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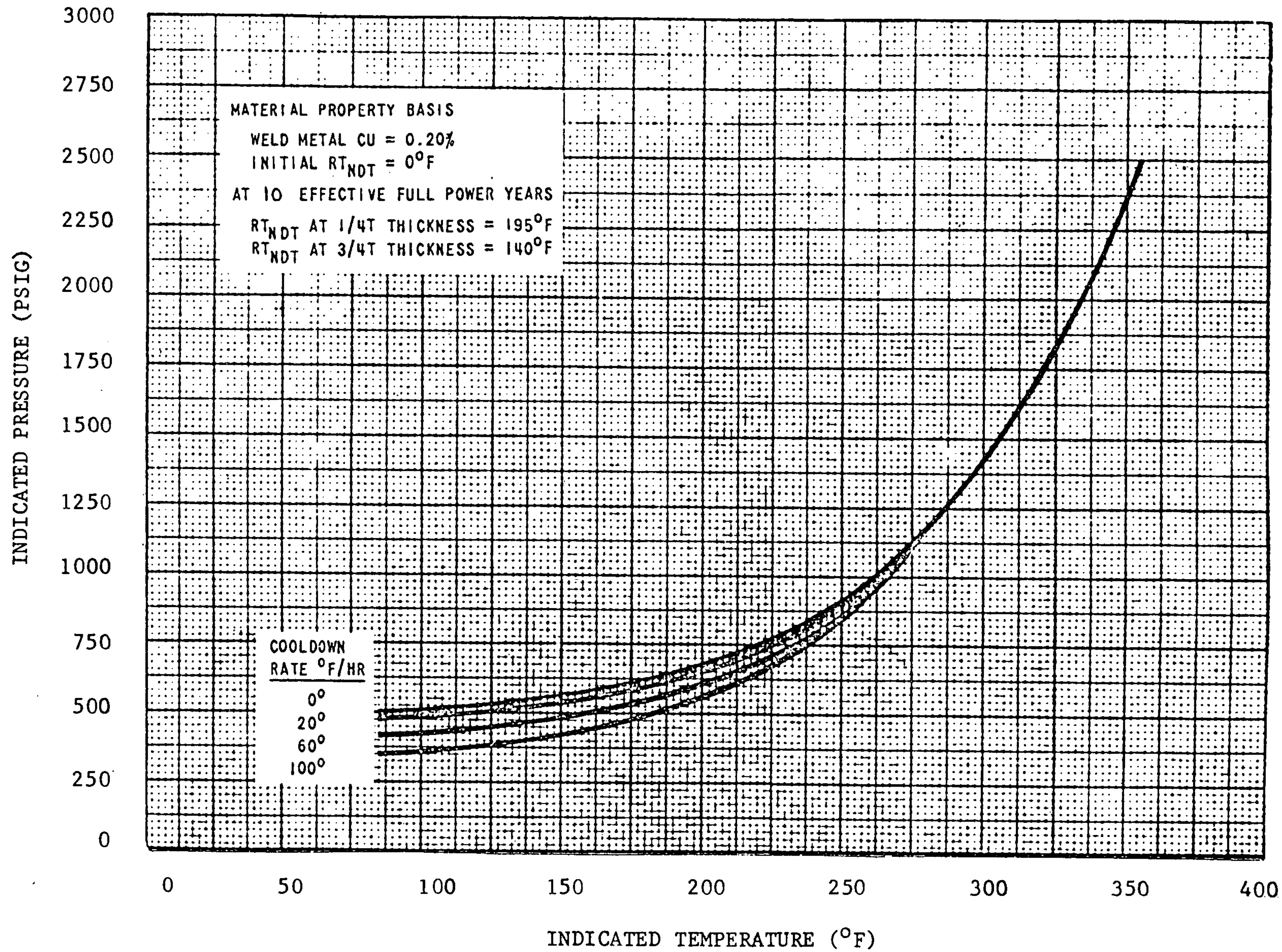

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Figure TS 3.1-2 Kewaunee Reactor Coolant System Cooldown Limitations Applicable for Periods up to 10 Effective Full Power Years. Margins of 60 PSIG and 10°F are Included for Possible Instrument Error. 46

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

Specification

a. The reactor shall not be heated above 350°F unless the following conditions are satisfied.

   1. Rated relief capacity of TEN steam system safety valves is available, except during testing.

   2. Three auxiliary feedwater pumps are operable. 46

   3. System piping and valves directly associated with the above components are operable.

   4. A minimum of 75,000 gallons of water is available in the condensate storage tanks and the Service Water System is capable of delivering an unlimited supply from Lake Michigan.

   5. The iodine-131 activity on the secondary side of the steam generators does not exceed 1.0 μCi/cc.

b. If, when the reactor is above 350°F, any of the conditions of Specification 3.4.a cannot be met within 48 hours, and except for the conditions of 3.4.C , the reactor shall be shutdown and cooled below 350°F using normal operating procedures. 46

c. When the reactor is above 350^{o}F, one auxiliary feedwater pump may be out of service provided the pump is restored to operable status within 72 hours, or the reactor shall be shutdown and cooled below 350^{o}F using normal operating procedures. 46

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. $F_Q(Z)$ Limits

      (i) Westinghouse Electric Corporation Fuel

      $F_Q(Z) \leq (2.22/P) \times K(Z)$ for $P > .5$ 46

      $F_Q(Z) \leq (4.44) \times K(Z)$ for $P \leq .5$

      (ii) Exxon Nuclear Company Fuel

      $F_Q(Z) \leq F_Q^T(Ej) \times K(Z)$ for $P > .5$

      $F_Q(Z) \leq (4.42) \times K(Z)$ for $P \leq .5$

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 F_Q

F_Q^T (Ej) is the function given in Figure TS 3.10-7

E_j is the fuel rod exposure for which F_Q is measured

b. $F_{\Delta H}^N$ Limits

(i) Westinghouse Electric Corporation Fuel

$F_{\Delta H}^N \leq 1.55 \; [1 + 0.2(1-P)]$ For 0 to 24,000 MWD/MTU burnup fuel

$F_{\Delta H}^N \leq 1.52 \; [1 + 0.2(1-P)]$ For greater than 24,000 MWD/MTU fuel

(ii) Exxon Nuclear Company Fuel

$F_{\Delta H}^N \leq 1.55 \; [1 + 0.2(1-P)]$

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

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2. If either measured hot channel factor exceeds the values specified in 3.10.b.1, the reactor power shall be reduced so as not to exceed a fraction of the design value equal to the ratio of the F_Q^N or $F_{\Delta H}^N$ limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. 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. For the purpose of this confirmation:

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

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

4. The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation using the most recent measured value and the value predicted for the end of the cycle life.

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

1. If a full length or part length rod cluster control assembly is misaligned from its bank by more than 22.5 inches, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and specification 3.10.b applied. If peaking factors are not determined within 2 hours, the reactor power shall be reduced to 85 percent of rating. 46

2. And, in addition to 3.10.e.1 above, if the misaligned rod cluster control 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 full length or part length rod 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.

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.

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direct control over $F^N_{\Delta H}$ 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 by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance.

The $F^N_{\Delta H}$ limits of specification 3.10.b.1.b include consideration of fuel rod bow effects for fuel fabricated by Westinghouse Electric Corporation. Since the effects of rod bow are dependent on fuel burnup an additional penalty is incorporated in a decrease in the $F^N_{\Delta H}$ limit of 2% for -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 $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ 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 536.5°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 $F^N_{\Delta H}$ for high burnup fuel, 24000 MWD/MTU. 46

There are no rod-bow penalties associated with fuel fabricated by Exxon Nuclear Co. 46

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 or 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 identify operational anomalies which would, otherwise, affect these bases.

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 22.5 inches from the bank demand position. 46
2. Control rod banks are sequenced with overlapping banks as shown in Figure TS 3.10-4.
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.

The permitted relaxation in F^N_H allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

The $F_Q(Z)$ limits of specification 3.10.b.1.a include consideration of off-gassing effects in fuel manufactured by Exxon Nuclear Company. References 7 and 8 discuss this phenomena. Since the fission gas release becomes enhanced as burn-up increases, 46

an additional penalty in the form of the function $BU(E_j)$, as shown in Figure TS 3.10-7, is applied to Exxon fuel. 46

In specification 3.10.b.1.a, F_Q is arbitrarily limited for $P < 0.5$ (except for low power physics tests).

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.

Conformance with specification 3.10.b.6 through 3.10.b.9 ensures the F_Q upper bound envelope is not exceeded and xenon distributions are not developed which at a later 46
time would cause greater local power peaking, even though the current flux difference is within the limits specified.

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 noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). 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 deviation of $\pm$ 5% ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure TS 3.10-6 shows a typical construction of the target.

The rod position indicator channel is sufficiently accurate to detect a rod $\pm$7-1/2 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.

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.

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

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

REFERENCES

(1) Section 4.3

(2) Section 4.4

(3) Section 14

(4) "Rod Misalignment Analysis," 7/27/81. Submitted to NRC with proposed Technical Specification Amendment 46, by letter from E. R. Mathews (WPSC) to D. G. Eisenhut, (USNRC) 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. 46

(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. 46

(8) Letter from L. C. O'Mally (Exxon Nuclear Company) to E. D. Novak (WPSC) providing F_Q exposure dependence as a function of rod burnup.

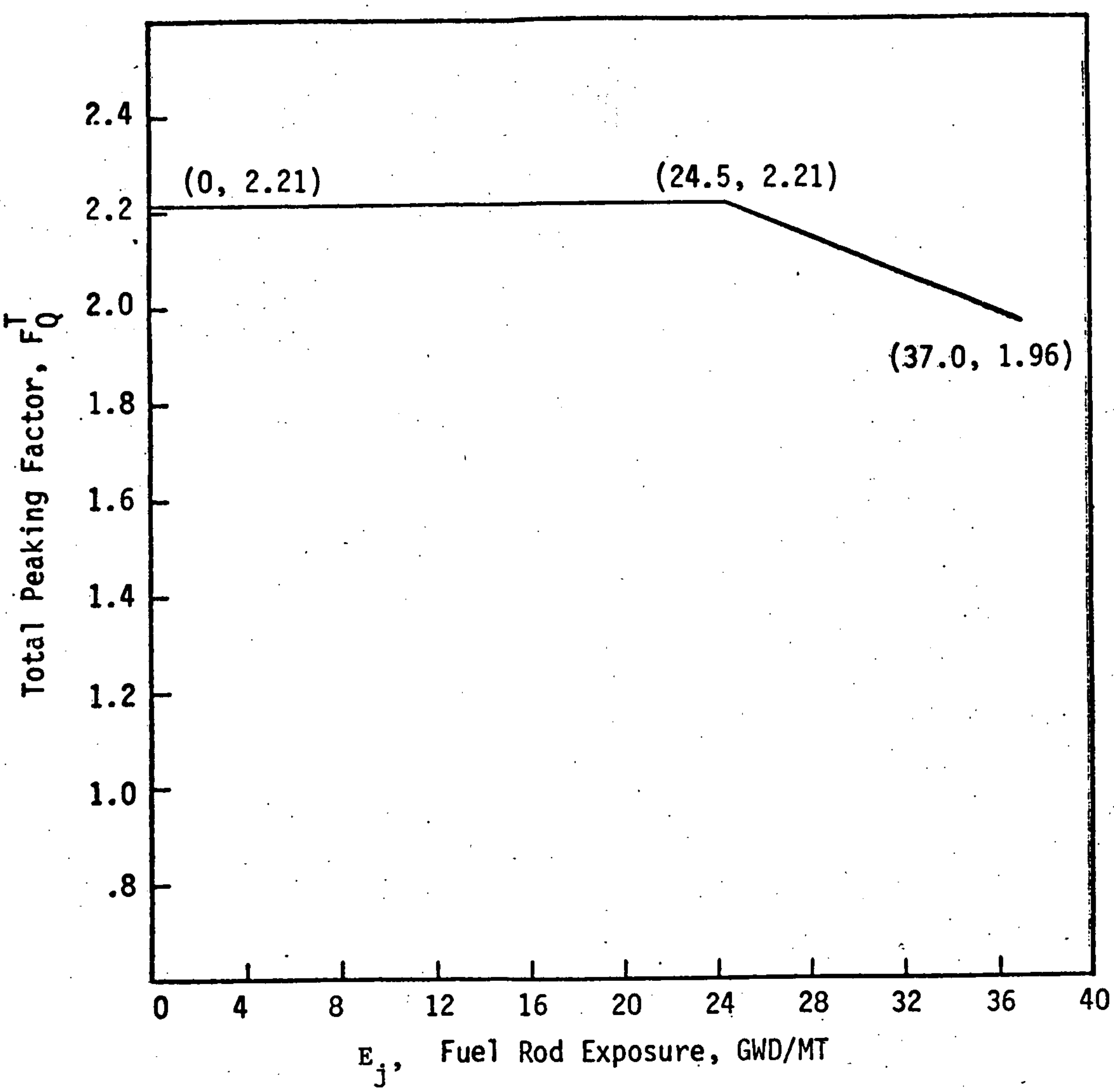

Figure TS 3.10-7

F_Q^T versus Rod Exposure: $F_Q^T(E_j)$

(Reference specification 3.10.b.1.a.(ii))

10. The following Surveillance Tests Shall be Accomplished:

a. Periodic leakage testing (1) on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the Hot Shutdown mode of operation when recovering from a refueling operation or after maintenance, repair or replacement work is performed on these valves. 46

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(1) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. 46

The exclusion criteria of IS-121 have been applied to determine which parts of systems or components are subject to surface or volumetric examinations and which parts are subject to a visual examination for evidence of leakage during the system hydrostatic test. A description of the system boundaries, delineating those parts subject to volumetric examination, those parts subject to surface examination and those parts requiring visual inspection during hydro are given in the notes to FSAR Table 4.4-2, titled Tables 4.4-2A, 4.4-2B and 4.4-2C.

The plant was not specifically designed to meet the requirements of Section XI of the code; therefore, 100 percent compliance may not be feasiable or practical. However, access for inservice inspection was considered during the design, and modifications have been made where practical to make provision for maximum access within the limits of the current plant design.

The Reactor Coolant System shall initially be free of gross defects, and the system has been designed such that gross faults or defects should not occur throughout the plant lifetime. The ten-year surveillance program will reveal possible fault areas before any leak develops, should such problems actually occur.

Basis for Reactor Coolant System Pressure Isolation Valves

Experience with check valves acting as a boundary between high pressure and low pressure piping has shown a very low probability of failure. Failure, when it occurs, happens during or as a result of transient conditions at the valve rather than at the static operating pressure. The surveillance frequency is sufficient to assure the required level of performance. 46

c. At least one licensed operator shall be in the control room when fuel is in the reactor.

d. At least two licensed operators shall be present in the control room during reactor startup, turbine generator synchronization to the grid, and during recovery from reactor trips.

e. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. This individual may be one of the shift operators.

f. Refueling operations shall be directed by a licensed Senior Reactor Operator assigned to the refueling operation who has no other concurrent responsibilities during the refueling operation.

g. A Fire Brigade of at least three members shall be maintained at all times. The Fire Brigade shall not include a minimum crew of two control operators necessary for safe shutdown of the unit during a fire emergency. This change is effective 90 days after issuance of this amendment.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 Qualifications of each member of the Plant Staff shall meet or exceed the minimum acceptable levels of ANSI-N18.1-1971 for comparable positions, except for the Health Physics Supervisor who shall meet or exceed the requirements of Regulatory Guide 1.8, Revision 1-R, September, 1975, or their equivalent. 46

6.4 TRAINING

6.4.1 A retraining and replacement training program for the Plant Staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI-N18.1-1971 and Appendix A of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Fire Marshall and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions shall be held quarterly.