Docket No. 50-213

B11911

Attachment No. 1

Proposed Technical Specification Changes
Haddam Neck Plant
Cycle 14

December, 1985

TECHNICAL SPECIFICATION SECTION NUMBERS	TITLE OF SECTION	PAGE NUMBER AND REVISION	
1.0	Definitions	1-5	Rev. 5
2.4	Maximum Safety Settings - Protective Instrumentation	2-5	Rev. 29
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3.17	Limiting Linear Heat Generation Rate	3-30	Rev. 58
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3.18	Power Distribution Monitoring and Control	3-32	Rev. 23
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		Fig. 3.18-1a	Rev. 58
		Fig. 3-18-1b	Rev. 52
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		Fig. 3-18-2a	New
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		Fig. 3-18-2c	New
3.20	Reactor Coolant System Flow, Temperature and Pressure	3-39	Rev. 15
		4-40	Rev. 15

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water return flow from the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 The QUADRANT POWER TILT RATIO shall be the ratio of the maximum quadrant power to the average quadrant power as determined by the excore detector outputs.

DOSE EQUIVALENT I-131

1.19 Not used

1.20 Not used

FREQUENCY NOTATION

1.21 The frequency notation specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

2.4 Maximum Safety Settings - Protective Instrumentation

Applicability: Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow.

Objective: To provide for protective action in the event that the principle process variables approach a safety limit.

Specification: Protective instrumentation trip settings shall be as follows:

		Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating
(1)	Pressurizer Pressure	$\leq$ 2300 psig	$\leq$ 2300 psig
(2)	Pressurizer Level*	$\leq$ 86% of range	$\leq$ 86% of range
(3)	Variable Low Pressure***	$\geq 17.4\ (T_{avg}+1.17\Delta T)-8850$	$\geq 17.4(T_{avg}+1.17\Delta T)-8850$
(4)	Nuclear Overpower**	$\leq$109% of rated power	$\leq$74% of rated power
(5)	Low Coolant Flow***	$\geq$90% of normal loop flow	$\geq$90% of normal loop flow
(6)	Reactor Coolant Loop Valve-Temperature Interlock	$\leq 20^{o}$F	$\leq 20^{o}$F
(7)	High Steam Flow	110% of full load steam flow	110% of full load steam flow

* May be bypassed when the reactor is at least 1.5%Δk subcritical.

** The nuclear overpower trip is based upon a symmetrical core power distribution. If any asymmetric power distribution should occur, resulting in the power in any quadrant being 2% greater than the average core power as indicated by the neutron detectors, within 2 hours either

a) reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) reduce thermal power at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Nuclear Overpower trip setpoint within the next 4 hours.

When the reactor power is <10% the overpower trip setpoint is reduced to 25% of rated power.

*** May be bypassed below 10% of rated power.

Basis: The reactor protective system is designed and constructed such that no single failure in any of the instrument systems will prevent the desired safety action if an applicable parameter exceeds a safety setpoint.

and shutdown. It is safe to block this trip below 10% power since the protection afforded by this trip is not required at this low level. Removal of unnecessary trip signals will reduce the number of spurious trips.

(4) Nuclear Overpower. As explained above, the nuclear overpower reactor trip, in conjunction with the variable low pressure reactor trip, provides overpower, overtemperature protection. The nuclear overpower trip channels will respond first to rapid reactivity insertion rates, detected by the increase in flux, before there are any significant changes in the system process variables. A maximum error of 9% of full power due to setpoint, instrumentation, and calorimetric determination (see Section 4.3.6 of the FDSA) is considered in establishing the setpoint. In order to reduce the time to trip for certain accidents occurring at low power, the overpower setpoint is lowered to 25% when reactor power is below 10%. This low overpower trip would terminate the postulated large steamline break accident from the hot zero power condition. The lower setting for three loop operation provides protection at the reduced power level equivalent to that provided by the setting for four loop operation at full power. The reduction in setting in the event of an asymmetric power distribution provides protection for the more adverse hot channel factors. The asymmetry is detected by observation of changes in neutron detector ion chamber current readings.

(5) Low Coolant Flow. The low coolant flow reactor trip protects the core against an increase in coolant temperature resulting from a reduction in coolant flow while the reactor is at substantial power[3]. This trip will prevent DNB in any loss-of-flow incident, which eliminates the possibility of clad damage. Flow detection in each reactor coolant loop is from a measurement of pressure drop from inlet to outlet of each steam generator. The 90% low flow signal is high enough to activate a trip in time to prevent DNB, and low enough to reflect that a loss-of-flow condition truly exists. A maximum instrument and setpoint error of 5% full flow is considered in determining the setpoint. Loss-of-flow protection is also provided by reactor coolant pump breaker and from undervoltage

3.16 ISOTHERMAL COEFFICIENT OF REACTIV'TY

Applicability: Applies to the isothermal coefficient of reactivity for the core.

Objective: To limit the maximum positive moderator coefficient that can exist in the reactor.

Specification: The isothermal temperature coefficient as measured at zero power shall be such that when corrected to operating conditions the value of the moderator temperature coefficient shall be:

1) Less positive than a calculated $+0.5 \times 10^{-4}$ delta k/k/°F for the all rods withdrawn, beginning of life (BOL), hot zero thermal power condition.

2) Less positive than a calculated 0.0 delta k/k/°F for the al! rods withdrawn, beginning of life (BOL), rated thermal power condition.

3) Less negative than a calculated -2.9×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), rated thermal power condition.

Basis: The transient analyses described in Section 10 of the Facility Description and Safety Analysis (FDSA) include a range of moderator temperature coefficients between -3.5×10^{-4} per °F and $+1.0 \times 10^{-4}$ per °F. The specified range of coefficients from -2.9×10^{-4} per °F to $+0.00 \times 10^{-4}$ per °F is bounded by these analyses.

Deleted

3.17 LIMITING LINEAR HEAT GENERATION RATE

Applicability: Applies to measured peak linear heat generation rate (Kw/ft) in the reactor core.

Objective: To establish limits on linear heat generation rate (Kw/ft) which are based on the postulated loss-of-coolant accident (LOCA) with (1) appropriate allowances for fuel densification and (2) a reanalysis of LOCA considering upper head fluid temperature equal to reactor vessel outlet temperature and (3) reanalysis considering an increased fuel pellet/clad gap and fuel rod pre-pressurization and (4) reanalysis considering coastdown and (5) reanalysis considering steam generator tube plugging.

Specification:

A. During steady state power operation, the peak linear heat rate values shall not exceed those limits shown below as defined in Reference (2) and modified by References (4)-(7).

1. Cycle residency time less than 3000 EFPH:

   14.3 Kw/ft

2. Cycle residency time greater than or equal to 3000 EFPH but less than 6000 EFPH:

   14.5 Kw/ft

3. Cycle residency time greater than or equal to 6000 EFPH but less than end of design life:*

   15.5 Kw/ft

B. During coastdown operation following end of design life* the peak linear heat rate values shall not exceed 13.0 Kw/ft.

C. Measured values of core power peaking factors used in determining measured linear heat generation rates in the Section A specification shall include allowances for the following:

1. Normal power peaking.

*For the purpose of this specification, the end of design life is defined as: minimum attainable PPM boron in the RCS, all control rods fully withdrawn, and T_{avg} at the normal hot full power.

2. Flux peaking augmentation factors (Power Spike), using Figure 3.17.1.

3. Measurement uncertainty (1.05).

4. Statistical density factor (1.012).

5. Engineering factor (1.02).

6. Stack shortening/thermal expansion factor (1.007).

7. Power level uncertainty (1.02).

These factors are multiplicative and Items 1 and 2 shall be chosen at a core height so as to maximize their product.

D. Three loop operation at above 65% of full licensed power shall not be permitted until additional analysis is performed and proposed technical specification changes submitted. During three loop operation, the (Kw/ft) limits of Specification 3.17.A shall be multiplied by 0.65.

Basis: Specification A sets limits that assure the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300°F limit specified in the Interim Acceptance Criteria (IAC) issued in June, 1971, considering the postulated effects of fuel densification.

These limits incorporate the results of a recalculation of the LOCA event considering upper head fluid temperature equal to the reactor vessel outlet temperature (T_{HOT}), and an increased pellet/clad gap and fuel rod pre-pressurization incorporated in Batch XIV and subsequent reloads.

The basis for Specification B is that the decrease in reactor inlet temperature during coastdown affects the system response during the postulated loss-of-coolant accident.

The basis for the multiplicative factors in Specification C are described in Section 8.1 of Reference (2). The engineering factor has been reevaluated based on current manufacturing tolerances. This factor includes the effects of variations on pellet diameter, density and enrichment and fuel rod diameter.

The basis for Specification D is that previous analysis (Ref. 3) submitted December 5, 1972, showed the acceptability of 75% of rated power for three loop operation. An additional 10% margin in rated power was added as a conservative allowance for fuel densification.

References:

(1) WCAP-8213, Effects of Fuel Densification on the Connecticut Yankee Reactor, October, 1973.

(2) Description and Safety, Including the Effects of Fuel Densification on the Connecticut Yankee Reactor, Cycle V, CYAPCO, November, 1973.

(3) Switzer, D. C. to Assistant Director for Operating Reactors, USAEC, CYAPCO letter of December 5, 1972.

(4) D. C. Switzer letter to A. Schwencer (USNRC), dated May 2, 1977.

(5) D. C. Switzer letter to A. Schwencer (USNRC), dated October 31, 1977.

(6) W. G. Counsil letter to D. M. Crutchfield (USNRC), dated December 14, 1982.

(7) W. G. Counsil letter to D. M. Crutchfield (USNRC), dated March 30, 1984.

3.18 POWER DISTRIBUTION MONITORING AND CONTROL

Applicability: Applies to monitoring of the reactor power distribution as a function of reactor power level and control rod position. This specification is based upon the control rod bank "B" consisting of eight (8) RRC's consisting of rod position numbers 10, 11, 12, 13, 30, 31, 32 and 33.

Objective: To assure operation within limits as specified in Technical Specification 3.17.

Specification: A. Full Core Power Distribution Measurements

1. Full core power distributions shall be measured at least once per full power month during normal power operation, using the movable incore neutron detector system.

2. Prior to commencement of full power operation for each new operating cycle, denoted by a core refueling, a full core power distribution measurement will be performed before exceeding 80% of rated power using the movable incore neutron detector system. These measurements will be adjusted to 100% of rated power and evaluated for compliance with Specification 3.17 limits prior to increasing power to 100% of rated power.

3. These measurements shall be repeated at 100% power equilibrium xenon conditions approximately 40 full power hours after initially reaching 100% power in each new operating cycle. These measurements will be evaluated in a time period not to exceed 5 days after satisfactory completion of the measurements. If measured values are found to exceed Specification 3.17 limits, core thermal power shall be reduced by the percent exceeded until compliance can be demonstrated.

B. Power Distribution Monitoring

1. Periodic surveillance of the core power distribution shall be performed above 40% of rated power as described below:

1.1 Determination of the axial offset by use of excore power range detectors.

1.1.a The incore axial offset shall be continuously monitored using at least two calibrated power range channels and verified, using incore/excore correlation to be within the power dependent envelope specified in Figure 3.18-1 or 3.18-2 with an appropriate allowance made for excore calibration uncertainty, which shall not be less than +3% in excore axial offset units.

1.1.b Should the axial offset be found to exceed the operating limit curves for 4 loop operation (Figures 3.18-1) or 3 loop operation (Figures 3.18-2), corrective action via control rods or power reduction shall be taken, and the axial offset returned to within the operating band, unless it can be verified by the Technical Specification 3.18.B.1.2 procedure that the Technical Specification 3.17 limits are not being violated.

1.1.c The excore-incore correlations shall be verified every full power month using the results from the full core power distribution map specified in 3.18.A.1. Every three full power months, the excore-incore axial offset correlation will be verified and adjusted based upon results from the movable incore neutron detector system. The excore-incore correlation shall be initially checked with each change in fuel configuration and after a major change in excore instrumentation using results of incore measurements specified in 3.18.A.2 and 3, and/or supplemental core power maps. The excore detectors will be calibrated/correlated within 7 days after the satisfactory completion of the incore measurements.

1.2 Incore detector measurements in two (2) thimble locations may be used to monitor the power distribution:

1.2.a An appropriate correlation specific to the two selected thimbles is applied to their measured axial power distributions so as to determine a total core peaking factor to assure compliance with Specification 3.17. The thimble location which yields the higher total core peaking factor shall be used for verification of the Specification 3.17 limits. If these limits are found to be exceeded, the core thermal power shall be reduced by the percent exceeded until compliance with the 3.17 limits can be demonstrated.

1.2.b The correlation factor applied to these thimble measurements shall be based upon previous full core power distribution measurements within the cycle with the movable incore detector system and the correlation shall be checked every full power month during normal power operation.

1.2.c The frequency of incore detector power distribution monitoring shall be once per 8-hour shift for the steady state operation, and at least once per hour for periods of non-steady state operation. For this specification, "steady state operation" shall be considered at those periods of operation for which the reactor has operated above 80% of rated power for a continuous period of 24 hours, with control rod bank "B" at a height no less than 270 steps. "Non-steady state operation" is defined as those periods of operation not complying with the above definition.

C. Control Rod Insertion Limits

1. Except for lower power physics tests at or below 10% of full power or determination of "just critical" rod positions, operation of the control group banks shall be maintained above the limits shown in Figure 3.10-1.

2. The monthly average position of control rod bank "B" shall be at least 280 steps withdrawn when above 20% of rated thermal power when weighted by the daily average thermal energy generation. This requirement shall be evaluated twice per month.

Bases:

A.1 Use of the movable incore neutron detectors provides an accurate means for determination of three dimensional power distributions when evaluated using the Westinghouse INCORE program.(1) Results of these measurements are used to check compliance with Technical Specification 3.17 limits.

A.2 Full core power distribution measurement at or below 80% of rated power for the startup of each new operating cycle provides verification of design predictions to assure compliance with core thermal limits before proceeding to 100% of rated power.

A.3 An additional full core power distribution measurement at 100% of rated power just after reaching equilibrium xenon conditions for each new operating cycle provides further verification of the acceptability of the power distribution.

B.1 Monitoring below 40% of rated power is considered unnecessary because of the substantial margin in local fuel rod heat flux at this reduced power.

B.1.1.a Provides continuous monitoring of the incore power distribution by means of the out-of-core power range detectors. Operation within the axial offset envelopes of Figure 3.18-1 and Figure 3.18-2 assures that the local heat flux will not exceed the peak linear heat rate limits defined in Specification 3.17. Specific axial offset versus power curves for future reloads will be calculated and checked against Figure 3.18-1 and Figure 3.18-2 to insure continued applicability.

B.1.1.b An appropriate allowance for incore/excore calibration uncertainty is used by the reactor operator. The bases for the excore detector calibration and its uncertainty are described in Appendix B to Reference 3.

B.1.1.c Monthly checks and calibrations every three full power months assure maintenance of the excore detector calibration. Provides for axial offset monitoring calibration after a new core loading or changes in nuclear instrumentation.

B.1.2 Use of two movable incore thimble measurements along with an appropriate correlation converting the measurement axial

peaking factors to F_Q^N provides an alternate means of verification of compliance with the Technical Specification 3.17 limits. For conservatism, the highest F_Q^N determined will be used. The correlation is based upon full core power distribution measurements and has allowances for measurement uncertainties and the spatial effects of control rod insertions. A frequency of once per 8-hour shift is considered adequate because core power distributions do not change substantially during this mode of operation. A frequency of at least once per hour during non-steady state operation is to monitor for control rod and xenon induced power peaking.

C.2 This specification limits the potential for unfavorable axial power distributions due to operation for long burnup intervals at deep control rod insertions.

References:

(1) WCAP-7149, Leggett, W. D. and Eisenhart, L. D., The INCORE Program, Westinghouse Electric Company, December, 1967.

(2) Technical Report Supporting Cycle VI Operation and Proposed License Amendments, Docket 50-213, May, 1975.

(3) "Axial Offset Monitoring Including Revised Control Rod Grouping for the Connecticut Yankee Reactor, Cycle V," Docket No. 50-213, August 1974.

(4) D. C. Switzer (CYAPCO) letter to Director of Nuclear Reactor Regulation (NRC), dated October 23, 1975.

FIGURE 3.18-1a
POWER vs OFFSET, 0-125 EFPD AND COAST DOWN, FOUR LOOP

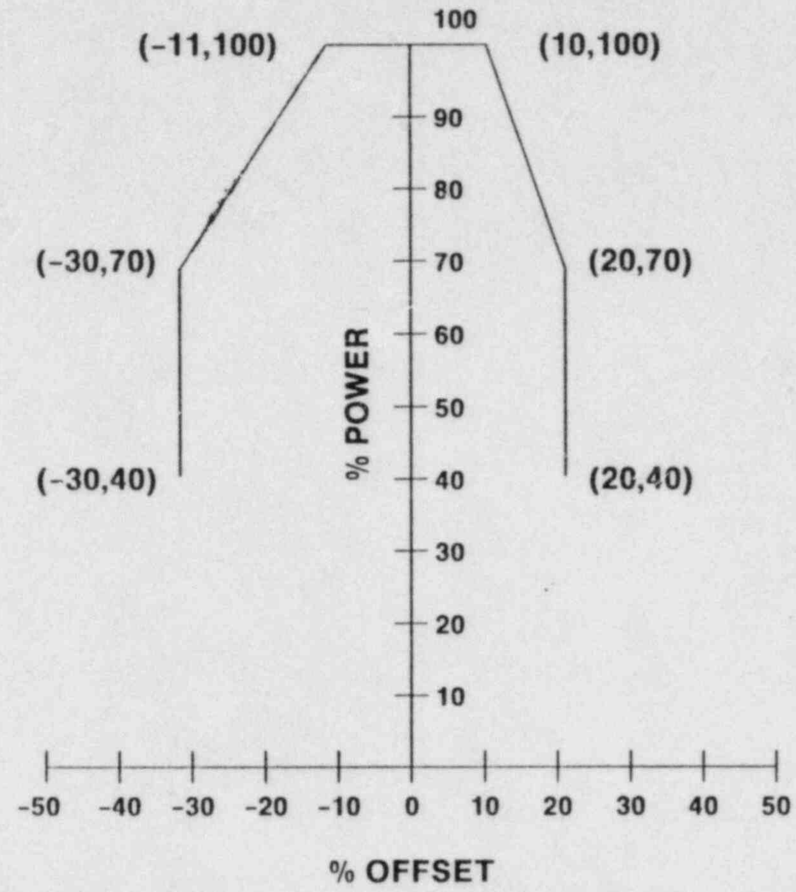


FIGURE 3.18-1b
POWER vs OFFSET, 125-250 EFPD, FOUR LOOP

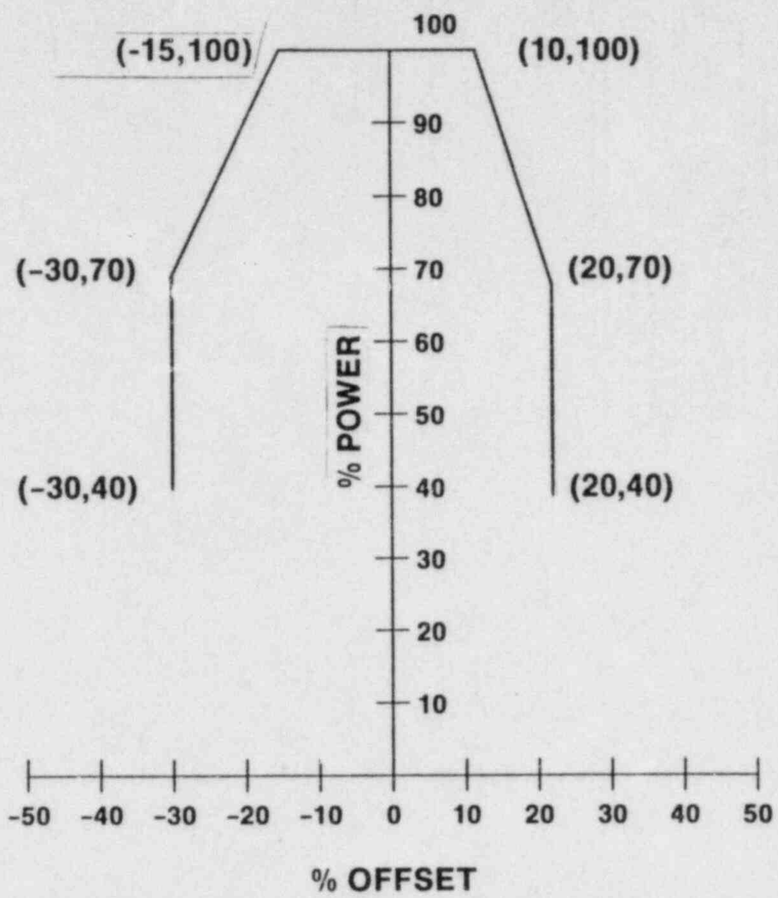


FIGURE 3.18-1c
POWER vs OFFSET, 250 EFPD-EOC, FOUR LOOP

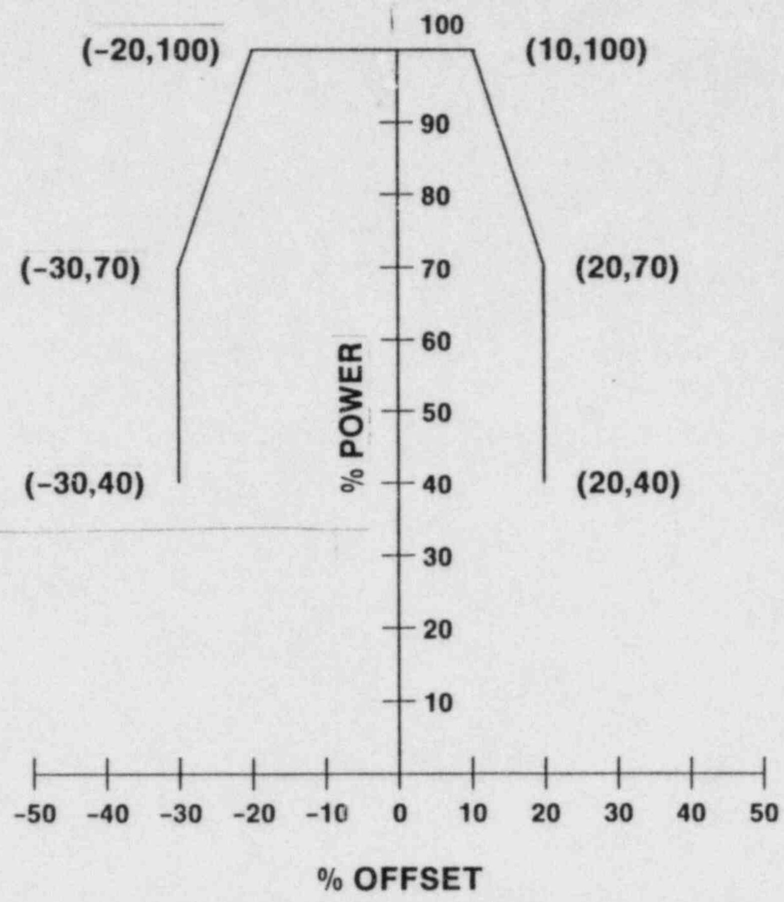


FIGURE 3.18-2a
POWER vs OFFSET, 0-125 EFPD AND COAST DOWN, THREE LOOP

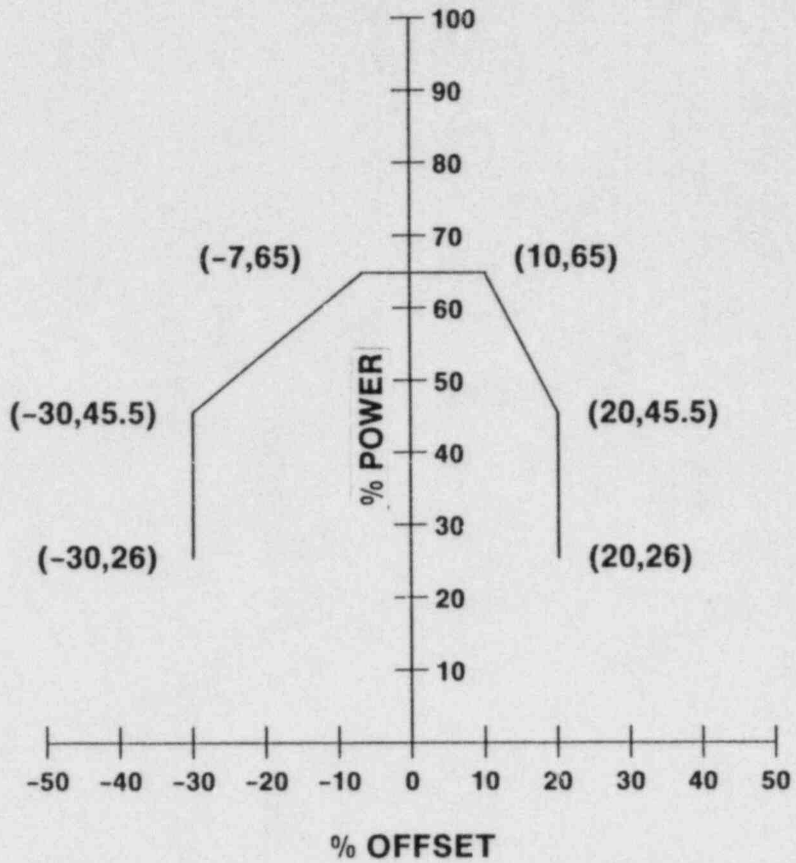


FIGURE 3.18-2b
POWER vs OFFSET, 125-250 EFPD, THREE LOOP

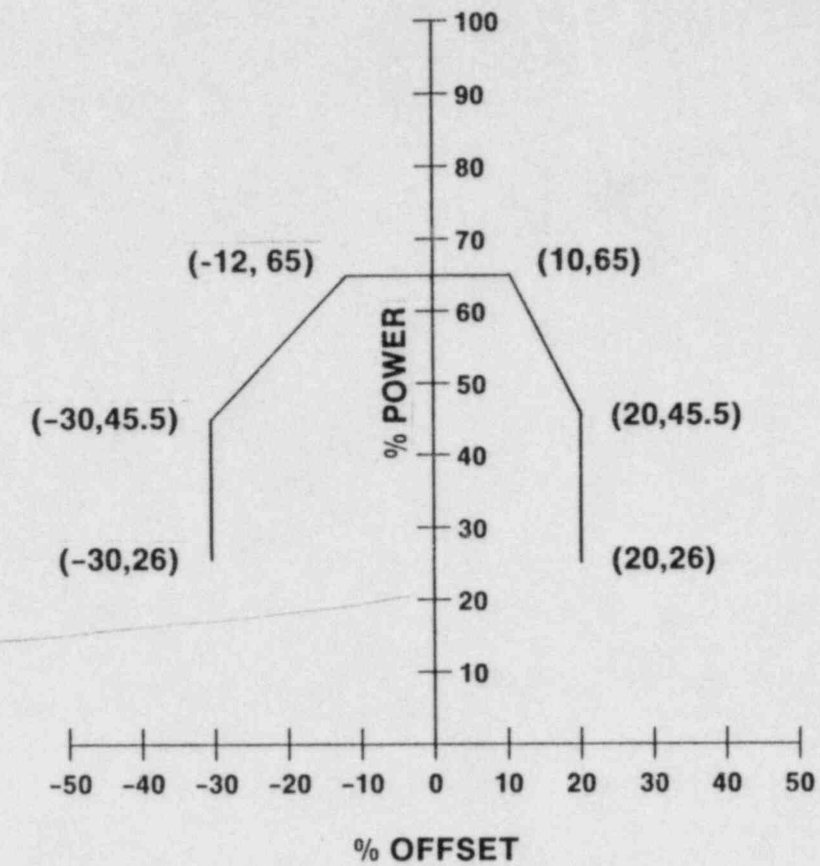
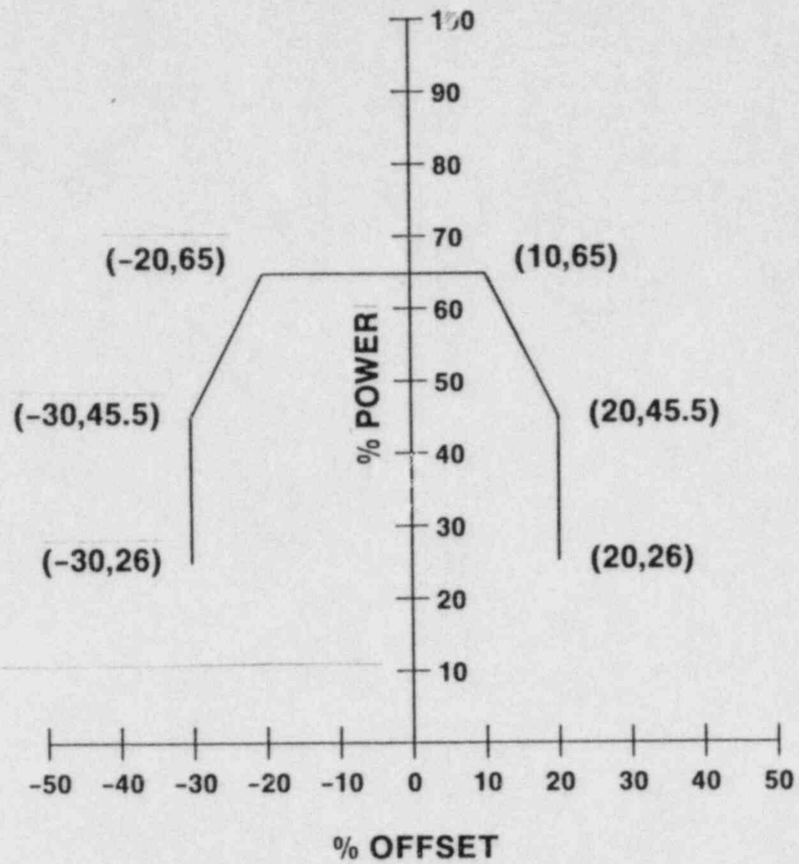


FIGURE 3.18-2c

POWER vs OFFSET, 250-EOC, THREE LOOP



3.20 Reactor Coolant System Flow, Temperature and Pressure

Applicability: Applies to MODE 1 steady state operation with four reactor coolant loops operating.

Objective: To set limiting conditions for operation for minimum nominal reactor coolant flow and pressure and maximum inlet temperature.

Specification:

A. Reactor Coolant Flow Rate

1) RCS Flow Rate $\geq$ 257,000 gpm

B. Reactor Coolant Temperature

1) T inlet $\leq$540.6°F

C. Reactor Coolant Pressure

1) Pressurizer Pressure$\geq$2000 psig*

D. The RCS flow rate shall be determinec by a heat balance within 7 EFPD of achieving 100% RATED THERMAL POWER after refueling.

E. Following the completion of Section D above, the above parameters shall be verified to be within the limits at least once per 12 hours. If any of the above parameters exceeds its specified limit, restore the parameter to within its above specified limit within two hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

Basis: The limiting conditions for operation have been expanded to include limits on flow, inlet temperature and pressure.

The flow rate requirements are based on a steam generator plugging/sleeving level consistent with a maximum of 500 equivalent plugged tubes per steam generator. Additionally, an evaluation has been performed to reduce the core bypass flow fraction from 9% to 4.5%. The reactor vessel flow rate decreases due to the assumed steam generator equivalent plugging levels, but the core flow increases due to the reduction in bypass flow.

* This limit is not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

The core inlet temperature of 540.6°F includes a ±4°F instrument error and deadband which would allow a maximum core inlet temperature of 544.6°F at 100% power. The maximum inlet temperature of 544.6°F is used in all current safety analyses, with the exception of the dropped rod analysis which used 533.9°F Tin. Sensitivity studies show that increasing the steady state temperature 10.7°F from 533.9°F to 544.6°F will result in a small reduction in minimum DNBR. Starting the rod dropped accident from a 10.7°F hotter condition will yield an increase in the end point temperature. If the dropped rod accident had been analyzed for 544.6°F, the maximum DNBR would be well above the fuel design limit of 1.3.

The minimum reactor coolant pressure of 2000 psig assumes ± 30 psig for instrument error and deadband which would allow a minimum core pressure of 1970 psig at 100% power.

The limiting values of the parameters in this specification are equal to, or more conservative than those assumed as the initial conditions in the accident and transient analyses; therefore, operation must be maintained within the specified limits for the accident and transient analyses to remain valid.

Docket No. 50-213

B11911

Attachment No. 2

Description of Proposed Technical Specification Changes
Haddam Neck Plant
Cycle 14

December, 1985

Description of Proposed Technical Specification Changes for Haddam Neck Plant, Cycle 14

The proposed Technical Specification changes have been prepared to support Cycle 14. The Technical Specifications being changed are those that are directly related to the fuel cycle design and safety/LOCA analyses that are impacted by steam generator tube plugging/sleeving. The reload fuel cycle design impacts several of the Technical Specifications due to the proposed conversion to Standard Technical Specifications and the design basis re-analysis methodology development. The changes due to steam generator tube plugging/sleeving are based on analyses that are affected by assuming a sleeving/plugging level equivalent to 500 plugged tubes per steam generator.

The proposed Haddam Neck Standard Technical Specification conversion amendment is scheduled to be submitted in 1986 during Cycle 14.

1.0 Definitions

The change to Definition 1.18, Quadrant Power Tilt Ratio (QPTR), eliminates the use of core ΔT as a primary means of determining the QPTR. This change is consistent with the Westinghouse Standard Technical Specifications (STS) and proposed reload methodology treatment of tilt. This treatment uses the excore neutron detectors to determine tilt and verify compliance. The relocation of the cold leg thermowells and replacement of the hot and cold leg RTDs during the refueling outage will yield a more accurate core ΔT. However, the use of core ΔT should only be used as a backup.

2.4 Maximum Safety Settings - Protective Instrumentation

The three requirements being changed in this Technical Specification are the variable low pressure trip setpoints, the footnote to the nuclear overpower setpoint concerning quadrant power tilt and three loop nuclear overpower setpoint.

The variable low pressure trip setting was previously determined taking into account streaming effects on the Tave and ΔT measurements. The trip was designed for a ΔT of 40°F corresponding to 100% power. The RTD's exhibited streaming effects because they were located on the Reactor Coolant pump suction piping. This causes an underprediction of ΔT. The RTD's are being relocated to the Reactor Coolant pump discharge piping. Because of the mixing that occurs in the Reactor Coolant pump, the Tave and ΔT measurements are no longer affected by streaming effects. The measured ΔT at 100% power will be higher, reflecting a more accurate measurement. The trip setting is being changed based on an expected ΔT of 46°F corresponding to 100% power, while maintaining the same margin between the trip setpoint and the safety limit curve. This setpoint is valid provided the indicated ΔT remains above 45°F.

The footnote to the nuclear overpower trip is being modified to remove the implied quadrant power tilt limit of 10%. Additionally, all references to using loop ΔT as a primary indication of tilt are being removed, consistent with the revised definition of the QUADRANT POWER TILT RATIO.

An evaluation of the required tilt allowance has been performed due to the conversion to Standard Technical Specifications (STS) and the development of in-house nuclear and safety analysis reload capabilities. The Westinghouse STS treatment of tilt is being proposed in this Technical Specification change and the proposed Haddam Neck STS. This treatment established a 2% tilt as the amount of allowable excore tilt change between monthly incore power distribution surveillances. This tilt can be accommodated since a peaking allowance consistent with 2% tilt is included in the Cycle 14 core offset limits. This peaking allowance is sufficient to account for core tilt due to control rod misalignment and flow and temperature maldistributions.

The final change is to the Nuclear Overpower reactor trip setting, while in three loop operation, which reduces the trip setpoint from 84% of rated power to 74% of rated power. Technical Specification 3.3, Reactor Coolant System Operational Components, establishes 65% rated thermal power as the upper limit for three pump operation, while the nuclear overpower trip setpoint is 84% rated thermal power. In order to provide the equivalent protection consistent with a 9% margin to the 84% trip setpoint, the gain of the nuclear instrumentation (NI) is currently adjusted to indicate 75% rated thermal power. The proposed nuclear overpower reactor trip setpoint of 74% provides the 9% margin at 65% rated thermal power, without increasing the indicated power through NI gain adjustments. The required margin to the DNB safety limit during an inadvertent control rod withdrawal accident is preserved. As a result, should reactor power increase above the maximum permissible 3-loop operation power level (65% of rated power), a reactor trip would be generated when 2 out of 3 channels indicate a power level no higher than 74% of rated power.

3.16 Isothermal Coefficient of Reactivity

This Technical Specification is being changed to be consistent with the proposed design basis re-analysis and delete references to a superseded LOCA analysis and correction factors in the BASIS.

The proposed range of allowable moderator temperature coefficients of $-2.9 \times 10^{-4}\ \Delta k/k/^{o}F$ to $0.0 \times 10^{-4}\ \Delta k/k/^{o}F$ is a more restrictive subset of the current range of $-3.5 \times 10^{-4}\ \Delta k/k/^{o}F$ to $+0.71 \times 10^{-4}\ \Delta k/k/^{o}F$ and is therefore acceptable with respect to the current design basis. The extra requirement at Hot Zero Power conditions is to account for the increase in MTC with decreasing temperature in the design basis reanalyses.

The current BASIS statement references a LOCA evaluation performed in the mid 1960's that was submitted as Amendment 14 to the Full Term Operating License application dated March 2, 1967. This LOCA analysis has been superseded by a revised model consistent with the AEC Interim Policy Statement "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors."

The correction factors provided in the BASIS, that are used to convert the isothermal coefficient to a moderator coefficient, have been deleted since these data were based on the Cycle 1 core design. Cycle specific parameters will be used to perform the adjustments as part of the startup physics test program.

3.17 Limiting Linear Heat Generation Rate

This Technical Specification is being changed to incorporate revised allowable peak linear heat generation rates that include the effects of steam generator tube plugging/sleeving on the large break LOCA. An additional change to the specification is a reduction in the engineering factor.

The large break LOCA has been re-analyzed by Westinghouse. The effects of heat transfer area and flow reduction based on 500 equivalent plugged tubes were included. A review of the model identified several input conservatisms that yielded margin to the peak cladding temperature (PCT) criteria of 2300°F for the currently allowed Linear Heat Generation Rate (LHGR limits), even with the assumed plugging levels. The resulting LHGR limits that yield a PCT of 2300°F are 14.5, 14.8 and >17.0 kw/ft for the 0-125 EFPD, 125-250 EFPD, and 250 EFPD-EOC burnup windows respectively. The proposed limits were conservatively reduced from these results to provide margin to the PCT limit.

The engineering factor (F_q^E) has been re-evaluated as part of the STS technical review. As with many of the design values in the fuel cycle design and safety analysis areas, the values were conservatively established by Westinghouse in the late 1960's. The review of the factor was performed by the current fuel vendor. This evaluation accounts for variations in pellet diameter, density, and enrichment and rod diameter. The engineering factor based on specified manufacturing tolerances was calculated to be 1.02 instead of the original design basis value of 1.04. The factor was also evaluated based on as-built data for Batch 15 (Cycle 13) and is equal to 1.009.

3.18 Power Distribution Monitoring and Control

The following items are being changed in this specification:

- o delete the requirement A.3 that requires a margin allowance for potential additional power peaking caused by control rod insertion
- o revise the requirement C.2 that requires the monthly average position of Bank B be at least 240 steps withdrawn
- o expand the requirement B.1.1.b to include specific three loop axial offset operating limits and provide revised 4 and 3 loop figures
- o delete the requirement B.1.1.d that specifies that the four loop axial offset limits are applicable to three loop operation by scaling the full power level.

Amendment 24 to the Provisional Operating License (D. C. Switzer, November 20, 1973, and D. J. Skovholt, April 2, 1974) added maneuvering restrictions to minimize xenon induced power peaking and implemented a power distribution monitoring requirement using axial flux traces and appropriate correlations from two incore thimbles. This monitoring requirement included adjustments to 100% power and maximum allowable rod insertion. These allowances were required since manual and/or automatic axial offset monitoring had not yet been implemented.

Amendment 1 to the Full Term Operating License (D. C. Switzer, August 21, 1974 and September 19, 1974, and R. A. Purple, January 28, 1975) added the following requirements:

1) prior to full power operation after refueling, verify the LHGR limits prior to exceeding 80% power; these measurements shall be adjusted to 100% power

2) repeat the measurement approximately 30 hours after achieving full power, including an allowance for the maximum allowable rod insertion

3) periodic power distribution monitoring using either:

   a) manual axial offset calculations using the excore detectors and compare the results with the calculated axial offset limits

   b) the 2 thimble method for verifying the LHGR limits

4) monitoring monthly Bank B average position when $\geq$20% power.

This was the first time that axial offset monitoring was used. Amendment 3 added the monthly average Bank B requirement when > 80% power (D. C. Switzer, May 12, 1975 and R. A. Purple, June 20, 1975). Amendment 6 (D. C. Switzer, August 20, 1975 and October 23, 1975, R. A. Purple, November 24, 1975) implemented continuous and automatic axial monitoring equipment and deleted the Bank B requirement when >80% power that had been added in Amendment 3.

Requirement 2 above included an allowance for rod insertion which was no longer needed when automatic and continuous axial offset monitoring was implemented. The axial offset limits are established to prevent exceeding the LHGR limits at the maximum allowable rod insertion. Therefore, as long as the automatic and continuous offset monitoring is used, no adjustments to account for rod insertion to the measured full core power distribution is required. This adjustment is required, however, if the 2 thimble method is used due to the unavailability of the automatic and continuous offset monitoring system.

The requirement to monitor the monthly average Bank B position has been in place since Amendment 24 to the Provisional Operating License (POL). This requirement is to minimize the potential for unfavorable axial power shapes and the possibility of exceeding design peaking limits, and has been in place prior to the concept of axial offset monitoring.

The first set of axial offset limits was provided in Amendment 1 to the Full Term Operating License (FTOL) and was based on Westinghouse methodology as applied to Haddam Neck (D. J. Miller, Connecticut Yankee Reactor Cycle 5 Peaking Factors Versus Axial Offset Study, Appendix A, WCAP-8386, August 1974). This methodology included maneuvering transients that had been initiated from depletion with Bank B above the insertion limit, thus assuring Technical Specification compliance.

The Babcock & Wilcox methodology for determining axial offset limits was approved and recognized in Amendment 10 to the FTOL (D. C. Switzer, May 3, 1976 and June 22, 1976, A. Schwencer, August 9, 1976). This methodology shows that the limiting transient is initiated off an all rods out depletion to maximize the negative axial offset, since the LHGR limit establishes the allowable negative axial offset. It has been shown in past cycle designs that there is no achievable positive offset that would cause the LHGR limits to be exceeded. The cycle specific positive axial offset limits have had to be determined using a conservative extrapolation. The generic positive axial offset limits have been conservatively reduced by 15-20% (axial offset units) to assure compliance with the monthly B average insertion limit.

The Cycle 14 design includes a conservative evaluation of a design maneuver off a nominal rod insertion depletion. Again, there was no achievable positive offset that would cause the LHGR limits to be exceeded, even with changes due to the conversion to STS and the design basis re-evaluation. The proposed positive generic axial offset limits have been reduced and the allowable Bank B insertion requirement has been made more restrictive. This reduction in the generic axial offset limits assures that the proposed bank average insertion of 280 steps is acceptable.

The axial offset limit requirements in Specification B.1.1.b have been expanded to include operating tents for three loop operation. These new tents are required in order to preserve the three loop LHGR limits. The four loop axial offset limits scaled to three loop conditions do not preserve the LHGR limits for Cycle 14. Additional information was provided in J. F. Opeka letter to J. A. Zwolinski, dated August 16, 1985.

The revised generic axial offset limits for four and three loop operation are shown in Figure 3.18. These limits are more restrictive than previous cycle limits even though the allowable LHGR's were increased. The net reduction of

the limits is due to the following:

- o Two consecutive coastdowns.
- o Conservative axial offset methodology due to the proposed STS conversion and design basis re-analysis.

3.20 Reactor Coolant System Flow, Temperature and Pressure

This Technical Specification is being changed due to the potential for steam generator tube plugging/sleeving during the 1986 outage. In order to support this proposed change, the following evaluations have been performed:

1) Determine the Reactor Coolant System (RCS) flowrate as a function of symmetric tube plugging levels.

2) Determine the core bypass flow fraction.

The reactor vessel flow rate assuming 500 tubes plugged per steam generator is based on the original measured flow rate of 278,200 GPM and the change in the RCS flow resistance due to plugging. The estimated flow rate is 271,300 GPM. A conservative measurement uncertainty reduces the vessel flow rate to 257,000 GPM. This flow rate is equivalent to 96% of the original design flow rate of 268,800 GPM.

A study has been performed by Westinghouse to determine the core bypass flow fraction. Westinghouse had initially indicated that the design basis value of 9% may have been based on cruciform control blades rather than Rod Cluster Control Assemblies. The Westinghouse study confirmed that the bypass flow fraction can be reduced from 9% to 4.5%.

The design basis flow rate assumptions used in the safety analysis and proposed Technical Specification are summarized below:

	FDSA	PROPOSED
Vessel Flow Rate, gpm	268,000	257,000
Core Flow Rate, gpm	244,600	245,400
Bypass Flow	9%	4.5%

The net core flowrate, which is derived from the vessel flow rate and bypass fraction is the key input in the safety analysis. Even though the vessel flow rate has been reduced due to tube plugging, the net core flow remains above the FDSA assumed core flow rate due to the reduction in bypass flow fraction. Therefore, the transient analyses and safety limit curves in Technical Specification 2.1 remain valid.

Docket No. 50-213

Attachment No. 3

Safety Evaluation for
Proposed Technical Specification Changes
Haddam Neck Plant
Cycle 14

December, 1985

Safety Evaluation for Proposed Technical Specification Changes, Haddam Neck Plant, Cycle 14

Effect on Design Basis Accident Analysis

Section 1.0

The definition of Quadrant Power Tilt Ratio (QPTR), eliminates the use of core ΔT in determining QPTR and replaces the definition with one consistent with Westinghouse Standard Technical Specifications. In the proposed definition, the QPTR is determined using the excore detectors. This method of determining QPTR is considered to be more accurate than basing tilt on the output of the reactor coolant system RTDs.

The proposed changes to Section 1.0 do not:

1. increase the consequences of a previously analyzed accident,

2. create the potential for an accident of a different type than already analyzed,

3. decrease the margin of safety as specified in the basis of any technical specification.

Section 2.4

Variable Low Pressure Trip - The variable low pressure trip (VLPT) provides protection for slow reactivity insertion events (rod withdrawal). The present variable low pressure trip accounts for an offset in the cold leg temperature measurement due to the non-uniform temperature distribution at the location of the cold leg RTDs (upstream of the reactor coolant pumps). Since the RTDs will be moved to a location downstream of the reactor coolant pumps during the 1986 refueling outage, the temperature offset will no longer exist. As a result, the present VLPT will result in more margin to the safety limit but not enough operating margin. The proposed change to the VLPT will provide the same margin to the DNB limit as the present specification as long as the actual 100% power vessel ΔT is greater than or equal to 45°F. It is expected that, as a result of relocating the RTDs, full power ΔT will exceed 46°F. The value of expected full power ΔT will be determined from plant data before returning to full power after the 1986 refueling outage.

Tilt Limit - The reduction in the allowable QPTR from 10% to 2%, and the change in the required actions are consistent with Westinghouse standard technical specifications. This tilt limit can be accommodated since a peaking allowance consistent with 2% tilt is included in the core offset limits. This peaking allowance is sufficient to account for core tilt due to control rod misalignment and flow and temperature maldistributions. The proposed changes are more restrictive than the present specifications allow.

Nuclear Overpower Trip - The purpose of the Nuclear Overpower Trip setpoints are to cause the reactor to trip before fuel design limits are exceeded during normal operation and design transients. The maximum allowable reactor power is not changed by this proposed change.

This change only clarifies the Technical Specifications to indicate the actual reactor protection system trip setpoint.

The proposed changes to Section 2.4 do not:

1. increase the consequences of a previously analyzed accident,

2. create the potential for an accident of a different type than already analyzed,

3. decrease the margin of safety as specified in the basis of any technical specification.

Section 3.16

The proposed change imposes a more restrictive limit on the hot full power (HFP) moderator temperature coefficient (MTC) and adds limits for beginning of cycle (BOC) hot zero power (HZP) and end of cycle (EOC) HFP conditions.

The Amendment 14 LOCA analysis referenced in the Basis has been superseded by analyses in accordance with the Interim Acceptance Criteria and approved by the AEC (NRC). The adjustment factors to convert the measured Isothermal Temperature Coefficient to an Moderator Temperature Coefficient (MTC) will be deleted, since these data were based on the Cycle 1 core design.

As indicated in the basis of the proposed specification, the MTCs assumed in the safety analysis bound the proposed MTC values.

The proposed changes to Section 3.16 do not:

1. increase the consequences of a previously analyzed accident,

2. create the potential for an accident of a different type than already analyzed,

3. decrease the margin of safety as specified in the basis of any technical specification.

Section 3.17

The Linear Heat Generation Rate (LHGR) limits are being revised to account for the effects of plugging/sleeving steam generator tubes on the large break LOCA analysis. Other changes to the analysis inputs (upper and lower plenum volumes) dominate the effects of tube plugging. As a result, the LHGR limits can be increased as proposed without changing the consequences of the large break LOCA analysis (peaking cladding temperature). The revised LHGR limits are based on a maximum steam generator plugging/sleeving level of 500 equivalent tubes per steam generator. The inputs to the currently approved licensed model were reviewed and the effects of reduced flow and heat transfer area were included. These revised LHGR limits assure that the peak cladding temperature during a LOCA does not exceed the 2300°F criterion.

An additional change to this Technical Specification is a reduction in the hot channel engineering factor from 1.04 to 1.02. As part of the design basis

verification effort to support conversion to Standard Technical Specifications (STS), a review of this factor was performed by the fuel vendor. This review shows that a value of 1.02 is justified based on the current fuel design. This factor is used in the fuel cycle design task that establishes the axial offset limits and is used as an adjustment factor in the monthly incore flux map evaluation.

The proposed changes to Section 3.17 do not:

1. increase the consequences of a previously analyzed accident,

2. create the potential for an accident of a different type than already analyzed,

3. decrease the margin of safety as specified in the basis of any technical specification.

Section 3.18

The proposed changes to the Technical Specification provide revised generic four loop and new three loop axial offset limits. Other minor changes made to this Technical Specification clarify a monitoring requirement and conservatively revise a fuel cycle design basis consistent with the proposed reload analysis methodology.

The revised four loop and new three loop generic axial offset limits were derived based on assumptions that are consistent with the revised allowable tilt in Specification 2.4 and the revised LHGR limits in Specification 3.17. These limits will also preserve a proposed STS limit on the enthalpy rise hot channel factor throughout the cycle. These revised bases for the axial offset limits are more restrictive than the current basis since no credit is taken for the radial peak reduction due to burnup.

The proposed change to delete the monitoring requirement of including an allowance for potential additional power peaking due to allowable rod insertion could have been removed when automatic and continuous monitoring of axial offset was implemented. The current requirement is a holdover from the era when the only means of monitoring the power distribution was the monthly incore flux map. The deletion of this requirement will affect the LHGR adjustment factor used in the evaluation of the monthly incore flux maps.

The proposed change in the allowable monthly average position of Bank B is a more restrictive requirement. Limiting the monthly average Bank B insertion as proposed, assures that the more restrictive positive axial offset limits are preserved.

These proposed changes are based on more restrictive analysis assumptions (e.g., tilt, radial power peaking, allowable Bank B insertion) and revised LHGR limits. These restrictions are consistent with the proposed conversion to STS.

In summary, the changes to Specification 3.18 are based on more restrictive analysis assumptions. The proposed changes do not:

1. increase the consequences of a previously analyzed accident,

2. create the potential for an accident of a different type than already analyzed,

3. decrease the margin of safety as specified in the basis of any technical specification.

Section 3.20

The changes being made include changing the applicability to all Mode 1 operation instead of 100% power, and including a specification on the required minimum Reactor Coolant System (RCS) flowrate. Previously the value for RCS flowrate was identified only in the Basis section of the specification. The decrease in minimum allowable RCS flowrate will bound the anticipated effects of steam generator repair scheduled for the 1986 refueling outage. This repair effort is expected to be bounded by the assumed 500 equivalent plugged tubes in each steam generator.

Surveillance requirements are added that require a heat balance be performed within seven EFPD of achieving 100% RATED THERMAL POWER after refueling and that the flow rate, temperature, and pressure be verified at least once per 12 hours. The action statement was expanded to require a power reduction to less than 5% of RATED THERMAL POWER within 4 hours if a parameter cannot be restored to within its limit within 2 hours.

Effect on Design Basis Analyses

Although the RCS flowrate is expected to be reduced due to steam generator repair, the flow through the core is expected to be greater than previously analyzed in the accident analysis. This is because the existing design basis analyses conservatively assume that 9% of the RCS flow bypasses the core. Recent calculations by Westinghouse, using conservative as-built information, indicate that the bypass flow is only 4.5%.

LOCA

The change in RCS flowrate has a negligible effect on the design basis small break LOCA analysis. The increase in the core flow will slightly reduce the initial fuel temperatures and stored energy. The eff ; of the lower fuel temperatures on the small break LOCA results will therefore be potentially beneficial to the small break LOCA response. Moreover, for the limiting break analysis, reported in a letter from D. C. Switzer to D. J. Skovolt, dated May 19, 1972, core heat up does not occur until after 380 seconds. This time span allows the stored energy in the rods to be removed. Therefore, changes in the initial stored energy due to changes in core flow will have a negligible effect on the small break LOCA peak clad temperatures. The impact on the design basis large break LOCA was discussed in Section 3.17 and the impact on the LHGR limits.

Non-LOCA

To determine the effect of tube plugging, the thermal performance of the reactor during hypothetical incidents was evaluated to ensure that it is not degraded. An evaluation of each transient was made and if the event was found to be affected by the tube plugging, it was reanalyzed.

The consequences of tube plugging are as follows:

- o Reactor coolant flow is reduced due to increased steam generator flow resistance.
- o The primary flow and steam generator heat transfer area are reduced. Thus, to maintain the 100% steam flow, Tavg must be increased or steam pressure reduced.
- o Primary reactor coolant mass inventory is reduced.

In order to facilitate this evaluation, the Haddam Neck design basis accident analyses were grouped into four categories as outlined below:

Departure from Nucleate Boiling (DNB) events
Overcooling events
Overheating events
Various reactivity events.

DNB Design Basis Events

The Standard Review Plan (Section 4.2) specifies the acceptance criteria for various fuel design limits. One of these criteria is that there will be at least a 95%/95% probability/confidence level that DNB will not occur on the limiting fuel rods during normal operation, operational transients and any transient conditions arising from Condition 1 and 2 events.

In order to demonstrate that the Haddam Neck Plant meets these criteria, the minimum transient DNB ratios were calculated for various transients in the design basis analyses.

Tube plugging could effect the DNB design basis events as a result of:

1 - Reduced initial core flow

2 - More rapid primary flow coastdown for the loss of flow event due to increased steam generator flow resistance.

In order to accommodate the first effect, a study has been performed by Westinghouse to determine the core bypass flow fraction. Westinghouse had initially indicated that the design basis value of 9% may have been based on cruciform control blades rather than the RCCAs. The Westinghouse study, confirmed that the bypass flow fraction can be reduced from 9% to 4.5%.

With a conservative vessel flow rate of 257,000 GPM and a 4.5% bypass flow, the net core flow is 245,400 GPM which exceeds the original design value of 244,600 GPM. Since the net core flow rate, which is the key input in the safety analysis, remains above the design value, it can be concluded that the initial starting point for the DNB design basis events remains valid. (Note, the above conclusion can also be applied to the Technical Specification safety limit curves).

The more rapid primary flow coastdown will tend to increase the consequences for the loss of flow event. However, the Technical Specification package also includes a reduction in the allowable positive moderator temperature coefficient (MTC). The proposed change imposes a more restrictive limit on the HFP moderator coefficient (Technical Specification 3.16), as outlined below:

HFP Moderator Coefficient

Original Design	Proposed Tech. Spec.
10. PCM/F	0.0 PCM/F

The reduction in the allowable positive MTC will more than offset any impact of the faster flow coastdown.

Based on the above discussion, it has been concluded that plugging 500 equivalent tubes per steam generator will not impact the DNB design bases of the plant.

Overcooling Design Basis Events

Tube plugging could affect the consequences of this class of transients in the following ways:

1. Reduced heat transfer surface area would lower the rate and extent of the primary cooldown.

2. Increased steam generator resistance would lower the RCS flow rate, and therefore lower the rate of primary cooldown.

Both of these effects are in the conservative direction. Therefore, tube plugging would actually reduce the consequences of the overcooling events.

Overheating Design Basis Events

The purpose of analyzing the overheating transients is as follows:

- To demonstrate that the RCS pressure will not exceed 110% of design pressure during Condition 1 and 2 events. (This criterion is assured by applying the more stringent requirement that the pressurizer must not be filled with water during the event.)

- To determine the design bases of the auxiliary feedwater system.

The most limiting transients analyzed under this category are the loss of load and loss of normal feedwater. The impact of tube plugging is as follows:

1. Reduced RCS flow rate

2. Reduced steam generator heat transfer area

3. Decreased RCS mass inventory

However, the most important parameter for this class of events is the moderator temperature coefficient (MTC). As indicated in Attachment 1, Technical Specification 3.16 is being revised to impose a more restrictive limit on the hot full power moderator temperature coefficient. The reduction in the maximum allowable MTC more than compensates for the effects of tube plugging.

Various Reactivity Events

The following transients can be grouped under this category:

- Rod withdrawal
- Boron dilution
- Dropped rod
- Control rod ejection

This class of transients is primarily affected by changes to core kinetics characteristics, control rod worths, and core power distribution.

As indicated previously, tube plugging will not cause a significant change in any of the key neutronic parameters of the plant.

In summary, the proposed changes to Section 3.20 do not:

1. increase the consequences of a previously analyzed accident,

2. create the potential for an accident of a different type than already analyzed,

3. decrease the margin of safety as specified in the basis of any technical specification.

Summary And Conclusion

Because of the above assessments, the changes implemented by these proposed changes are considered to be safe and do not constitute an unreviewed safety question or a significant hazard consideration, as defined in 10 CFR 50.59 and 10 CFR 50.92, since they do not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

2. create the possibility for an accident or malfunction of a different type than previously analyzed in the safety analysis report, or

3. reduce the margin of safety as defined in the basis of any technical specification.