2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. ~~through the W-3L correlation. The W-3L DNB~~ This correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

~~The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value bounds a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.~~

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, pressurizer pressure and core inlet temperature for which the ~~minimum~~ calculated DNBR is no less than ~~1.30, and the core outlet void fraction is no greater than 0.32.~~ the design DNBR value.

These curves are based on total enthalpy hot channel factors, $F^N_{\Delta H}$, as stated in technical specification 3.2.3.1. ~~of 1.60 for four loop operation.~~ An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power.

The limiting hot channel factor is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. This insertion limit is described in Specification 3.1.3.6 and shown in the Technical Report Supporting Cycle Operation. The required reduction in power level as dictated by the control rod insertion limits insures that the DNB ratio is always greater than ~~1.30.~~ the design DNBR value.

The DNB design basis is as follows: The uncertainties in the WRB-1 or W-3L correlation, such as power peaking parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with a 95 percent confidence level that DNB will not occur. This establishes a design DNBR value which must be met in the plant safety analyses using values for power peaking parameters without uncertainties. All uncertainties in plant operating parameters are applied in the conservative direction when used in the safety analyses. Margin has been maintained to take into account mixed core effects and rod bow

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4.2.2.1.2 Measured values of core power peaking factors used in determining LHGRs shall include the following allowances:

		B&W FUEL	W FUEL
~~a.~~	~~Normal power peaking* **,~~		
a ~~b.~~	Measurement uncertainty of 1.05,	1.05	1.05,
b ~~c.~~	Statistical density factor of 1.012,	1.012	—,
c ~~d.~~	Engineering factor of 1.02,	1.02	1.03,
d ~~e.~~	Stack shortening/thermal expansion factor of 1.007, and		—
e ~~f.~~	Power level uncertainty of 1.02.	1.02	1.02

~~* Items a. is chosen at a core height to maximize the product.~~

~~** Determined in accordance with Specification 4.2.2.1.1, using the thimble location which yields the higher total core peaking factor.~~

the design DNBR value

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to ~~1.30~~ during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS Interim Acceptance Criterion limit of 2300°F peak cladding temperature for stainless steel clad fuel and the 10CFR50.46 and Appendix K limit of 2200°F peak cladding temperature for zircaloy fuel are not exceeded.

3/4.2.1 AXIAL OFFSET

The AXIAL OFFSET specification provides continuous confirmation of acceptable LINEAR HEAT GENERATION RATES (LHGR) during the time interval between incore measurements.

3/4.2.2 LINEAR HEAT GENERATION RATE

Limiting the peak LINEAR HEAT GENERATION RATE (LHGR) during Condition I events provides assurance that the initial condition assumed for LOCA analyses are met and the peak cladding temperature limits are not exceeded.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$

The limit on the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) ensures that the minimum DNBR limit is not exceeded.

The $F^N_{\Delta H}$ is measurable, but will normally only be determined periodically as specified in Specification 4.2.3.1.2. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. The control rod insertion limits provided in the TECHNICAL REPORT SUPPORTING CYCLE OPERATION are maintained, and

b. The AXIAL OFFSET limits provided in the TECHNICAL REPORT SUPPORTING CYCLE OPERATION are maintained.

The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. The full power limits include a 4% incore measurement uncertainty.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in power capability analysis. Radial

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4 or Type 304 stainless steel.* Each stainless steel clad fuel assembly shall have a nominal active fuel length of 120.5 inches and contain a typical weight of 411.5 kilograms uranium. Each Zircaloy-4 clad B&W fuel assembly shall have a nominal active fuel length of 118.6 inches and contain a typical weight of 363.8 kilograms uranium. The core loading shall have a maximum enrichment of ~~4.00~~ 5.00 weight percent U-235. EACH Zircaloy-4 clad WESTINGHOUSE fuel assembly shall have a nominal active fuel LENGTH OF 120.3 inches AND CONTAIN A TYPICAL WEIGHT OF 386.2 KILOGRAMS URANIUM.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 120.25 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 8780 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

~~*For Cycles 16 and 17 the core will also include two (2) solid Type 304 stainless steel "filler rods" in place of two (2) fuel rods.~~

6. Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$--Specification 3/4.2.3.1.

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

1. F. M. Akstulewicz to E. J. Mroczka, "Review of NUSCO Topical Report on Physics Methodology for PWR Reload Design (NUSCO-152)," August 3, 1987.

2. A. B. Wang to E. J. Mroczka, "Safety Evaluation for Northeast Utilities Topical Report 140-1, NUSCO Thermal Hydraulic Qualification, Volume I (RETRAN)," July 26, 1988.

3. F. M. Akstulewicz to J. F. Opeka, "NUSCO Thermal Hydraulic Model Qualification, Volume II (VIPRE), Topical Report NUSCO 140-2," October 16, 1986.

4. A. B. Wang to E. J. Mroczka, "Safety Evaluation of Northeast Utilities Topical Report 151, Haddam Neck Non-LOCA Transient Analysis," October 18, 1988.

5. Supplement to the Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission Docket No. 50-213, Connecticut Yankee Atomic Power Company, Haddam Neck Plant, December 27, 1974.

6. A. B. Wang to J. F. Opeka, "Issuance of Amendment (TAC Nos. M80864, M82284, M66958, M71769, and M72083)," Part 3 of Safety Evaluation--Large-Break Loss-of-Coolant Accident Analysis Zircaloy Clad Fuel Evaluation, dated January 17, 1992.

7. A. B. Wang to J. F. Opeka, "Issuance of Amendment (TAC Nos. M80864, M82284, M66958, M71769, and M72083)," Part 4 of Safety Evaluation--Small-Break Loss-of-Coolant Accident Analysis Zircaloy Clad Fuel Evaluation, dated January 17, 1992.

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

6.9.1.9.d The TECHNICAL REPORT SUPPORTING CYCLE OPERATION, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

8. , "REVIEW OF NUSCO TOPICAL REPORT ON PHYSICS METHODOLOGY FOR PWR RELOAD DESIGN (NUSCO-152), Addendum 3" , 1994.

9. , "VIPRE/WRB-1 DNBR THERMAL LIMIT FOR WESTINGHOUSE FUEL TYPES", , 1994.

Docket No. 50-213
B14808

Attachment 2

Haddam Neck Plant

Proposed Revision to Technical Specifications
Cycle 19 Operation - Retyped Pages

May 1994

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

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The DNB design basis is as follows: The uncertainties in the WRB-1 or W-3L correlation, such as power peaking parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that DNB will not occur. This establishes a design DNBR value which must be met in the plant safety analyses using values for power peaking parameters without uncertainties. All uncertainties in plant operating parameters are applied in the conservative direction when used in the safety analyses. Margin has been maintained to take into account mixed core effects and rod bow.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, pressurizer pressure and core inlet temperature for which the calculated DNBR is no less than the design DNBR value.

These curves are based on total enthalpy hot channel factors, $F^N_{\Delta H}$, as stated in Technical Specification 3.2.3.1. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power.

The limiting hot channel factor is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. This insertion limit is described in Specification 3.1.3.6 and shown in the Technical Report Supporting Cycle Operation. The required reduction in power level as dictated by the control rod insertion limits insures that the DNB ratio is always greater than the design DNBR value.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.1.2 Measured values of core power peaking factors used in determining LHGRs shall include the following allowances:

		B&W Fuel	*W* Fuel
a.	Measurement uncertainty	1.05	1.05
b.	Statistical density factor	1.012	—
c.	Engineering factor	1.02	1.03
d.	Stack shortening/thermal expansion factor	1.007	—
e.	Power level uncertainty	1.02	1.02

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the design DNBR value during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS Interim Acceptance Criterion limit of 2300°F peak cladding temperature for stainless steel clad fuel and the 10CFR50.46 and Appendix K limit of 2200°F peak cladding temperature for zircaloy fuel are not exceeded.

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b. The AXIAL OFFSET limits provided in the TECHNICAL REPORT SUPPORTING CYCLE OPERATION are maintained.

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5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 120.25 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium.

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c. For a temperature of 650°F, except for the pressurizer which is 680°F.

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5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 8780 cubic feet.

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5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

6. Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$--Specification 3/4.2.3.1.

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

1. F. M. Akstulewicz to E. J. Mroczka, "Review of NUSCO Topical Report on Physics Methodology for PWR Reload Design (NUSCO-152)," August 3, 1987.

2. A. B. Wang to E. J. Mroczka, "Safety Evaluation for Northeast Utilities Topical Report 140-1, NUSCO Thermal Hydraulic Qualification, Volume I (RETRAN)," July 26, 1988.

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4. A. B. Wang to E. J. Mroczka, "Safety Evaluation of Northeast Utilities Topical Report 151, Haddam Neck Non-LOCA Transient Analysis," October 18, 1988.

5. Supplement to the Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission Docket No. 50-213, Connecticut Yankee Atomic Power Company, Haddam Neck Plant, December 27, 1974.

6. A. B. Wang to J. F. Opeka, "Issuance of Amendment (TAC Nos. M80864, M82284, M66958, M71769, and M72083)," Part 3 of Safety Evaluation--Large-Break Loss-of-Coolant Accident Analysis Zircaloy Clad Fuel Evaluation, dated January 17, 1992.

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8. ________________________, "Review of NUSCO Topical Report on Physics Methodology for PWR Reload Design (NUSCO-152) Addendum 3," __________, 1994.

9. ________________________, "VIPRE/WRB-1 DNBR Thermal Limit for Westinghouse Fuel Types," __________, 1994.

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

6.9.1.9.d The TECHNICAL REPORT SUPPORTING CYCLE OPERATION, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.