Docket No. 50-423
B14378

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Design Features — Fuel Assemblies
Markup Pages

March 1993

5.3 REACTOR CORE

See Insert 5.3.1

FUEL ASSEMBLIES

~~5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rod locations. Fuel rod locations may at any time during plant life have any combination of, 1) fuel rods clad with zircaloy-4, 2) filler rods fabricated from zircaloy-4 or stainless steel, or 3) vacancies, as determined by cycle-specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum nominal enrichment of 3.4 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 5.0 weight percent U-235.~~

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 61 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 95.3% hafnium and 4.5% natural zirconium or 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel.

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 Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, 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 12,240 cubic feet at a nominal T_{avg} of 587°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.

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Insert 5.3.1

5.3.1 The core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 264 zircaloy-4 or ZIRLO clad fuel rods with an initial composition of natural uranium dioxide or a maximum nominal enrichment of 5.0 weight percent U-235 as fuel material. Limited substitutions of zircaloy-4, ZIRLO or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs tnat have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design bases. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CORE OPERATING LIMITS REPORT (Cont.)

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

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

4. Axial Flux Difference Limits, target band, and APL^{ND} for Specifications 3/4.2.1.1 and 3/4.2.1.2,

5. Heat Flux Hot Channel Factor, K(z), W(z), APL^{ND}, and $W(z)_{BL}$ for Specifications 3/4.2.2.1 and 3/4.2.2.2.

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

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

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

2. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1981 (W Proprietary).

3. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), January 31, 1980--Attachment: Operation and Safety-Analysis Aspects of an Improved Load Follow Package.

4. NUREG-800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981 Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Revision 2, July 1981.

5. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary). (Methodology for Specifications 3.2.1--Axial Flux Difference [Relaxed Axial Offset Control] and 3.2.2--Heat Flux Hot Channel Factor [W(z) surveillance requirements for F_Q Methodology].)

6. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS--SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July 1986 (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.)

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

Insert 6.9.1.6.b

8. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," September 1988 (W Proprietary).

CORE OPERATING LIMITIS REPORT (Cont.)

Insert 6.9.1.6.b

9. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," August 1985 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

10. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

11. WCAP-12610, "VANTAGE+ Fuel Assembly Report," June 1990 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

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Retyped Pages

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