Docket No. 50-423
B17004

Attachment 4
Millstone Nuclear Power Station Unit No. 3
License Amendment Request and Technical Specification
Changes For Full Core Off-load
**Marked-Up Technical Specification Page(s)**

January 1999

January 3, 1995

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

| AREA | TEMPERATURE LIMIT (°F) |
| --- | --- |
| 7. FUEL BUILDING | |
| FB-02, Fuel Pool Pump Cubicles, El 24'6" | ≤ ~~110~~ 119 |
| FB-03, General Area, El 52'4" | ≤ ~~104~~ 108 |
| 8. FUEL OIL VAULT | |
| FV-01, Diesel Fuel Oil Vault | ≤ 95 |
| 9. HYDROGEN RECOMBINER BUILDING | |
| HR-01, Recombiner Skid Area, El 24'6" | ≤ 125 |
| HR-02, Controls Area, El 24'6" | ≤ 110 |
| HR-03, Sampling Area, El 24'6" | ≤ 110 |
| HR-04, HVAC Area, El 37'6" | ≤ 110 |
| 10. MAIN STEAM VALVE BUILDING | |
| MS-01, Areas Above El 58'0" | ≤ 140 |
| MS-02, Areas Below El. 58'0" | ≤ 140 |
| 11. TURBINE BUILDING | |
| TB-01, Entire Building | ≤ 115 |
| 12. TUNNEL | |
| TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building | ≤ 112 |
| 13. YARD | |
| YD-01, Yard | ≤ 115 |

Docket No. 50-423
B17004

Attachment 5

Millstone Nuclear Power Station Unit No. 3
License Amendment Request and Technical Specification
Changes For Full Core Off-load
**Marked-Up Updated Final Safety Analysis Report Page(s)**

January 1999

## INDEX OF MARKUP OF PROPOSED REVISION

Refer to the attached markup of the proposed revision to the Updated Final Safety Analysis Report (UFSAR). The attached markup reflects the currently issued version of the Updated FSAR.

The following changes are included in the attached markup.

### Affected UFSAR Sections

| Section | Title | Page |
|---|---|---|
| Table 1.3-15 | Comparison of Other Reactor Plant Systems | 1 of 4 |
| Table 1.9-1 | Summary of Differences from SRP | 14 of 16 |
| Table 1.9-2 | SRP Differences and Justifications | 33 of 41 |
| Table 3.6-3 | Moderate-Energy Systems Outside Containment Remote from Essential Systems, Components and Structures | 2 of 4 |
| 3.8.4.3 | Loads and Loading Combinations | 3.8-40 |
| Appendix 3B | Attachment 1, Appendix E, Environmental Design Conditions | |
| Chapter 9 | List of Figures | 9-viii |
| 9.1.2.1 | Design Bases | 9.1-2 |
| 9.1.3.1 | Design Bases | 9.1-6 |
| 9.1.3.2 | System Description | 9.1-6 |
| 9.1.3.3 | Safety Evaluation | 9.1-8 |
| 9.1.3.3 | Safety Evaluation | 9.1-9 |
| Figure 9.1.7 | Normal Refueling | |
| Figure 9.1.7A | End of Cycle Full core Offload | |
| Figure 9.1-8 | Emergency Core Offload | |
| Table 9-1.1 | Fuel Pool Cooling and Purification System Principal Component Design Characteristics | |
| Table 9.1-2 | Performance Characteristics of the Fuel Pool Cooling System | |
| 9.4.2.1 | Fuel Building Ventilation System - Design Bases | 9.4-10 |

### Details of Changes

1. Table 1.3-15, Fuel Pool Cooling and Purification System (Section 9.1.3): Under the column labeled "Millstone 3", delete "1 1/3" for "Number of cores cooled" and replace with "15 1/2 (3048 fuel assemblies)."

   Under the column labeled "Millstone 3", delete "140" for "Fuel pool temperature, normal (°F)" and replace with "150".

2. Table 1.9-1, page 14 of 16, SRP Section 9.1.3, delete "Decay heat .... BTP ASB 9-2." and replace with "Decay heat removal is based on DECOR (based on ORIGEN2) computer code and credit for evaporative cooling instead of BTP ASB 9-2."

   Insert the additional SRP Section 9.1.3 as shown below.

| SRP Section | Specific SRP Acceptance Criteria | Summary Description of Difference | Corresponding FSAR Section |
|---|---|---|---|
| 9.1.3 (Rev. 1) | III.1.d - for maximum normal heat load, the pool temperature should be kept at or below 140°F | The maximum temperature for a normal heat load is 150°F. | 9.1.3.3 |
| 9.1.3 (Rev. 1) | III.1.h (ii) - maximum heat load is after 150 hours of decay | The decay time for the maximum heat load is based on the heat removal capacity of the spent fuel pool heat exchangers and varies from 165 hours to 349 hours. | 9.1.3.3 |

3. Table 1.9-2, page 33 of 41, SRP 9.1.3, Section A: delete "Westinghouse generated curves," and replace with "the DECOR computer code (based on ORIGEN2) and credit for evaporative cooling," Number this difference "1".

   Insert the following as difference 2. "The maximum temperature for a normal heat load is 150°F, not 140°F as required by SRP 9.1.3 Paragraph III.1.d."

   Insert the following as difference 3: "The decay time for the maximum heat load in the spent fuel pool is based on the heat removal capacity of the spent fuel pool heat exchangers and varies from 165 hours to 349 hours, not 150 hours as required by SRP 9.1.3, Paragraph III.1.h(ii)."

   Section B: delete "current NSSS curves utilized for the fuel that will actually be supplied to Millstone 3 " and replace with "the DECOR computer code (based on ORIGEN2) and credit for evaporative cooling in order to get a more accurate value of decay heat loads". Number this justification "1".

   Insert the following as justification 2: "All SSCs associated with the Spent Fuel Pool have been evaluated and have been found to be acceptable for an increase over the SRP limit of 140°F. The decay heat of the fuel is removed and the water coverage of the fuel is maintained for all anticipated scenarios."

Insert the following as justification 3: "The decay time for the maximum heat load in the spent fuel pool is based on heat removal capacities that are dependent on the actual cooling water temperatures. Colder cooling water temperatures result in greater the heat removal capacities which permit larger heat loads to be placed in the pool and the shorter the decay time."

4. Table 3.6-3, page 2 of 4, item "Fuel Pool Cooling and Purification", under column "Location" add "FB". Under column "Temperature" change "140" to "150".

5. Section 3.8.4.3, page 3.8-40: Delete the second sentence ("The spent fuel pool walls... Figures 3.8-79 and 3.8-80.") and replace with the following insert.

   "The historical design of the spent fuel pool walls and mat considered the thermal effects based on the temperatures indicated in Figures 3.8-79 and 3.8-80. The analysis for classifying a full core off-load as a normal evolution evaluated the thermal effects based on temperatures indicated in Figure 3.8.82."

6. Add Figure 3.8.82

7. Appendix 3B, Attachment 1, Appendix E, Environmental Design Conditions: Revised to show the updated environmental conditions associated with a normal fuel temperature of 150°F in accordance with calculation P(B)-1118, rev. 0, CCN 1. Note 1 was revised to show that the expected post LOCA pool temperature transient is enveloped by the transient used to determine the EEQ environment.

   Insert the following in pages 1 through 6 as marked.
   Insect B: "and the increased area temperatures during filtration operation."

   Insect C: "These temperatures account for the increased area temperatures during filtration operation."

   Insert D: "The above postulated condition envelopes the actual expected post DBA pool heat up temperature transient. See section 9.1.3.3 for further information."

8. Revise the List of Figures as follows:
   Figure 9.1-7, delete Title and add "Bulk Pool Transient Temperature Plot (Full Core Offload)"
   Figure 9.1-7a, delete Title and add "Bulk Pool Transient Temperature Plot (Emergency Core Offload)"
   Figure 9.1-8, delete Title and add "Cooldown Curve for Normal Operation (4 Hours Loss of Pool Cooling)"
   Add "Figure 9.1-20, Fuel Assembly Transfer Limit Verses CCP Temperature"

9. Section 9.1.2.1, page 9.1-2 second paragraph, last sentence, replace "2169" with "3048".

10. Section 9.1.3.1, page 9.1-6, number 13, replace "140°F" with "150°F".

11. Section 9.1.3.1, page 9.1-6, number 19, replace "2169" with "3048".

12. Section 9.1.3.1, page 9.1-6, add the following (Insert A) after item number 19. "20. A full core off-load is designated as a normal evolution."

13. Section 9.1.3.2, page 9.1-6, Delete the first paragraph and second paragraph and replace with the following (Insert B):

    "The spent fuel pool cooling system has been analyzed to remove the decay heat load of up to 3048 fuel assemblies and maintain a bulk pool temperature at or below 150°F using a single train of spent fuel pool cooling. A thermal-hydraulic analysis for these bounding heat loads was performed which provided bulk pool temperature curves for three scenarios, a normal full core off-load heat load (Figure 9.1-7), an emergency full core off-load heat load (Figure 9.1-7A) and a normal operation - loss of fuel pool cooling event (Figure 9.1-8). These curves represent the analysis performed for cooling water to the spent fuel pool cooling heat exchangers (CCP) at 95°F, the upper design temperature limit. For normal and emergency full core off-loads, shorter core off-load times are permitted for lower CCP temperatures as shown in Figure 9.1-20."

    "The thermal-hydraulic analysis assumes that outages for full core off-loads will have a minimum duration of 25 days from reactor shutdown to entry into Mode 4 following core reload and that a maximum of 97 fuel assemblies recently discharge will remain in the spent fuel pool. Refueling outages outside these conditions (less than 25 days or greater that 97 assemblies) will require specific calculations to show that the spent fuel pool decay heat levels are less than or equal to 21.1 x E6 BTU/hr."

    "Cooling for the spent fuel pool consists of two cooling mechanisms. The first is the active cooling provided by the fuel pool heat exchangers. The spent fuel pool water flows from the fuel pool discharge through either of the two fuel pool cooling pumps and through the tube side of either fuel pool cooler, and then returns to the fuel pool. Table 9.1-2 lists the performance characteristics of the fuel pool cooling system. Cooling for the fuel pool coolers is provided by the reactor plant component cooling water system (Section 9.2.2.1). The second mechanism is the passive cooling provide by evaporative cooling from the surface of the pool."

14. Replace Figures 9.1-7, 9.1-7A and 9.1-8 with the attached figures 9.1-7, 9.1-7a and 9.1-8.

15. Section 9.1.3.3, page 9.1-8 & 9, following "These are:" in the fifth paragraph, delete subparagraphs "1.", "2.", "3.", "4." and "Note 1" and replace with the following (Insert C):

    1. Normal full core off-load (maximum bulk pool temperature - 150°F) - the full reactor core (193 assemblies) from the end-of life cycle is off-loaded to the spent fuel pool after one year of operation at full power. The core off-load rate and minimum fuel decay time prior to starting core off-load to the spent fuel pool is dependent on CCP inlet temperature to the Spent Fuel Pool heat exchangers as depicted in Figure 9.1.20.

    2. Emergency full core off-load (maximum bulk pool temperature - 150°F) - the full reactor core (193 assemblies) from the end of life cycle is off-loaded to the spent fuel pool after a previous outage lasting for 10 days with 36 days of operation at full power. The heat load to the spent fuel pool is fully bounded by the heat load for a normal full core off-load. The core off-load rate and minimum fuel decay time prior to starting core off-load to the spent fuel pool is dependent on CCP inlet temperature to the Spent Fuel Pool heat exchangers as depicted in Figure 9.1.20.

    3. Normal Operation/Loss of Pool Cooling - End of life core in the reactor vessel, the latest refueling load (97 assemblies) is in the spent fuel pool with 25 days (600 hours) of decay time. Following a design basis accident with loss of power, cooling to the spent fuel pool is lost for four hours due to diesel loading concerns before it is restored. Cooling to the pool is limited to the evaporative heat loss. Cooling water temperature prior to loss of cooling is assumed to be at the bounding limit of 95°F.

16. Section 9.1.3.3, page 9.1-9, delete second paragraph ("The decay heat load..."), and replace with the following (Insert D):

    "The decay heat in the spent fuel pool is the combination of decay heat from the previously discharged fuel assemblies and the decay heat from the most recently discharged fuel assemblies. The decay heat load for cycle 1 through cycle 5 discharged fuel was modeled on historical fuel discharge data. The projected fuel discharges for cycles 6 through the end of plant life are conservatively modeled at a bounding average batch burnup of 60,000 Mwd/MtU. The projected number of fuel assemblies discharged is conservative in that the most limiting scenario (yielding the largest discharge) was used. The most limiting scenario selected was a half core loading, consisting of alternating fresh fuel batches of 97 and 96 fuel assemblies per cycle. This resulted in discharging 1960 Millstone Unit 3 fuel assemblies at the end of plant life (including the final full core discharge). In addition to the fuel discharged from MP3 operating cycles, an additional heat load from 1088 MP3 fuel assemblies with 60,000 Mwd/MtU burnup and ten years of decay is assumed to be in the pool to conservatively bound discharges of MP1 and MP2 fuel into the MP3 spent fuel pool. Therefore the total number of fuel assemblies considered in the heat load analysis was 3048 Millstone Unit 3 fuel assemblies."

"A single active failure of the spent fuel cooling system was evaluated in the thermal-hydraulic analysis. The failure was assumed to disable the active train of cooling and 30 minutes was assumed in the T/H model to put the standby train into service. The T/H model indicates that with the pool temperature at 150°F, the spent fuel pool bulk temperature would increase to approximately 155°F before cooling was restored.

17. Section 9.1.3.3, page 9.1-9, third paragraph ("Following a design basis..."), delete from to the third sentence ("If possible, pool cooling...") to the end of the paragraph and insert the following (Insert E):

    "The loss of cooling to the spent fuel pool was evaluated for normal plant operation in Case 3 above where pool temperature rose to a maximum of 148.8°F. An additional analysis, which is outside the design basis of the spent fuel pool cooling system, was conducted as an input to the structural analysis of the spent fuel pool for the loss of spent fuel pool cooling during a full core off-load. The most limiting case occurs when the pool is at 150°F with the highest heat load in the pool. For this case, a pool temperature of 200°F would be reached after 4.41 hours. For all cases of a loss of pool cooling, the pool temperature is maintained below 200°F when pool cooling is restored after four hours."

18. Table 9.1-2, Replace table with the attached table.

19. Add Figure 9.1-20

20. Section 9.4.2.1, Table in Item 5, rename sub-items "a", "b" and "c" to read "b", "c" and "d". Insert F as new sub-item "a".

    Insert F: "a. Pool water temperature is greater than 140°F"    98°F    85°F

TABLE 9.1-2

PERFORMANCE CHARACTERISTICS OF
THE FUEL POOL COOLING SYSTEM
(ONE FUEL POOL COOLER OPERATING)

| Operating Condition | Full Core Offload | Emergency Full Core Offload | Normal Plant Operation |
|---|---|---|---|
| Heat Load BTU/hr | 95°F CCP $36.08 \times 10^6$<br>90°F CCP $39.19 \times 10^6$<br>85°F CCP $42.30 \times 10^6$<br>80°F CCP $45.41 \times 10^6$ | Bounded by Full Core Offload | $21.1 \times 10^6$ |
| Required Duty of One Fuel Pool Cooler BTU/hr | 95°F CCP $34.20 \times 10^6$<br>90°F CCP $37.31 \times 10^6$<br>85°F CCP $40.42 \times 10^6$<br>80°F CCP $43.53 \times 10^6$ | Bounded by Full Core Offload | $20.37 \times 10^6$ |
| Heat Removal by Evaporation BTU/hr | $1.88 \times 10^6$ | $1.88 \times 10^6$ | $0.73 \times 10^6$ |
| Maximum Temperature Long Term | 150°F | 150°F | 127.6°F |
| Maximum Peak Temperature Short Term | 155.7°F | Bounded by Full Core Offload | N/A |
| Maximum Peak Temperature (Accidents) | 200°F in 4.41 hours (Condition is outside Design Basis) | Bounded by Full Core Offload | 148.8°F - Following a DBA with a four hour loss of pool cooling |
| Design Limits | Maximum Long Term Temperature (Structural Requirement): 150°F<br>Maximum Short Term Temperature (Structural Requirement): 200°F<br>Maximum Temperature Loss of Pool Cooling: 155.7°F | | |
| Flow Rates | Tube Side - Fuel Pool Water (SFC): 3,500 gpm<br>Shell Side - Reactor Plant Component Cooling Water (CCP): 1,800 gpm | | |
| Cooling Water Temperatures | Reactor Plant Component Cooling (CCP) (Design): 95°F<br>Service Water (Cools CCP) (Design): 75°F | | |

TABLE 1.3-15

COMPARISON OF OTHER REACTOR PLANT SYSTEMS

| Systems with Components | Operating Parameters: Millstone 3 | Operating Parameters: North Anna 1 and 2 |
|---|---|---|
| **Fuel Pool Cooling and Purification System (Section 9.1.3)** | | |
| Fuel Pool Cooling Pumps: | | |
| Number | 2 | 2 |
| Design capacity (gpm) | 3,500 | 2,750 |
| Design total head (ft) | 92 | 80 |
| Fuel Pool Coolers: | | |
| Number | 2 | 2 |
| Duty per heat exchanger (Btu/hr) | 27,700,000 | 56,800,000 |
| Fuel pool cooling flow (gpm) | 3,500 | 2,750 |
| Component cooling flow (gpm) | 1,800 | 3,350 |
| Number of cores cooled | ~~1-1/3~~ 15½ (3048 Fuel Assemblies) | 1-1/3 |
| Fuel pool temperature, normal (°F) | ~~140~~ 150 | 140 |
| **Reactor Plant Component Cooling System (Section 9.2.2.1)** | | |
| Reactor Plant Component Cooling Pumps: | | |
| Number | 3 | 4 |
| Design capacity (gpm) | 8,100 | 8,000 |
| Design total head (ft) | 284 | 190 |

TABLE 1.9-1

SUMMARY OF DIFFERENCES FROM SRP

| SRP Section | Specific SRP Acceptance Criteria | Summary Description of Difference | Corresponding FSAR Section |
|---|---|---|---|
| 430.3 9.1.2 (Rev. 3) | III.2.e - Evaluation of lighter load drops at maximum heights. | This evaluation has not been performed. | 9.1.2.3 |
| 9.1.3 (Rev. 1) | II.1.d (4) - BTP ASB 9-2 decay heat removal. | ~~Decay heat removal is based on Westinghouse generated curves, not BTP ASB 9-2.~~ Decay heat removal is based on ~~ORIGEN2~~ DECOR (based on ORIGEN2) computer code and credit for evaporative cooling instead of BTP ASB 9-2. | 9.1.3.2 |
| Insert 9.1.3 9.1.4 (Rev. 2) | III.6 - Evaluation of lighter load drops at maximum heights. | This evaluation has not been performed. | 9.1.4.1 |
| 9.2.1 (Rev. 2) | III.3.d - Location of radiation monitors. | No manual valve in series with motor operated valve. | 9.2.1 |
| 9.2.2 (Rev. 1) | II.3.e - Loss-of-cool ant test for reactor-coolant-pumps. | Reactor-coolant-pumps have not been tested for the 20-minute time requirement. | 9.2.2 |
| 410.21 9.4.1 (Rev. 2) | II.4 - Compliance to Regulatory Guide 1.95. | The chlorine detectors are not Seismic Category I. | 9.4.1.3 |
| 9.4.5 (Rev. 2) | II.4 - Protection from dust accumulation. | The bottoms of the fresh air intakes are not all located at least 20 feet above grade elevation. | 9.4.5 |
| | II.5 - Detection and Control of airborne contamination leakage from the system. | Only normal building ventilation is monitored. | 9.4.5.2 |
| | III.3.b - Tornado protection. | No protection of duct-work from negative pressure due to tornado. | 9.4.5.4 |

B. Justification for difference from SRP

Electrical interlocks and load paths prevent any load from being carried over the spent fuel pool with the new fuel handling crane. Spent fuel bridge and hoist only carries fuel assemblies at their normal lifting height.

SRP 9.1.3

SRP TITLE: SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

A. Actual difference between FSAR and SRP

the DECOR computer code (based on ORIGEN2) and credit for evaporative cooling,

1. Decay heat removal is based on ~~Westinghouse generated curves~~, not BTP ASB 9-2, as required by SRP 9.1.3, Paragraph II.1.d(4). ||

Insert 2. and 3.

B. Justification for difference from SRP

the DECOR computer code (based on ORIGEN2) and credit for evaporative cooling in order to get a more accurate value of decay heat loads ||

1. Decay heat removal analysis is based on ~~current NSSS curves utilized for the fuel that will actually be supplied to Millstone 3~~ and is therefore more appropriate than the analysis recommended in BTP ASB 9-2.

2. Insert 2. and 3.

SRP 9.1.4

SRP TITLE: LIGHT LOAD HANDLING SYSTEM (RELATED TO REFUELING)

A. Actual difference between FSAR and SRP

SRP 9.1.4, Paragraph III.6, requires an evaluation of lighter load drops at maximum heights. This evaluation has not been performed.

B. Justification for difference from SRP

Electrical interlocks and load paths prevent carrying any load over the spent fuel pool with the new fuel handling crane. Spent fuel bridge and hoist only carry fuel assemblies at their normal lifting height.

SRP 9.2.1

SRP TITLE: STATION SERVICE WATER SYSTEM (NUCLEAR SERVICE COOLING WATER SYSTEM)

A. Actual difference between FSAR and SRP

SRP 9.2.1, Paragraph III.3.d, requires that radiation monitors be located on system discharge, and at components susceptible to the leakage, and that these components can be isolated by one automatic and one manual valve in series. There are motor-operated valves at the inlet and

TABLE 3.6-3

MODERATE-ENERGY SYSTEMS OUTSIDE CONTAINMENT REMOTE FROM ESSENTIAL SYSTEMS, COMPONENTS, AND STRUCTURES

| System | Location* | Maximum Operating Conditions Temperature (°F) | Pressure (Psig) |
|---|---|---|---|
| Fire Protection - Water | AB, ABR, FB, SB, TA, TB, WDB | 85 | 108 |
| Fuel Pool Cooling and Purification | AB, FB | ~~140~~ 150 | 130 |
| Generator Hydrogen and $CO_2$ | TB | 100 | 0 |
| Hydrogen Gas | AB, GSA, TB, YA | 90 | 100 |
| Instrument Air | AB, ABR, ESB, SB, TB, WDB | 120 | 100 |
| Nitrogen Gas | AB, ABR, NSP, SB | 90 | 5 |
| Office Building Chilled Water | | | |
| Oxygen Gas | | | |
| Primary Grade Water | AB, FB, PWH, ADB, YA | 45 | 100 |
| Quench Spray | ESB, YA | 140 | 160 |
| Radioactive Solid Waste | AB, FB | 100 | 140 |
| Reactor Plant Aerated Drains | AB, ESB, FB, WDB | 170 | 61 |
| Reactor Plant Gaseous Drains | AB | 165 | 104 |
| Reactor Plant Aerated Vents | AB, WDB | 120 | 0 |
| Reactor Plant Gaseous Vents | AB, WDB | 120 | 3 |

1. Yard Structures

<u>Vacuum Priming Pumphouse</u> (Nonsafety Related)

The vacuum priming pumphouse is a reinforced concrete structure located on top of the outfall structure. The area is approximately 40 ft x 35 ft with a floor 0 feet-6 inches above grade. A roof with insulation and 4-ply asphalt and gravel is provided at approximately 17 feet above grade.

2. Miscellaneous Yard Tankage

Boron recovery tanks, primary grade water tanks, demineralized water storage tank, refueling water storage tank, boron and waste test tanks, condensate storage tank, condensate surge tank and water treatment storage tanks are located on concrete pads with oil sand cushion 0 ft-6 inches above grade. The demineralized water storage tank is protected by 2 ft-0 inch thick reinforced concrete walls and roof. The boron recovery tanks are enclosed in a concrete and steel structure.

3. Electrical/Conduit Manholes

Electrical manholes are reinforced concrete structures constructed below grade with access through manhole covers at grade.

(10187) All other nonsafety related structures are located such that their failure does not damage safety related systems, structures, or components.

3.8.4.2 Applicable Codes, Standards, and Specifications

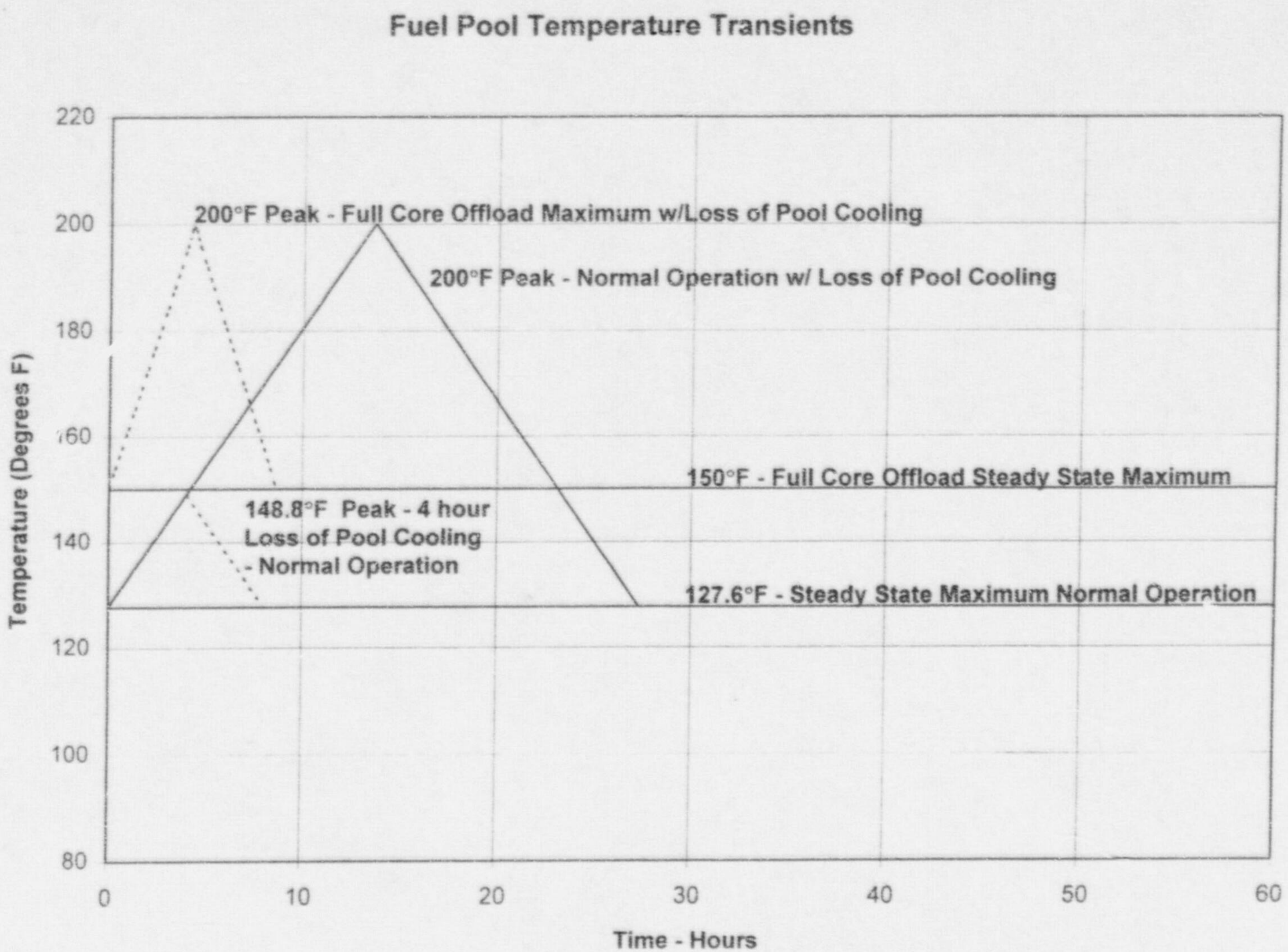
Codes, standards, specifications, and NRC regulatory guides used in establishing design methods and material properties for Seismic Category I concrete and steel structures other than the containment are given in Section 3.8.1.2.

3.8.4.3 Loads and Loading Combinations

All Seismic Category I structures other than the containment structure mat, shell, and dome are designed for the loads and load combinations in Table 3.8-3. Section 3.8.4.5 describes allowable stress levels.

For the spent fuel pool, the effects of loads imparted to the structure by the spent fuel racks as well as the effects of hydrostatic and seismically induced hydrodynamic loads are considered in the design. ~~The spent fuel pool walls and mat are also designed for thermal effects based on temperatures indicated in Figures 3.8-79 and 3.8-80~~. Insert The spent fuel pool (91-21 6/92) walls and mat were also investigated for the revised thermal transient effects due to the storage of higher enrichment fuels as shown in Figure 3.8-81. Utilizing the loads and load combinations in Table 3.8-3, the allowable stress levels described in Section 3.8.4.5 were satisfied.

# Millstone Unit 3




**FSAR Figure 3.8.82**
Fuel Pool Temperature Transients - Full Core Offload

# ENVIRONMENTAL DESIGN CONDITIONS

Attachment 1
Appendix E
Page 1 of 10

13 4/85 **FUEL BUILDING** - Zone: FB-01 (Note 5) Elevation 11 ft-0 in. (Pipe Tunnel Area) See Attachment 2, Appendix E, Page 1 of 2

## Normal Environment (40-yr life)

Temperature (Note 1) INSERT C

123

Range: 65 - 120°F
NMA: 85°F

13

113 MNE: 110°F — 104°F, 110°F (113) for 8 hr, 10 cycles/yr

123 MAE: 120°F — 104°F, 120°F (123) for 4 hr, 1 cycle/40 yr

Pressure: Atmosphere

Relative Humidity: 10 - 100%

Radiation Dose (rads) 40-yr life: $6.0 \times 10^3$

## One Time Accident Environment due to Separate Events:

(A) DBA Event (Note 1)

Temperature: 85°F, 120°F for 48 hr

Pressure: Atmosphere

Relative Humidity: 100% for 48 hr

(B) HELB Event (Note 2)

Max. Temp: 105°F (Note 4)
Max. Pressure: 16.3 psia
Relative Humidity: 100% for 48 hr
Profile FB-01, See Appendix E, Page 7 of 10

(C) Fuel Handling Accident (Note 3)

13 Accident Radiation Dose (rads): $5.3 \times 10^4$

Radiation Dose (rads) - 40-yr life plus accident: $5.9 \times 10^4$

Attachment 1
Appendix E
Page 2 of 10

**FUEL BUILDING**

Zone: FB-02 (Note 5) — Elevation 24 ft-6 in. (G-H/50.6-52.8) Fuel Pool Cooling Pump Area Spent Fuel Pool Cooler and Decontamination Rooms
See Attachment 2, Appendix E, Pages 1 and 2 of 2 13 4/85

Normal Environment (40-yr life)

INSERT C

Temperature (Note 1)

Range: 65 - 120°F 123
NMA: 85°F

MNE: 110°F 113 — 104°F, 110°F (113) for 8 hr, 10 cycles/yr

MAE: 120°F 123 — 104°F, 120°F (123) for 4 hr, 1 cycle/40 yr

13

Pressure: Atmosphere

Relative Humidity: 10 - 100%

Radiation Dose (rads) 40-yr life: $6.0 \times 10^3$

One Time Accident Environments due to Separate Events:

(A) DBA Event (Note 1)

Temperature: 85°F, 120°F for 48 hr

Pressure: Atmosphere

Relative Humidity: 100% for 48 hr

(B) HELB Event (Note 2)

Max. Temp: 117°F (Note 4)
Max. Pressure: 16.3 psia
Relative Humidity: 100% for 48 hr
Profile FB-02, See Appendix E, Page 8 of 10

(C) Fuel Handling Accident (Note 3)

Accident Radiation Dose (rads): $5.3 \times 10^4$

Radiation Dose (rads) - 40-yr life plus accident: $5.9 \times 10^4$ 13

## ENVIRONMENTAL DESIGN CONDITIONS

Attachment 1
Appendix E
Page 3 of 10

13 4/85 FUEL BUILDING - Zone: FB-04 (Note 5)
Elevation 24 ft-6 in.
New Fuel Pool and Receiving Areas
(G-H/52.8-53.8)
See Attachment 2, Appendix E, Pages 1 and 2 of 2

Normal Environment (40-yr life)

Temperature (These temperatures are based on a maximum spent fuel pool temperature of 140 °F.) 150 INSERT B

13 Range: 65 - 120 °F 123
NMA: 85 °F

13 MNE: 110 °F 113 — 104 °F, 110 °F 113, 8 hr, 10 cycles/yr

MAE: 120 °F 123 — 104 °F, 120 °F 125, 4 hr, 1 cycle/40 yr

Pressure: Atmosphere

Relative Humidity: 10 - 100%

Radiation Dose (rads) 40-yr life: $6.0 \times 10^3$

One Time Accident Environments due to Separate Events:

(A) DBA Event (Note 1)

Temperature: 85 °F, 120 °F, 48 hr

Pressure: Atmosphere

Relative Humidity: 100% for 48 hr

(B) HELB Event (Note 2)

Max. Temp: 155 °F (Note 4)
Max. Pressure: 16.3 psia
Relative Humidity: 100% for 48 hr
Profile FB-04, See Appendix E, Page 10 of 10

(C) Fuel Handling Accident (Note 3)

3 Accident Radiation Dose (rads): $5.3 \times 10^4$

Radiation Dose (rads) 40-yr life plus accident: $5.9 \times 10^4$

# ENVIRONMENTAL DESIGN CONDITIONS

Attachment 1
Appendix E
Page 4 of 10

FUEL BUILDING - Elevation 52 ft-4 in. and above
Zone: FB-03 See Attachment 2, Appendix E, Page 2 of 2
(Note 5)

13 (4/85)

Normal Environment (40-yr life)

Temperature (These temperatures are based on a maximum spent fuel pool temperature of 140 °F.) 150 INSERT B

Range: 65 - 110 °F 116
NMA: 90 °F

(13)

MNE: 102 °F 106 — 95 °F | 102 °F (106) 8 hr | 10 cycles/yr

MAE: 110 °F 116 — 95 °F | 110 °F (116) 4 hr | 1 cycle/40 yr

(13)

Pressure: Atmosphere

Relative Humidity: 10 - 100%

Radiation Dose (rads) 40-yr life: 6.0 x $10^3$

One Time Accident Environments due to Separate Events:

Admin 4/96

(A) DBA Event (Note 1)

Temperature: 85 °F | 120 °F 48 hr |

Pressure: Atmosphere

Relative Humidity: 100% for 48 hr

(B) HELB Event (Note 2)

Max. Temp: 125 °F (Note 4)
Max. Pressure: 16.3 psia
Relative Humidity: 100% for 48 hr
Profile FB-03, See Appendix E, Page 9 of 10

Admin 4/96

(C) Fuel Handling Accident (Note 3)

Accident Radiation Dose (rads): 5.3 x $10^4$

Radiation Dose (rads) - 40-yr life plus accident: 5.9 x $10^4$

(13)



Admin 4/96

FUEL BUILDING - Elevation 24 ft-6 in. Containment Tool
13 (4/85) Zone: FB-05 Storage Room (G.6-H/50.6-51.2)
(Note 5) See Attachment 2, Appendix E, Page 1 of 2

Normal Environment (40-yr life)

INSERT C

Temperature: (Note 1)

(13) Range: 65 - 120°F 123
NMA: 85°F

(13) MNE: 110°F 113 — 104°F, 110°F 113 for 8 hr, 10 cycles/yr

MAE: 120°F 123 — 104°F, 120°F 123 for 4 hr, 1 cycle/40 yr

Pressure: Atmosphere

Relative Humidity: 10 - 100%

Radiation Dose (rads) 40-yr life: $6.0 \times 10^3$

93-44 (6/94) Admin 4/96 One Time Accident Environment due to Separate Events: (Note 6)

(A) DBA Event (Note 1)

Temperature: 85°F, 120°F for 48 hr

Pressure: Atmosphere

Relative Humidity: 100% for 48 hr

(93-44) Admin 4/96 (B) HELB Event (Not Applicable: Mild Environment)

(C) Fuel Handling Accident (Note 3)

Admin 4/96 Accident Radiation Dose (rads): $< 1.0 \times 10^3$

(13) Radiation Dose (rads) - 40-yr life plus accident: $< 7.0 \times 10^3$

Attachment 1
Appendix E
Page 6 of 10

Notes on Fuel Building Environments

1. Following a postulated DBA, if component cooling water is not supplied to the fuel pool coolers due to diesel loading limitations, pool temperature increases to 200°F in ~~20~~ 13.7 hours. This results in a one-time accident condition approximated by the profile shown. This assumes initial pool temperature of 150°F and additional pool cooldown time subsequent to component cooling reinitiation. INSERT D 3/88

2. Due to High Energy Line Break (HELB) of hot water supply line.

3. Fuel handling accident. Does not include pipe tunnel area or equipment in contact with or in close proximity to spent fuel assemblies.

4. Equipment must be thermally stabilized at ambient temperature of 95°F prior to the increases to Tmax and Pmax.

5. Engineer is to determine applicable environmental conditions and zone index codes based upon specific equipment locations and operability requirements.

6. Zone FB-05 boundaries have been modified to provide an HELB seal. Parameters shown are not significantly more severe than the normal environment, therefore, this zone is considered a mild environment. 93-44 (6/94)

# CHAPTER 9

## LIST OF FIGURES



ADD

The racks are designated ANS Safety Class 3 and Seismic Category I and are designed to withstand normal and postulated dead loads, live loads, and loads caused by the operating basis earthquakes and safe shutdown earthquake events.

The design of the racks is such that $K_{eff}$ remains less than or equal to 0.95 under all conditions, including fuel-handling accidents and the optimum moderation configuration. Due to the use of fuel barriers and the close spacing of the cells, it is impossible to insert a fuel assembly in other than design locations or between the rack periphery and the pool wall.

The racks are also designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of 5 feet over the top of the racks. The fuel storage racks can withstand an uplift force equal to 2000 pounds.

All materials used in construction are compatible with the fuel building/vault environment and all surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. All the materials are corrosion resistant and do not contaminate the fuel assemblies or vault environment.

### 9.1.2 Spent Fuel Storage

#### 9.1.2.1 Design Bases

The spent fuel pool, located in the fuel building, is designed to accommodate fuel racks (Figure 9.1-1) that store spent fuel assemblies. At the time of initial operation, installed capacity was at least one and one-third cores.

The spent fuel is stored in racks which are located under water in the spent fuel pool. There are 756 fuel storage locations in 21 storage racks. Each rack consists of cells welded to a grid base and welded together at the top through an upper grid to form an integral structure (Figure 9.1-1). The vertical corners of adjacent cells are also welded
(91-21 together to form an integral structure. The spent fuel pool has the heat load design
6/92) capacity for ~~2160~~ 3048 fuel assemblies (Section 9.1.3).

The rack arrays (Figures 9.1-2 and 9.1-3) have a center-to-center spacing of 10.35 inches. Each storage cell incorporates a neutron absorber and is composed of boron carbide in a homogeneous stable matrix. This material is encapsulated in stainless steel for support but is not sealed as it is compatible with the pool environment. The spacing and the design of the racks are such that there is a 95 percent probability that the effective multiplication factor, including uncertainties, does not exceed 0.95 at a 95 percent confidence level.

97-109 For the storage of fuel assemblies with nominal enrichment levels between 3.85 and
4/97 5.0 w/o $U_{235}$, a regionalized fuel storage/pool configuration is implemented as follows:

- High enriched, low (or no) burnup fuel is stored in Region I in a 3-out-of-4 array with the fourth storage location blocked. Up to a maximum of 100 storage
(11/89) locations will be blocked as shown in Figure 9.1-2A.
- Low enriched, high burnup fuel is stored in Region II in a 4-out-of-4 array due to reactivity credit for burnup being taken into account as permitted by NRC Regulatory Guide 1.13.

9.1.3.1 (continued)

13. The temperature of the fuel pool water is maintained at or below ~~140~~ 150°F for the normal operating condition of the spent fuel pool.

14. The temperature of the fuel pool water should not exceed a maximum temperature of 150°F for any long-term period. The maximum peak temperature the spent fuel pool water can reach is 200°F.

15. Purity and clarity of the refueling cavity and fuel pool water is maintained to permit observation of fuel assembly handling during refueling operations.

16. Filtration and ion exchange capability are provided to remove suspended and dissolved radionuclides to allow access to required areas.

17. The fuel pool cooling system and the service water makeup lines are safety related, Seismic Category I, and are designated SC-3 and designed to the requirements of ASME III, Class 3.

18. The purification system is not safety related and is designated nonnuclear safety (NNS).

19. The spent fuel pool cooling and purification system is capable of handling the accumulated decay heat from ~~2169~~ 3048 spent fuel assemblies, which includes a full core offload of 193 fresh fuel assemblies.

INSERT "A"

9.1.3.2 System Description

INSERT "B"

~~The spent fuel pool has been designed to hold up to 2169 fuel assemblies. A heat load calculation using this loading was performed which provided decay heat load curves for three scenaries, a normal refueling heat load (Figure 9.1-7), end-of-cycle full-core off-load (Figure 9.1-7A), and an emergency full core offload heat load (Figure 9.1-8).~~

~~The spent fuel pool water flows from the fuel pool discharge through either of the two fuel pool cooling pumps and through the tube side of either fuel pool cooler, and then returns to the fuel pool. Table 9.1-2 lists the performance characteristics of the fuel pool cooling system. Cooling for the fuel pool coolers is provided by the reactor plant component cooling water system (Section 9.2.2.1).~~

The purification system consists of two purification pumps, two purification prefilters, one coarse filter, one purification demineralizer, and one postfilter. This equipment is not safety related.

The purification system provides means for filtering and demineralizing the following areas:

1. The fuel pool water to improve optical clarity for ease of underwater fuel handling and to reduce radioactive contamination in the water

2. The reactor cavity water during a refueling operation to improve optical clarity for ease of underwater fuel handling, and to reduce radioactive contamination in the water

arrangement be needed, a temporary flanged spool piece can be inserted in the line to enable one of the fuel pool purification pumps to pump the water within the spent fuel shipping cask storage area either through the prefilters or through the prefilters, demineralizer, and postfilter to the boron recovery tanks (Section 9.3.5). Administrative procedures are followed to assure that the cask storage area gate is inserted in the transfer slot in the wall separating the fuel pool from the spent fuel shipping cask storage area before pumping commences. However, the design of the gate is such that even with the gate open, the fuel pool cannot be drained below the top of the active fuel region of the fuel assemblies.

Piping, valves, and components of this system making contact with the fuel pool water are austenitic stainless steel which is corrosion- resistant to the boric acid solution.

A sample connection is provided downstream of the fuel pool demineralizer for sample removal to check the gross activity, particulate matter, boric acid concentration, and component performance.

### 9.1.3.3 Safety Evaluation

Two full-size fuel pool cooling pumps and two full-size fuel pool coolers are provided to ensure 100-percent redundant cooling capacity. This portion of the system is Seismic Category I and Safety Class 3. The Seismic Category I cooling portion of the fuel pool cooling and purification system is independent of the nonseismic purification portion. Failure of the purification portion in an earthquake does not affect the operation of the cooling trains.

Each pipe which enters the fuel pool has either a 1/2 inch vent hole drilled into the pipe to act as an anti-siphoning device or terminates at an elevation above these vent holes. These provisions prevent siphoning of the fuel pool water to uncover the spent fuel (see Figure 9.1-6).

One pump and one cooler are sufficient to maintain the pool temperatures as indicated in Table 9.1-2.

(96-22) 7/96 An evaluation of the capabilities of the spent fuel pool cooling system has been performed for normal and abnormal conditions.

(91-21) 6/92 The decay heat loads were calculated for a number of pool operating conditions at the end of pool life. These are:

INSERT C

(96-22) 1. ~~Normal refueling (2.177 x $10^7$ Btu/hr, Figure 9.1-7)--one refueling load core (1/2 core, 96 assemblies) transferred from the reactor vessel to the spent fuel pool starting eleven days after shutdown.~~

(96-22) 2. ~~Normal plant operation (1.859 x $10^7$ Btu/hr, Figure 9.1-7 at 600 hrs)--one refueling load (1/2 core) in the spent fuel pool after 25 days of decay time. This conservatively models the decay heat load in the spent fuel pool during normal plant operation (reactor coolant system operating, turbine on line).~~

3. End-of-cycle full-core off-load ($3.479 \times 10^7$ Btu/hr, Figure 9.1-7A)--one full core (193 assemblies) off-load after one year of power operation (note 1). Fuel transfer from the reactor starts eleven days after shutdown. (96-22 7/96)

4. Emergency full-core off-load ($3.505 \times 10^7$ Btu/hr, Figure 9.1-8)--a full-core off-load commencing 36 days after the start of the previous refueling (note 1). This corresponds to an "abnormal maximum heat load" as defined in SRP 9.1.3 and is used as the decay heat load for all full-core off-loads occurring less than one year after a previous refueling shutdown. (96-22)

Note 1: A maximum of six full-core off-loads are assumed to occur over the life of the plant (Conditions 3 and 4 combined).

To determine the decay heat load, the fuel bundles were analyzed as being transferred to the spent fuel pool at an average rate of three bundles per hour over the time it takes to off-load the assumed fuel load. (91-21 6/92)

INSERT D

The decay heat load for the older spent fuel in the pool was modeled as one refueling batch 1 year old, one refueling batch at 3 years old, and the remaining refueling batches at 18-month intervals for Cases 1, 2, and 3 above. For Case 4, the older spent fuel was modeled as one refueling batch 18 months old, and the remaining refueling batches at 18-month intervals over the life of the pool. The 36-day-old and fresh fuel heat loads were added to this baseline heat load. For conservatism, the fuel burnup rates used were consistent with 24-month operating cycles.

Following a design basis accident with loss of power, the reactor plant component cooling water system is not available to cool the spent fuel pool coolers until approximately 4 hours after the accident at which time cooling will be restored. Power from the emergency generators is not immediately available due to loading considerations. (97-541 1/98) However, in the event pool cooling is not available at this time, a loss of cooling evaluation has been performed which shows that it would take at least 15 hours before the spent fuel pool would reach its design temperature of 200°F. This provides sufficient time to initiate pool cooling. Once the cooling is restarted, the temperature decreases to 150°F in less than 6 hours. This scenario forms one of the bases for the qualification of the pool structure (Section 3.8.4.3) and conservatively envelops expected pool transients. (97-541)

INSERT E

For a safety grade cold shutdown with a normal plant operation spent fuel pool heat load up to and including Plant Cycle 20, the temperature of the fuel pool water is maintained at or below 140°F. Plant Cycles 20 through the end of plant life could result in short-term fuel pool water temperature transients above 140°F, but less than 150°F. This condition results from higher reactor plant component cooling water supply temperature of approximately 113°F which is permitted during a safety grade cold shutdown (see Section 9.2.2.1.2). 98-45 3/98

Redundant safety grade fuel pool temperature indication is provided on the main control board. Redundant safety class 3 level switches are connected to the fuel pool which alarm in the control room. They are set to provide indication before the water level falls below 23 feet above the top of the fuel racks. Piping penetration are at least 11 feet above the top of the spent fuel so that failure of inlets, outlets or accidental piping leaks cannot reduce the water below this level. (97-374 10/97)

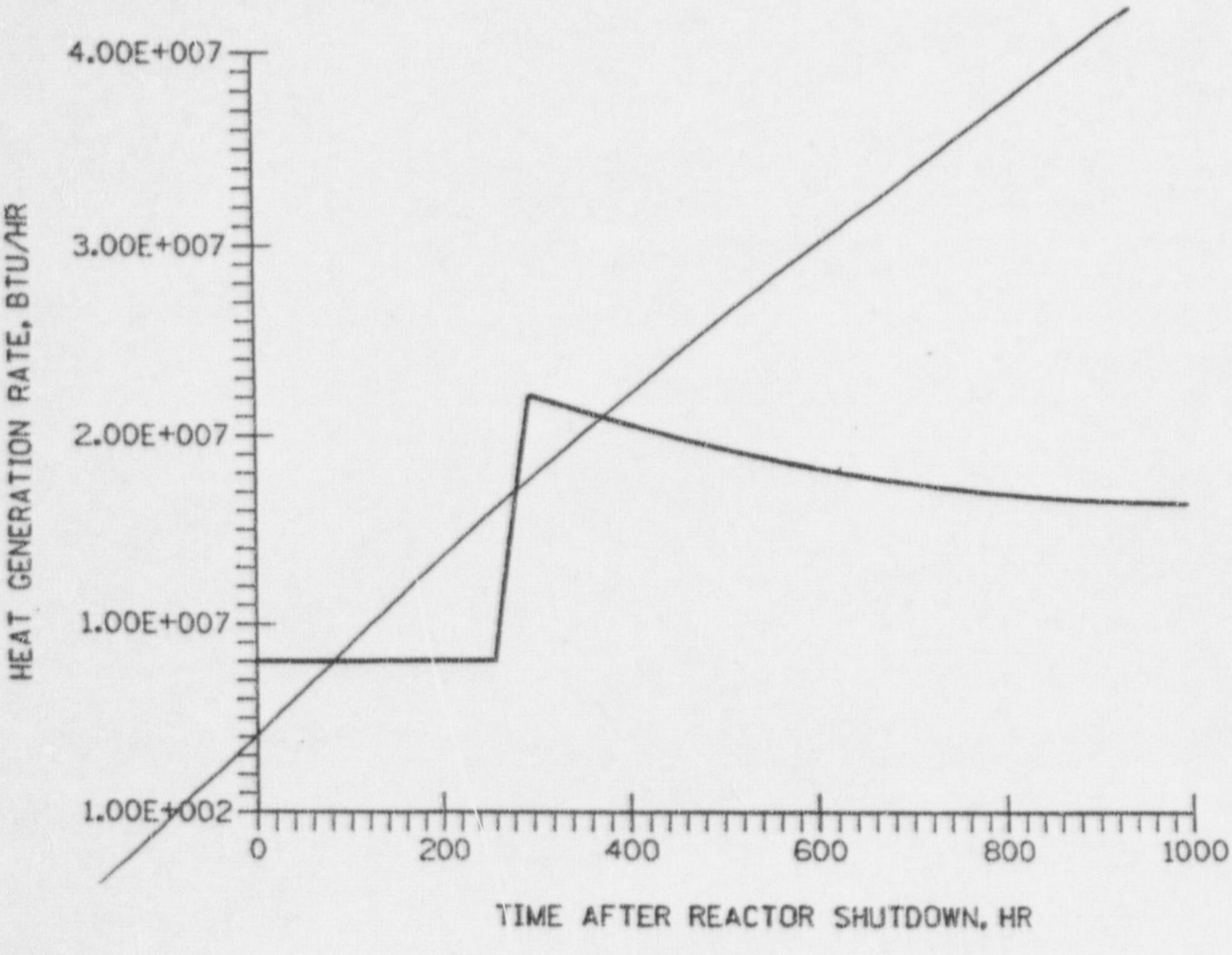



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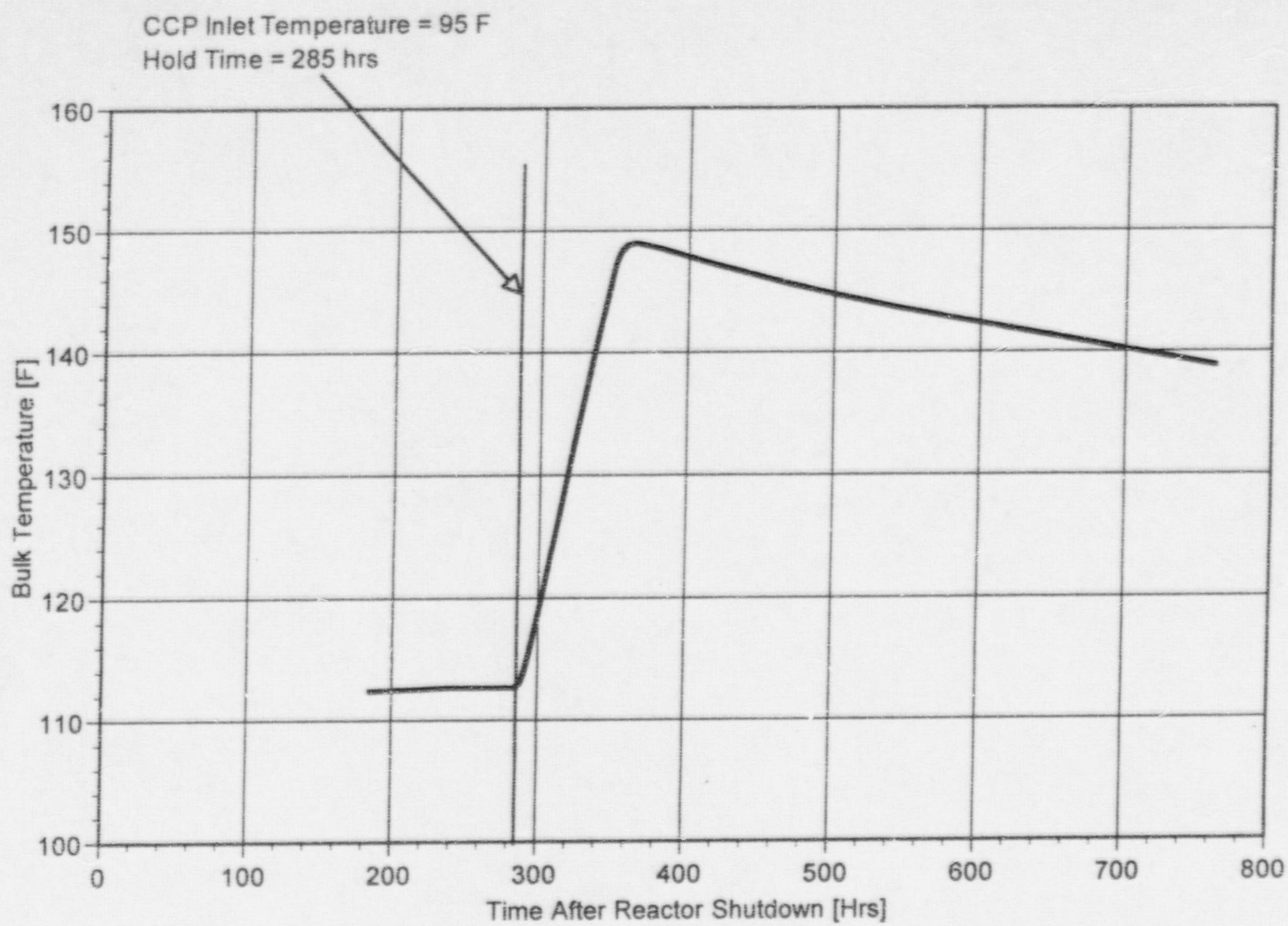
FIGURE 9.1-7
NORMAL REFUELING

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT




: BULK POOL TRANSIENT TEMPERATURE PLOT (FULL CORE OFFLOAD)

FIGURE 9.1-7

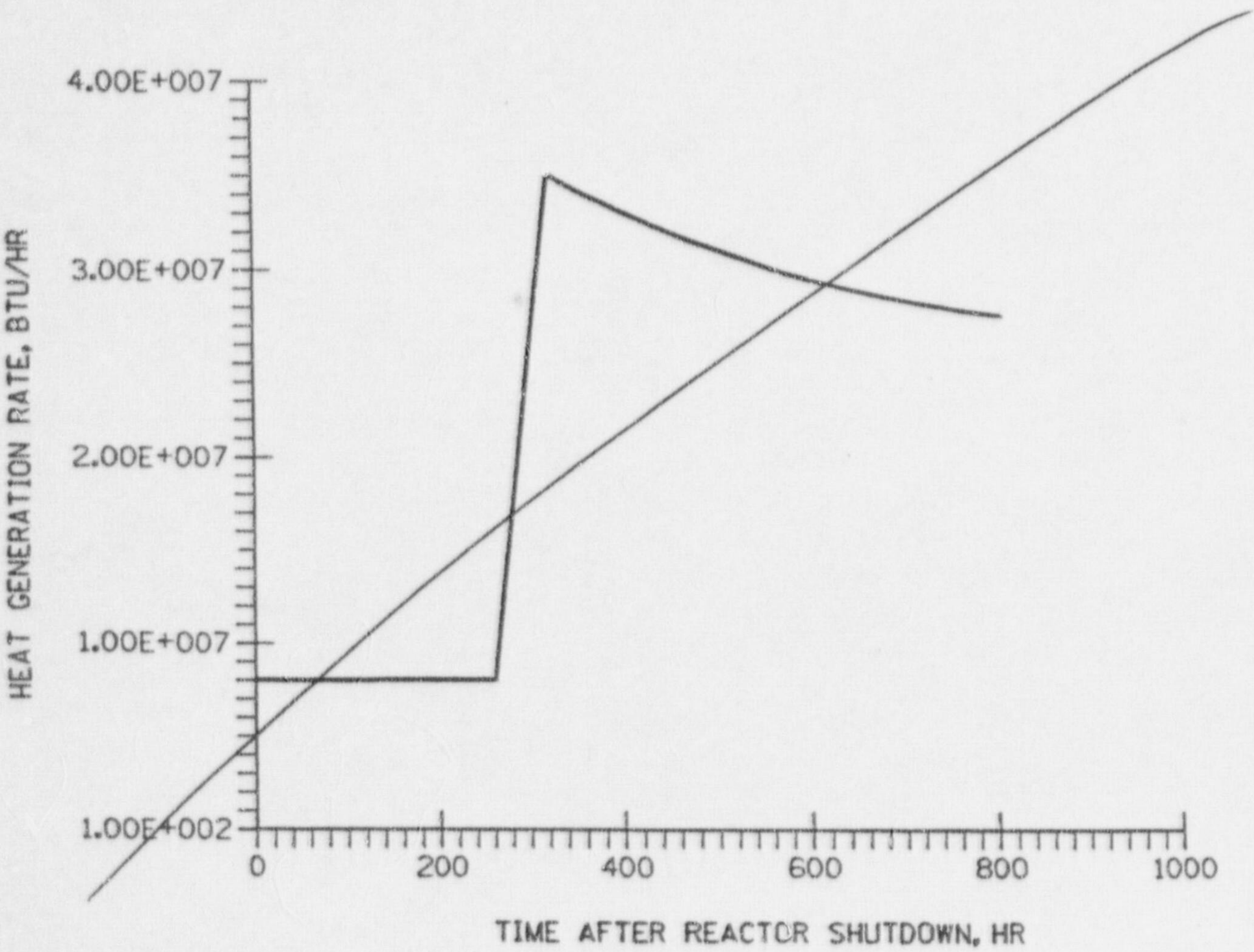



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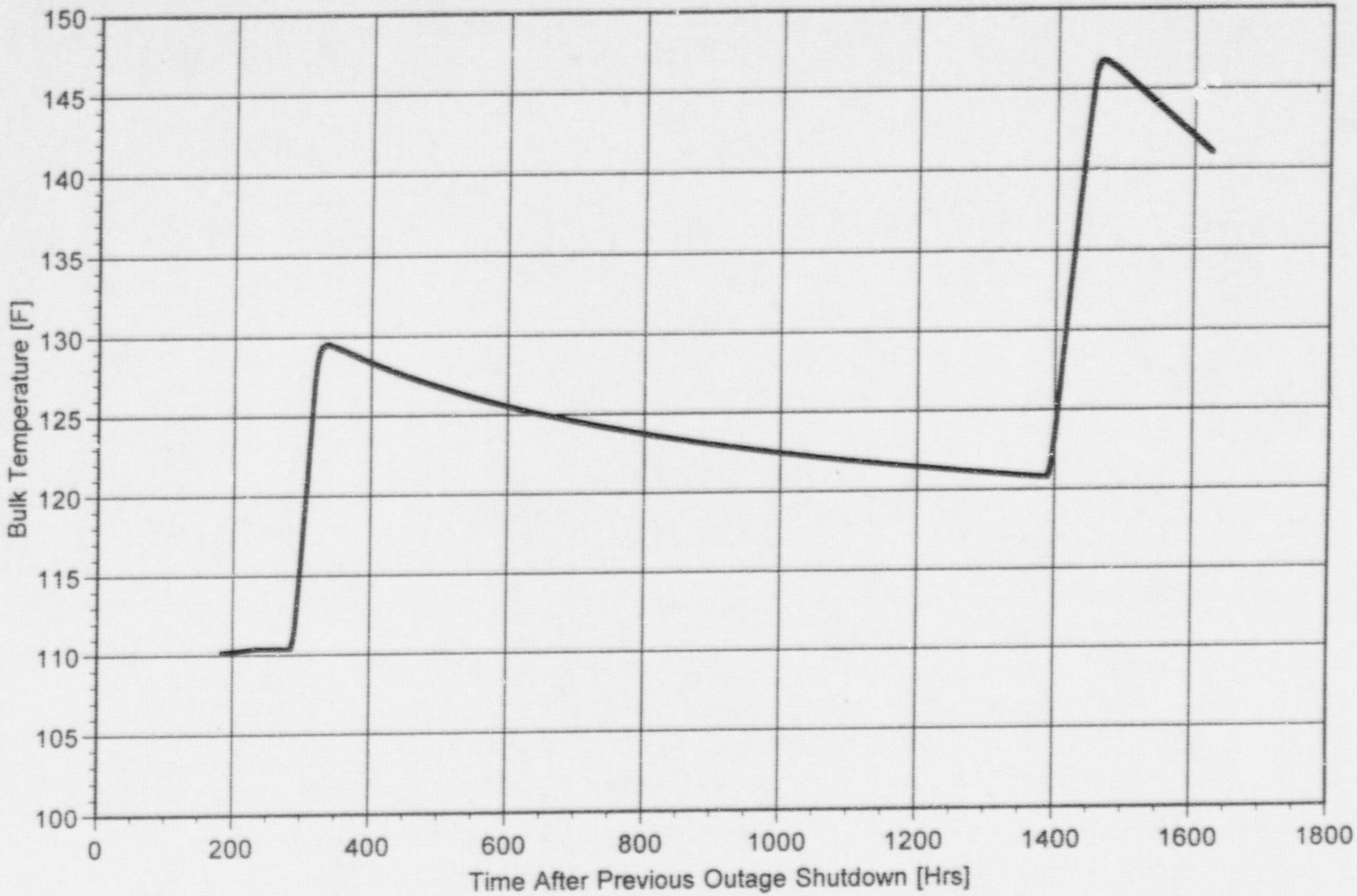
FIGURE 9.1-7A
END OF CYCLE FULL CORE OFFLOAD

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT




BULK POOL TRANSIENT TEMPERATURE PLOT (EMERGENCY CORE OFFLOAD)

FIGURE 9.1-7a

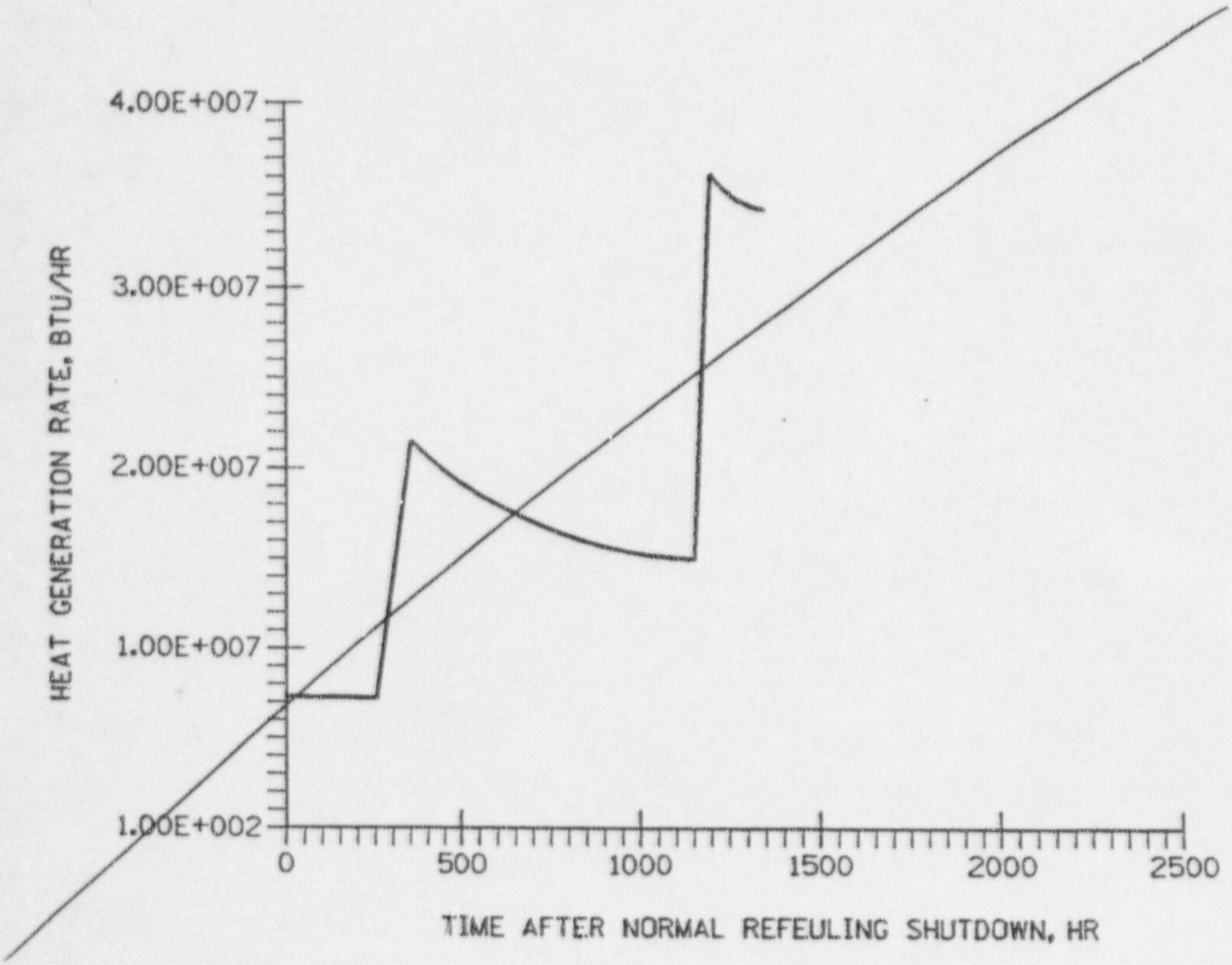



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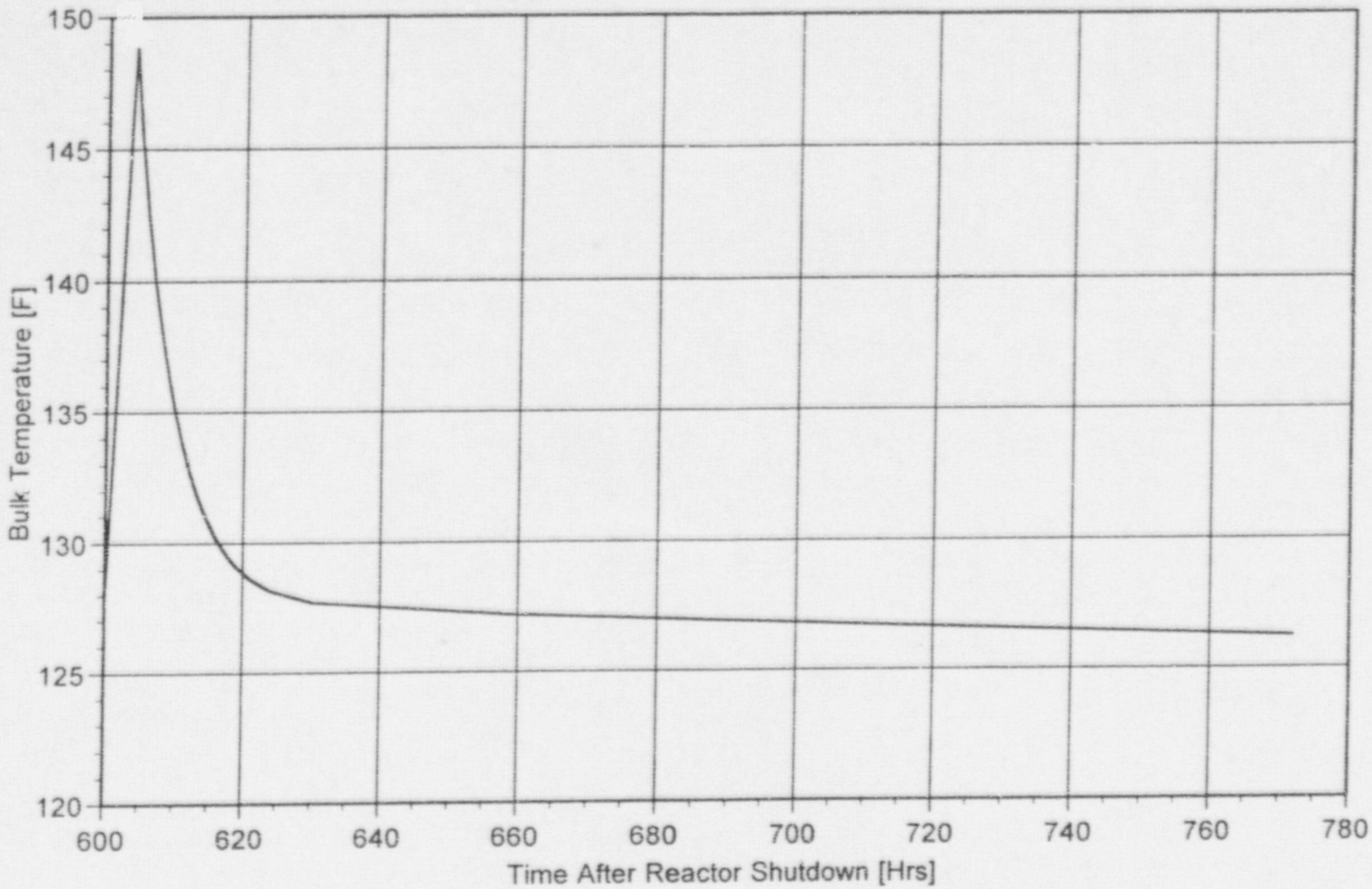
FIGURE 9.1-8
EMERGENCY CORE OFFLOAD

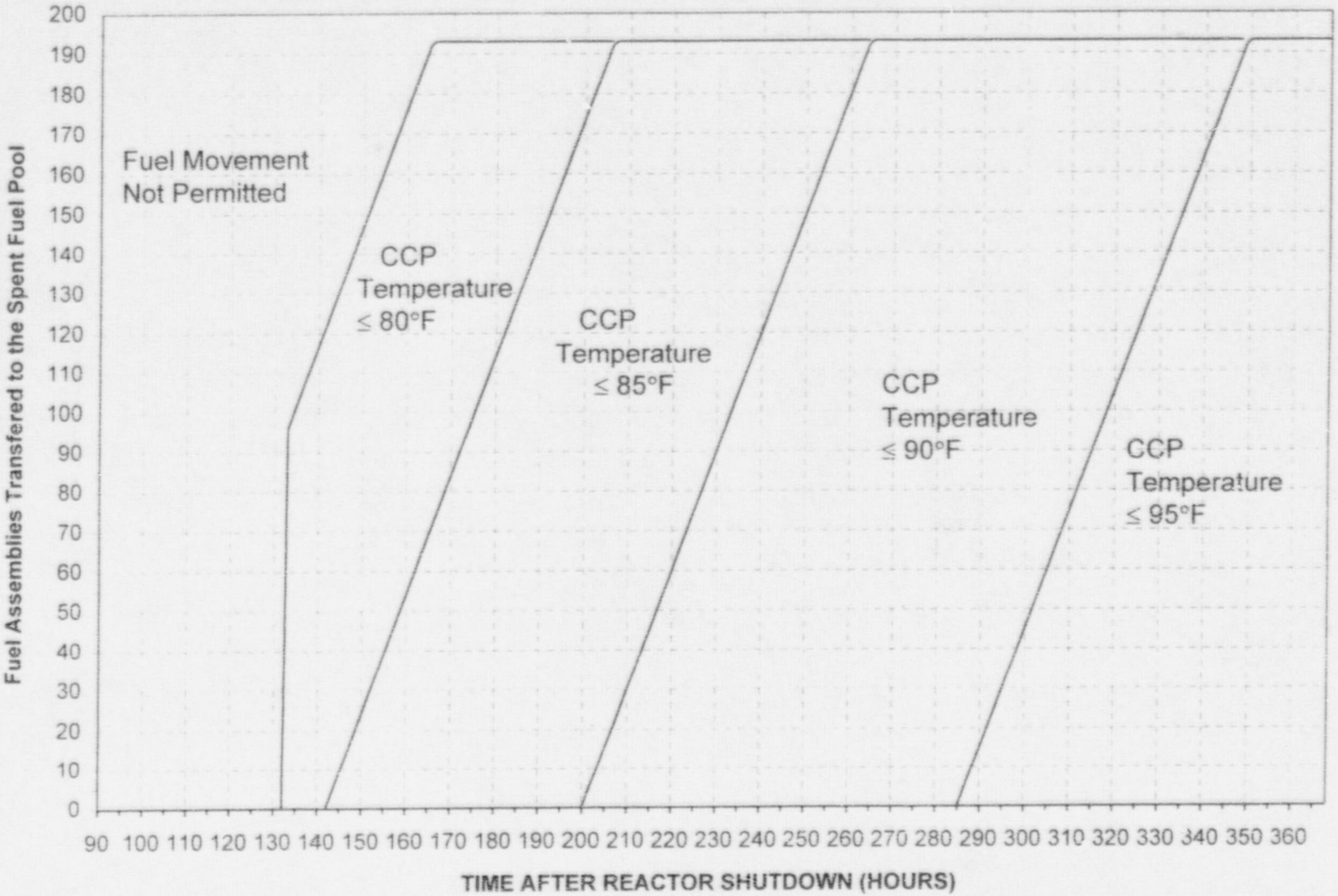
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT




COOLDOWN CURVE FOR NORMAL OPERATION (4-HOUR LOSS OF POOL COOLING)

Figure 9.1-8

## Figure 9.1-20
## Fuel Assembly Transfer Limit Verses CCP Temperature




**MILLSTONE UNIT 3 FULLCORE OFFLOAD**

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TABLE 9.1-2

PERFORMANCE CHARACTERISTICS OF
THE FUEL POOL COOLING SYSTEM
(ONE FUEL POOL COOLER OPERATING)

| Operating Condition | Normal Refueling (Fuel Shuffle) Cond. 1 | Normal Plant Operation Cond. 2 | End-of-Cycle Full-Core Off-load (6 events w/cond 4) Cond. 3 | Emergency Full Core Off-load (Abnormal Maximum 6 events w/cond 3) Cond. 4 |
|---|---|---|---|---|
| Heat Load BTU/hr | $2.177 \times 10^7$ | $1.859 \times 10^7$ | $3.479 \times 10^7$ | $3.505 \times 10^7$ |
| Required Duty of One Fuel Pool Cooler BTU/hr | $2.177 \times 10^7$ | $1.859 \times 10^7$ | $3.479 \times 10^7$ | $3.505 \times 10^7$ |
| Maximum Temperature Long Term | 130°F | 125°F | 150°F | 150°F |
| Maximum Peak Temperature Short Term | 140°F | 140°F | 155°F | 150°F (No Failures Assumed) |
| Maximum Peak Temperature (Accidents) | 140°F | 145°F | 155°F | 150°F (No Failures Assumed) |
| Design Limits | Maximum Long Term Temperature (Structural Requirement): 150°F<br>Maximum Short Term Temperature (Structural Requirement): 200°F<br>Maximum Temperature Loss of Pool Cooling (Cond. 1-3): 155°F | | | |
| Flow Rates | Tube Side - Fuel Pool Water (SFC): 3,500 gpm<br>Shell Side - Reactor Plant Component Cooling Water (CCP): 1,800 gpm | | | |
| Cooling Water Temperatures | Reactor Plant Component Cooling (CCP): 95°F<br>Service Water Temperature (Cools CCP): 75°F | | | |

97-2
4/97

Note: Conditions (Cond.) refer to descriptions in Section 9.1.3.3.

- vibration monitor (common).

Power not available status lights are provided on the rear of the main control board for each motor control center.

All radiation monitor alarms annunciate in the control room.

### 9.4.2 Fuel Building Ventilation System

The fuel building ventilation system (Figure 9.4-2) removes heat generated by equipment and water vapor from fuel pool evaporation, prevents moisture condensation on interior walls, provides a suitable environment for equipment operation and personnel. It also limits potential radioactive release to the atmosphere during normal operation or anticipated operational transients, and following a postulated fuel handling accident (FHA).

#### 9.4.2.1 Design Bases

The fuel building ventilation system is designed in accordance with the following criteria.

1. General Design Criterion 2, as related to the system being capable of withstanding the effects of earthquakes.

2. General Design Criterion 5, as related to shared systems and components important to safety.

3. General Design Criterion 60 and Regulatory Guides 1.52 and 1.140, for design testing and maintenance criteria for atmosphere cleanup systems.

4. General Design Criterion 61 and Regulatory Guide 1.13, for fuel storage and radioactive control.

5. Outdoor air design temperatures are listed under design weather data in Section 9.4. The fuel building ventilation system is designed to maintain the following space temperatures during normal operation.

94-37 (2/95)

Insert F

| Spent Fuel Pool Area | Maximum Space Temperature | Minimum Space Temperature |
|---|---|---|
| a. Pool water temperature is greater than 140°F | 98°F | 85°F |
| b ~~a~~. Pool water temperature is greater than 100°F | 95°F | 85°F |
| c ~~b~~. Pool water temperature is less than 100°F | 95°F | 65° |
| d ~~c~~. All other areas | 104°F | 65°F |

6. Air flow is directed from areas of lower potential radioactivity to areas of higher potential radioactivity.

Docket No. 50-423
B17004

Attachment 6

Millstone Nuclear Power Station Unit No. 3
License Amendment Request and Technical Specification
Changes For Full Core Off-load
**Retyped Technical Specification Page(s)**

January 1999

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

| AREA | TEMPERATURE LIMIT (°F) |
|---|---|
| 7. FUEL BUILDING | |
| FB-02, Fuel Pool Pump Cubicles, El 24'6" | ≤ 119 |
| FB-03, General Area, El 52'4" | ≤ 108 |
| 8. FUEL OIL VAULT | |
| FV-01, Diesel Fuel Oil Vault | ≤ 95 |
| 9. HYDROGEN RECOMBINER BUILDING | |
| HR-01, Recombiner Skid Area, El 24'6" | ≤ 125 |
| HR-02, Controls Area, El 24'6" | ≤ 110 |
| HR-03, Sampling Area, El 24'6" | ≤ 110 |
| HR-04, HVAC Area, El 37'6" | ≤ 110 |
| 10. MAIN STEAM VALVE BUILDING | |
| MS-01, Areas above El. 58'0" | ≤ 140 |
| MS-02, Areas below El. 58'0" | ≤ 140 |
| 11. TURBINE BUILDING | |
| TB-01, Entire Building | ≤ 115 |
| 12. TUNNEL | |
| TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building | ≤ 112 |
| 13. YARD | |
| YD-01, Yard | ≤ 115 |