UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 2, 1999

LICENSEE: Northeast Nuclear Energy Company

FACILITY: Millstone Nuclear Power Station, Unit 1

SUBJECT: SUMMARY OF NOVEMBER 15, 1999, MEETING TO DISCUSS THE LICENSEE'S EVALUATION OF THE POTENTIAL FOR A ZIRCONIUM FIRE FOLLOWING A BEYOND DESIGN BASIS COMPLETE LOSS OF WATER IN THE SPENT FUEL POOL (TAC MA6658)

A meeting was held at One White Flint North in Rockville, Maryland at 1:30 p.m. on Monday, November 15, 1999, between the Nuclear Regulatory Commission staff and the licensee for Millstone Nuclear Power Station, Unit 1 (MP1). The meeting was held at the request of the licensee to address the staff's interest in the licensee's plant specific evaluation of the potential for a zirconium fire following a beyond design basis complete loss of water in the spent fuel pool. The evaluation is referenced in a September 28, 1999, licensee request for an exemption from the secondary financial indemnity requirements of 10 CFR 140.11(a)(4) which is currently being reviewed by the staff. Enclosure 1 is the list of meeting attendees, and Enclosure 2 is a handout of the slides used by the licensee in their presentation. Enclosure 3 is an Executive Summary of the licensee's evaluation, which had been provided to the staff on November 10, 1999, via telefax, in preparation for the meeting, and was also given to the staff at the conclusion of the meeting.

The licensee made a presentation of the information in the enclosed slides. After the presentation, the staff asked specific questions regarding various technical assumptions and parameters used for the evaluation.

The licensee stated that it should not be necessary for the staff to have the complete evaluation. It was the licensee's view that sufficient data already exist for an MP1 safety evaluation in the generic studies performed to date on the runaway zirconium oxidation phenomenon, in addition to information in the Executive Summary for the MP1 evaluation, the attached slides, and answers to staff questions at the meeting, which the licensee offered to provide in formal, docketed correspondence. The licensee also pointed out there is no approved methodology or acceptance criteria for granting an exemption involving an assessment of the zirconium fire potential in a spent fuel pool.

Following the questions and answers, the staff stated that, based on the licensee's exemption justification, which included the assertion that the potential loss of spent fuel pool water inventory is no longer a major concern with regard to offsite dose, it would be necessary for the licensee to submit the analysis in order for the staff to prepare a safety evaluation of the exemption request.

The staff noted that the data and conclusions developed to date in generic studies do not provide all the technical information necessary to form an acceptable basis for a regulatory decision for a specific plant.

The staff informed the licensee that it would provide a response to their position that the staff should be able to review the exemption request without having a complete evaluation available for review.

ORIG. SIGNED BY
Louis L. Wheeler, Senior Project Manager
Decommissioning Section
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-245

Enclosures: As stated

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The staff noted that the data and conclusions developed to date in generic studies do not provide all the technical information necessary to form an acceptable basis for a regulatory decision for a specific plant.

The staff informed the licensee that it would provide a response to their position that the staff should be able to review the exemption request without having a complete evaluation available for review.

Louis L. Wheeler, Senior Project Manager
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LIST OF ATTENDEES

Millstone, Unit 1 Public Meeting

November 15, 1999

NRC Staff

Louis Wheeler	NRR, Project Directorate IV & Decommissioning
Diane Jackson	NRR, Plant Systems Branch
Joseph Staudenmeier	NRR, Reactor Systems Branch
Daniel Barss	NRR, Operator Licensing, Human Performance and Plant Support Branch
James Shepherd	NMSS, Decommissioning Projects Branch
Ronald Bellamy*	Region I, Decommissioning and Laboratory Branch

Northeast Nuclear Energy Company and Contractors

Bryan Ford
Vincenzo Veglia
Michael Annon
Stuart Thickman
Jack Dallman

Members of the Public

Jenny Weil	McGraw-Hill, Inc.
Steve Love	BNFL, Inc.

* Attended the last 5 - 10 minutes

ENCLOSURE 1

Millstone Unit 1 Insurance Exemption

Bryan Ford
Director, NSRA
November 15, 1999

Meeting Purposes

- Summarize Unit 1 Secondary Layer of Insurance Coverage Exemption Request and basis
- Answer Staff questions about the basis for Millstone's conclusion concerning Zirconium fires in the Spent Fuel Pool

ENCLOSURE 2

Background

- Millstone Unit No. 1 Shutdown November 4, 1995 (4 years)
- Secondary Insurance Coverage Exemption Request submitted September 28, 1999
- Scope of Exempt Request very limited

Basis for Exemption Request

- Analyses/Tests previously performed
  - Ultimate Heat Load Analysis
  - Spent Fuel Pool Heat Up Test
  - Fuel Handling Accident
- Available industry information indicates Zirconium fires are not credible
  - Long decay time (4 years)
  - Low burnup
- Plant specific evaluation of potential for Zirconium fire

Zirc Fire Analysis

Jim Veglia
Manager, Engineering

Generic Considerations

- Temperatures
  - 1600°C - Ingition point of Zirc-4 tubing based on testing (*Cooper*)
  - 1400°C - Ingnition point of zirconium (*NFPA Fire Protection Handbook*)
  - 1300°C - Ignition point of bulk zirconium based on testing (*Mellor*)
  - 950°C - Onset of zirconium oxidation (*Sandia labs 1)*
  - 850°C - Onset of zirconium oxidation (*Sandia labs 2*)
  - 800°C - Onset of self-sustaining zirconium oxidation (*GSI-82*)
  - 565°C - Onset of clad swelling (*SECY-97-186, Draft Technical Study of SFP Accidents for Decom Plants*)

Generic Considerations

- **Existing Spent Fuel Heatup Analysis Critical Parameters**
  - **Runaway Oxidation Comparison:**
    - **SFUEL (GSI-82 and draft Tech. Eval.)**
      >180 days <14 kW/MTU decay power per assy.
    - **MP1**
      >4 years 3.4 kW/MTU (worst case assemblies)
  - **Clad Swelling Comparison (565°C):**
    - **SHARP (GSI-82 and draft Tech. Eval.)**
      >7 months <40 GWD/MTU burnup
    - **MP1**
      >4 years 34.3 GWD/MTU (worst case bundles)

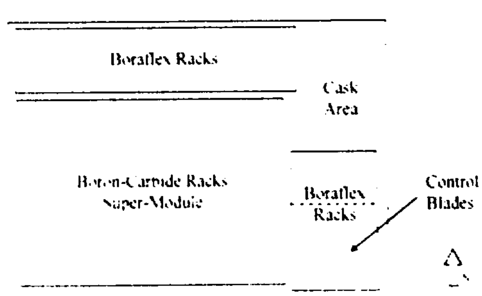
Analysis Overview

- **Objectives**
- **Spent Fuel Pool Layout**
- **Scenario**
- **Assumptions**
- **Methodology**
- **Results**
- **Conclusions**

Objectives

- Demonstrate that on a complete loss of water inventory in the MP1 spent fuel pool:
  - Fuel clad will not swell
  - Fuel clad will not oxidize
  - A fuel clad zirconium fire will not occur
- Provide Reactor Building HVAC flow condition sensitivity analysis
- Input to defining Safe Shutdown Basis for MP1 (part of larger MP1 effort)

Spent Fuel Pool Layout



Scenario

- Instantaneous loss of all water from the Spent Fuel Pool
- No compensatory actions taken to replace water
- Various reactor building air flow configurations:
  - Zirc Fire Analysis reference flow conditions:
    - 100%
    - Loss of HVAC (RB tornado panels & RR door open)
  - Sensitivity analysis of flow conditions
    - 50%
    - 20%

Assumptions

- Event assumed to occur 4 years after shutdown.
- RB volumes & structures below refueling floor & outside SFP ignored.
- Thermal capacities of fuel racks, water rods, miscellaneous RB equipment neglected.
- Flow through fuel racks modeled for maximum resistance using 8x8 fuel bundle geometry.
- Fuel heat structure geometry modeled as if all bundles were 8x8 design with a single water rod.

Assumptions

- In failed HVAC case, all seven tornado panels were assumed to be opened.
- Flow length under fuel racks modeled as length from periphery of pool to pool center.
- RB initialized to worst-case summer conditions.
  - Daytime environmental conditions maintained throughout event.
  - Outside RB walls modeled with no-wind condition
- 20% of RB total flow assumed to cool spent fuel pool (i.e., 80% of flow ignored)
- Adiabatic boundary conditions assumed on worst case bundle (i.e., only convective heat transfer).

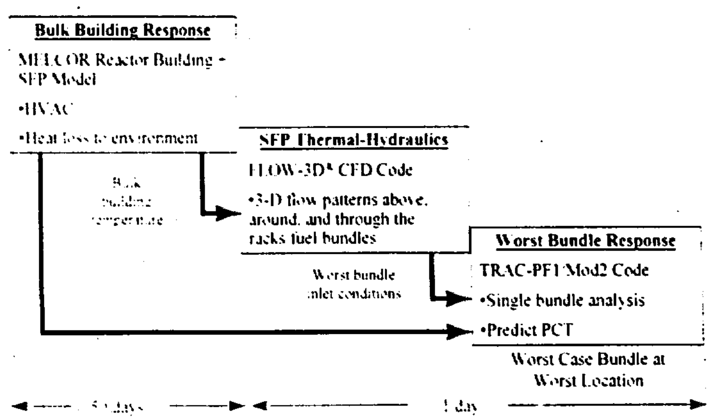
Methodology

- Apply appropriate mainstream reactor safety/engineering codes:
  - MELCOR
  - FLOW-3D©
  - TRAC-PF1/Mod 2
  - ORIGEN2

Methodology

- **Analysis Complications**
  - **Relative scale of features being modeled meant a single modeling code was not adequate:**
    - Model 1 - Bulk reactor building response
    - Model 2 - SFP fluid dynamic response
    - Model 3 - Bundle and rod clad response
  - **Need to consider physical interactions under various RB HVAC flow conditions.**
  - **Long duration event & multiple time scale event:**
    - 10 days to reach 90% of RB temperature rise & 50 days for RB temperature to stabilize.
    - Clad takes less than a day to stabilize following perturbation.

Methodology



Results

HVAC Flow	ORIGEN2 PCT	Delta to 565°C
100 %	310°C	255°C
Failed	310°C	255°C
50%	331°C	234°C
20%	362°C	203°C

Analysis Conclusion

Millstone 1 Spent Fuel <u>IS NOT</u> Susceptible to Clad Swelling or Zirconium Fire on Complete Loss of Spent Fuel Pool Water Inventory

Summary

- Request should be acceptable
  - Limited Exemption for Insurance Coverage
  - Consistent with previous NRC determinations
  - Design Basis Accidents have been shown to be acceptable
  - Generic industry information indicates that of a Zirconium Fire is not credible
  - Plant specific analysis indicates that of a Zirconium Fire is not credible
  - Analysis shows that a Zirconium Fire would be a long lead time transient allowing significant time (days to months) to take mitigating action

EXECUTIVE SUMMARY

An analysis of a loss-of-coolant inventory scenario was performed for the Millstone 1 spent fuel pool (SFP). The analysis used a three-level approach to accurately capture the scale of the phenomena present in the scenario. First, MELCOR calculations were run to assess the bulk building response in the room above the SFP. As shown in Table ES-1, the MELCOR calculations showed the peak room temperature was dependent upon the magnitude of the HVAC flow. A calculation was also performed assuming the HVAC system was unavailable but the operator successfully opened the railroad door and the tornado panels above the SFP. These two openings allowed a natural circulation flow pattern to remove heat from the building approximately as effectively as the nominal 100% HVAC configuration.

Table ES-1. Summary of Peak Building Temperatures as a Function of HVAC Configuration.

Calculation	Date of Scenario	HVAC Flow	Peak Building Temperature [°F]
1	Sept 1. 1999	20% Capacity	217
2	Sept 1. 1999	50% Capacity	183
3	Sept 1. 1999	100% Capacity	160
4	Sept 1. 1999	Failed[a]	160

Note (a) Includes the operator actions to open the railroad door and the tornado panels.

Next, 3-dimensional flow calculations in the SFP were performed using FLOW-3D©. The FLOW-3D© model used 36,000 computational cells in the SFP to accurately resolve the flow patterns through, around, and above the fuel racks. From the FLOW-3D© calculations, the worst bundle location was identified as next to the large open region in the northeast corner of the SFP (see Figure ES-1). The northeast corner of the SFP had the largest contiguous open area for downflow and also had the highest circulation flow under the racks. The bundles adjacent to the open space had the lowest flow due to a "Bernoulli" effect from the relatively high-speed flow under the rack. At this location, there was also an 20°F increase in the bulk air temperature at the worst bundle inlet from the air entering the SFP due to thermal mixing of the plume exiting the fuel bundles with cooler air from above the SFP.

Finally, TRAC-PF1 Mod2 single bundle calculations were performed using the highest power bundle at the worst bundle location from the FLOW-3D© simulation. The single bundle calculations were varied across a range of HVAC conditions generated by MELCOR. The TRAC-PF1 Mod2 results show that the best-estimate peak cladding temperature was only 684°F (362°C), based on decay heat values from ORIGEN2 calculations for September 1, 1999 (see Table ES-2). In addition, the very conservative NRC ASB 9.2 decay heat level was input into the TRAC-PF1 Mod2 model. The peak cladding temperature for this calculation stabilized at 887°F (475°C). Even with the most conservative decay heat value, the peak cladding temperature was 90°C below the 565°C value conservatively used by the NRC to identify the onset of fuel cladding ballooning [11].

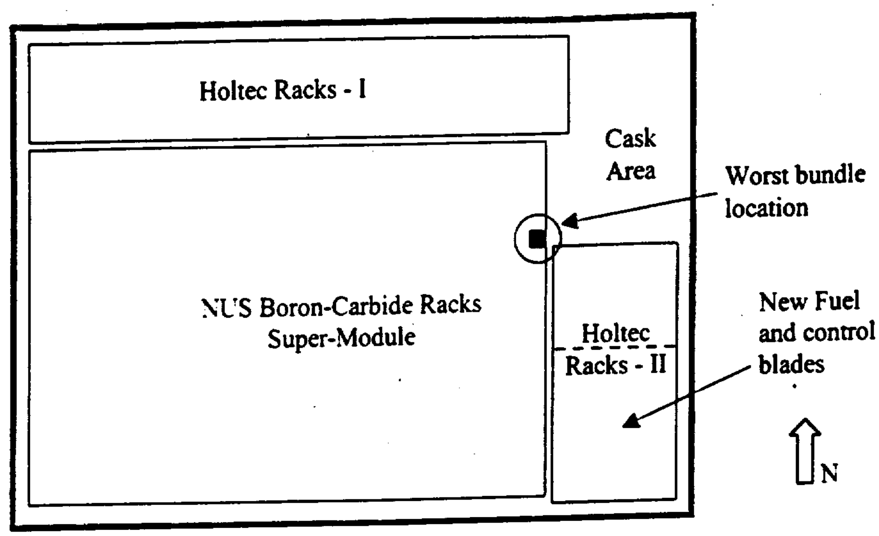


Figure ES-1. Worst Bundle Location.

Table ES-2. Summary of the Peak Cladding Results.

HVAC Flow	Worst Bundle Inlet Temperature	Peak Cladding Temperature [°F] ORIGEN2 Best-Estimate Decay Heat	NRC ASB 9.2 Decay Heat
20% Capacity	237°F	684°F (362°C)	887°F (475°C)
50% Capacity	203°F	628°F (331°C)	822°F (439°C)
100% Capacity Failed[a]	180°F	590°F (310°C)	779°F (415°C)

Note (a): Includes the operator actions to open the railroad door and the tornado panels.