Docket File

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

May 8, 1992

Docket No. 50-336

LICENSEE: Northeast Utilities

FACILITY: Millstone, Unit 2

SUBJECT: SUMMARY OF MEETING OF APRIL 22, 1992, WITH REPRESENTATIVES OF NORTHEAST UTILITIES CONCERNING THE PROGRAM FOR THE REPLACEMENT OF THE STEAM GENERATORS FOR MILLSTONE 2 IN 1992

INTRODUCTION

On April 22, 1992, representatives of the NRC and Northeast Utilities met in the NRC offices in Rockville, Maryland, to discuss the Millstone 2 steam generator replacement project. The purpose of the meeting was to specifically address the integrated safety evaluation that will determine that the modification can be implemented within the criteria of 10 CFR 50.59. The attendance list is provided in Enclosure 1. Enclosure 2 provides the agenda and copies of the viewgraphs supporting the licensee's presentation.

DISCUSSION

This was the sixth meeting with the staff on the steam generator replacement project. The focus of this meeting was the integrated safety evaluation that will support the licensee's determination that the modifications can be made under the criteria of 10 CFR 50.59. Enclosure 2 provides a good discussion of this scope.

Individual safety evaluations have been prepared by different disciplines and the integrated safety evaluation will bring all these together to form an integrated safety evaluation which will make the determination that the replacement can be accomplished under the requirements of 10 CFR 50.59. The licensee has tentative plans to provide the PCDR to the PORC members during the week of April 26 and present it to PORC for approval in mid May 1992. The shutdown for replacement is scheduled for May 30, 1992.

All FSAR accident analyses, nozzle dam failure, fuel assembly/vessel internals uplift, heavy loads drop, and the seismic analysis have been considered with many being redone. The operational activities have been studied with the simulator. The licensee believes operational response will be improved with the wide range level instrumentation. They do not believe the slight difference in inventory will be significant. However, as requested by the staff, the licensee will take another look at the plant response and provide its testing program prior to start up. Since the travel route of the old and the new steam generators in and out of the plant will not be over the spent fuel pool or its equipment, heavy loads over the spent fuel pool will not be a concern. The licensee has determined that there will be no change in EOPs or

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technical specifications. Also, the licensee does not anticipate any request for relief from the ASME Code requirements. The most sensitive weld is the girth weld joining the upper drum sections of the steam generators to the new lower tube sections. This large weld could be subject to cracking. The annealing of the weld and heat control during the welding will be controlled more conservatively than Code requirements.

/s/

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cc: See next page

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ENCLOSURE 1

MEETING ATTENDANCE
4/22/92

Name	Organization
Steve Routh	SERCH - Bechtel
Peter Wielhouwer	ABB Comb. Engineering
John Resetar	NU
Tim Honan	NU
Forrest Kocon	NU
John Rhodes	NU
Raymond Necci	NU
Prem Godha	NU
R. Hal Young	NU Licensing
A. J. Silvia	NU
J. E. Carrasco	USNRC, Reg. I
Guy S. Vissing	USNRC/NRR/PD I-4
Kenneth S. Kolnczyk	USNRC, Resident Inspector, MP-1
Kulin D. Desai	SRXB/NRR
James Tatum	NRR/SPLB
Horace Shaw	NRR/DET/EMEB
Jim Wiggins	NRR/EMCB
Janak Raval	NRR/SPLB
G. Hornseth	NRR/EMCB

Millstone 2 Steam Generator Replacement Project

NRC Safety Assessment Presentation

MP2 S/G REPLACEMENT PROJECT

NRC - APRIL 22, 1992

ENCLOSURE 2

Agenda

• Introduction & Review	R. Necci
• Integrated Safety Evaluation (System Interface Reconciliation)	T. Honan
• PDCR / 10CFR50.59 Conclusion	J. Resetar
• Project Conclusion	R. Necci

Introduction

Purpose

- "To Present the Results of the Integrated Safety Evaluation"

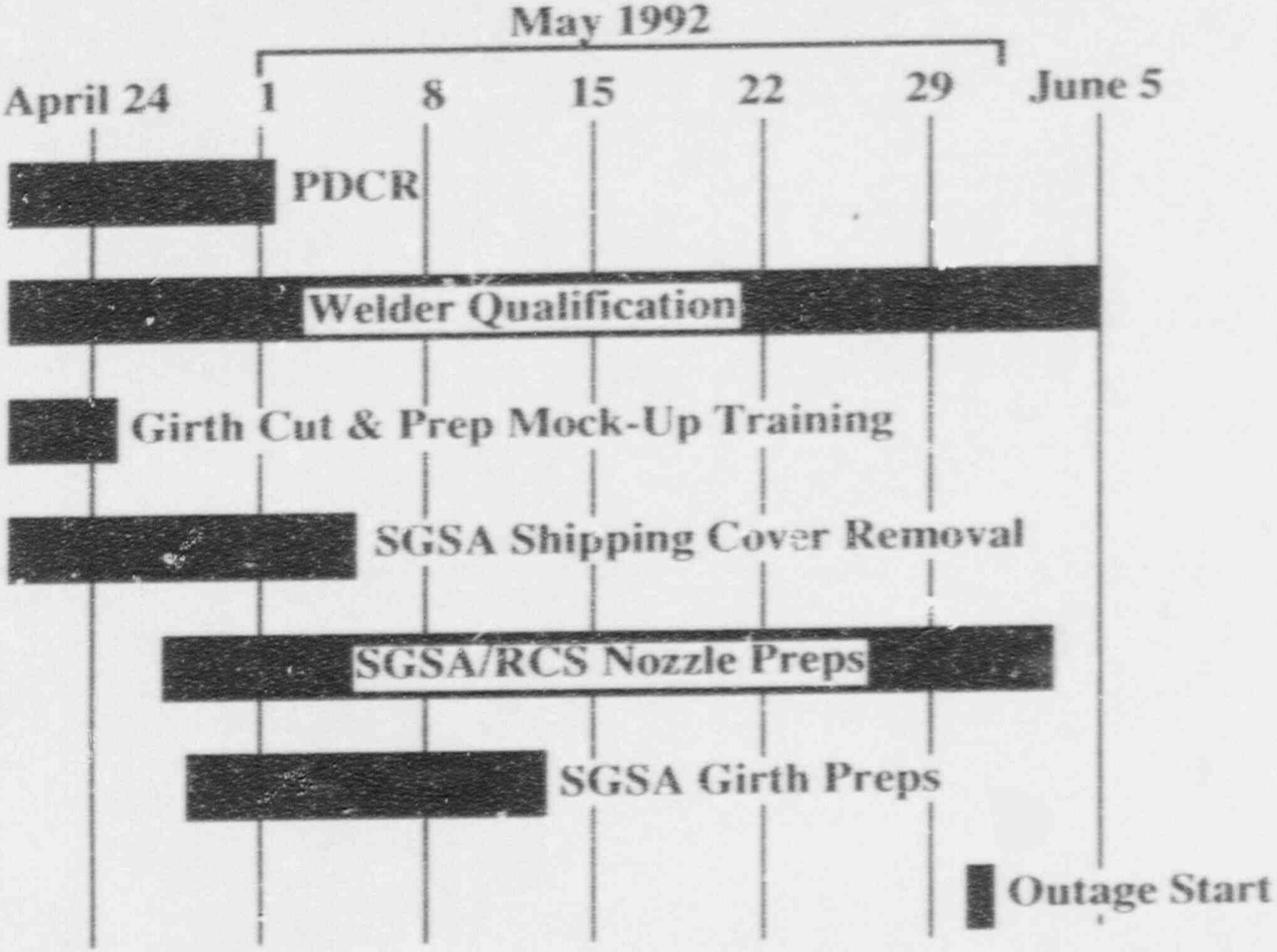
Conclusion

- "The Replacement Can be Accomplished via 10CFR50.59"

Project Status

- Staffing
  - Fluor
    - Approximately 200 Craft
    - Approximately 106 Staff
  - NU
    - Approximately 40 Staff
- Training
  - Basic Site Training
  - Welder Qualification
  - Mock-up Training
- Outage Status
  - May 30, 1992 Start Date

MP2 SGRP
Pre-Installation Activities



Open Communications With NRC

1. NRC / NU Counterparts Meeting
   - Project Overview
2. NRC / SGRP
   - Project History
   - New S/Gs' Design & Maintenance
3. NRC / SGRP
   - Rigging
   - S/G Fit-up Measurements
   - Cutting / Welding
   - ASME Code

Open Communications With NRC
(continued)

4. NRC / SGRP
   - Radiation Protection
   - Quality Programs
   - NRC Procurement Audit

5. NRC / SGRP
   - Safety Evaluation (New S/G's)
   - Safety Evaluation (Installation)

6. Other Meetings
   - Audit of B&W-C
   - S/G Disposal

SGRP PDCR's

- Support Facilities
- Polar Crane Upgrade
- Structural Modification
- Wide Range Monitoring
- Haul Route Upgrade
- S/G Replacement

S/G Replacement PDCR Format

- Component Design Reconciliation
- Installation
- System Interface Reconciliation
- Integrated Safety Evaluation

Safety Evaluation

- Input - NU, ABB/CE, Siemens (ANF), Bechtel, F/D, B&W
- Safety Evaluation - NU Engineering
- Review - NNECO PORC
- Summary
  - Final Assessment - MP2 SG Replacement Meets 10CFR50.59
  - Final Assessment - No Unreviewed Safety Questions

Integrated Safety Evaluation

(System Interface Reconciliation)

MP2 S/G REPLACEMENT PROJECT
NRC - APRIL 22, 1992

Overview

- Integration of Wide Range of Safety Evaluations
  - NU
  - ABB/CE
  - Siemens (ANF)
  - B&W
  - Fluor Daniel
- Conclusion
  - Safe
  - Not an Unreviewed Safety Question

Accidents Considered

- All FSAR Chapter 14 Events
- Nozzle Dam Failure
- Fuel Assembly/Vessel Internals Uplift
- Heavy Loads Drop
- Seismic Analysis

Basic Considerations

- Replacement S/G's Similar to Original
  - Heat Transfer
  - Liquid Mass
  - Structural Characteristics
- Plant Will Return to Original Unplugged Condition

Effect of Change in Plugging Level

- Current Analysis Assumptions
  - Max Plugging Where Low Heat Transfer is Conservative
  - Zero Plugging Where High Heat Transfer is Conservative
- Cooldown Analyses Unaffected
- Heatup Analyses Less Severe

Analyses of Interest

- Steam Line Break (Core Response)
- Steam Generator Tube Rupture
- Loss of Coolant Accident (Core Response)
- DNB Analysis
- Heatup Events
- Containment Analysis
- Nozzle Dam Failure
- Fuel Assembly / Internals Uplift
- Heavy Loads Drop
- Seismic Analysis

Steam Line Break (Core Response)

- Current Analysis Assumes Zero Plugging
- Similar Liquid Masses
- Thermal-Hydraulics Similar
- Integral Flow Restrictors
  - Substantial Analytical Credit
- Decrease in Severity

Steam Generator Tube Rupture

- Increased Reliability
- Greater Tube I.D.
  - Flow Area 1.2% Higher
  - Increased Leak Rate
  - Maximum Dose Increase
    - 1% Whole Body
    - 3.4% Thyroid
- Increase in Doses Must be Graeter Than 10% Value to be a USQ

Loss of Coolant Accident (Core Response)

- Large Break
  - Max Plugging is Conservative
    - Delays Blowdown
  - Reduced Plugging Levels Decrease Severity
- Small Break
  - S/G is Heat Sink Early in Transient
  - Second Order Effect
  - No Significant Change in Severity

DNB Analysis

- Limited by Loss of Flow
  - No Change in Core Power Distribution
  - Reduced Plugging Decreases Flow Coastdown
  - Slight Improvement in Minimum DNBR

Heatup Events

- Loss of Load
  - Reduced Plugging Improves Heat Transfer
  - Increased Secondary Pressure Minimizes Heatup
  - Slight Decrease in Peak Pressure
- Loss of Feedwater
  - Slight Decrease in Secondary Side Inventory
  - Accommodated with Available Minimum Inventory

Containment Analysis

- Considered Both LOCA and Steam Line Break
- Identical Cases Considered for Current & Replacement Generators
- Cases Are Those Considered for Current FSAR Analysis

Containment Analysis
(Continued)

- Steam Line Break
  - ○ Limiting Temperature Case
    - Hot Full Power
    - Less Limiting Due to Flow Restrictors
  - ○ Limiting Pressure Case
    - Hot Zero Power
    - Peak Pressure - 50.8 psi
    - Slightly (<0.3 psi) Higher
  - Peak Pressure Still Well Below Design Limit of 54 psi

Containment Analysis
(Continued)

- Loss of Coolant Accident
- Hot Leg Break Limiting
- Peak Pressure - 52.0 psi
  - Slightly (<0.6 psi) Higher
  - Peak Pressure Still Below Design Limit of 54 psi
- Peak Temperature - 281.0 °F
  - Slightly (<0.8 °F) Higher
  - Peak Temperature Still Below Design Limit of 289 °F

Nozzle Dam Failure

- Substantially Different Design
- Less Susceptible to Mispositioning
- Includes Passive Seal
- No Increase in Worst Case Leakage
- New Design is Acceptable

Fuel Assembly/Internals Uplift

- Fuel Assembly
  - Function of Top Nozzle Spring Force
  - Fuel Assembly Design Flow Exceeds Max Expected Flow
- Internals Uplift
  - Bounded by Original Component Analysis
  - Hold-down Ring to be Inspected

Heavy Loads Drop & Seismic Analysis

- Heavy Loads Drop
  - Core to be Removed
  - Core Damage Precluded
  - No Significant Safety Equipment Affected
- Seismic Analysis
  - Increased Mass & Raised Center of Gravity
  - Impact Evaluated on Reactor Coolant Loops
  - All Design Criteria Met

PDCR / 10CFR50.59 Conclusion

Plant Review

- Plant Design Change Record (PDCR)
- 10CFR50.59 Determination

Steam Generators

- Meet Current & Original Standards
- Address Health & Safety

Replacement

- Meets Design Code
- Addresses Construction Issues
- Highest Safety Standards

Design Change Criteria

- Safe
- Review Against 10CFR50.59

Plant Design Change Record (PDCR) PORC Review

- Detailed Design of New Sub-Assembly & Installation Activities
- All Safety Evaluations
- Integrated Safety Evaluation
- PORC Concurrence / Acceptance of Conclusions That S/G Replacement is Not an Unreviewed Safety Question

10CFR50.59 Determination

- Submitted to PORC for Preliminary Review
- Special PORC Held to Discuss PDCR (NRC Resident Inspector & Project Manager Present)
- Final PORC Meeting for Approval of Complete PDCR

Conclusion

Summary