Docket File

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

May 6, 1992

Docket No. 50-336

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: Guy S. Vissing, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

SUBJECT: REPORT OF VISIT TO MILLSTONE 2 OF APRIL 13 THROUGH APRIL 17, 1992 (TAC NO. M83065)

INTRODUCTION

This was a quarterly project manager visit to the plant for the purpose of maintaining project manager awareness of Millstone, Unit No. 2, status, performance of inspections, and making observations relating to the steam generator replacement project training program. A major activity was the review of the licensee's 10 CFR 50.59 determinations on plant modifications and changes made during 1991. Another major activity was the inspection of the steam generator replacement project training activities and equipment for supporting the removal of the old steam generators and the installation of the new steam generators. The control room was visited to review logs of plant incident reports (PIRs) and control room operations in general.

The most significant recent PIR was the leak in the bottom plate of the "B" service water strainer. The "B" service water pump was taken out of service until the strainer will be repaired or replaced. The service water and screen house was visited to verify the above condition.

The Millstone site has discontinued the regular morning station staff meetings. Instead the licensee has installed TV monitors at key locations throughout the plant site that provide continuous information on each plant's status, significant events, weather information, staffing at each unit and other information of interest to the staff.

The afternoon unit staff meetings were attended. One PORC meeting was attended. A Nuclear Review Board (NRB) meeting was attended. The NRB was held in Berlin with the plant staff tied in by telephone. The NRB meeting was a special meeting to consider a proposed Technical Specification (TS) change to the spent fuel pool storage arrangement. Because of an error in the original criticality analysis of Region I area, the licensee determined that another arrangement was necessary to accommodate the storage of fresh fuel for refueling operations. Region I is the area where the fresh fuel and the core off load would have been stored. Fuel could be stored in each cell location since each cell location contained Boraflex in each side panel. Since the licensee determined that the criticality analysis of Region I would not

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support the storage of fresh fuel of 4.5 weight percent U-235 with a K effective less than .95, the proposed TS change would divide Region I into two areas. One area would contain blocking devices such that fuel assemblies loading would be limited to a three out of four arrangement. This area would be specifically for fresh fuel. The other area would remain the same. This would be for the core off load. A probing discussion ensued and the proposed change was found acceptable.

Another meeting was attended in which a conference call was being held with Region I staff and the NRR to discuss the notification of an event that could potentially effect the grid instability. A brush fire in the Waterford area took the 345 KV line to the plant out of service. The dispatcher requested power level at Unit 1 be reduced from 673 MW to 520 MW and power level at Unit 3 be reduced from 900 MW to 880 MW as a contingence to maintain grid stability.

Persons contacted included:

H. Young, Licensing Engineer *
John Riley, Engineering Manager*
Ray Necci, Steam Generator Project Manager*
Mike Ciccone, Staff Engineer* **
R. Spurr, Training operations
Dave Pascal, Training operations
Joe B[illegible]gin, engineering supervisor**
R. Whi[illegible], Steam Generator Replacement project
S. Orefice, Steam Generator Replacement project
R. Fosdick, Training Operations

(* - attendance at entrance meeting, ** - attendance at exit meeting)

DISCUSSION

Steam Generator Replacement Project Inspection

A tour of the training facilities and the equipment storage yard for the steam generator replacement project was made. This included the area for qualification of welders, the mockup of the bottom area of the new steam generators and the area where the practice in setup and operation of the special cutting and milling tool that will be used for preparation of the old steam generators for removal and the new steam generators for installation. The storage yard contained the special crane that will be installed on top of the block house in the containment to support the removal and installation of steam generators and the pulling devices for assisting in removal and installation of the steam generators. Also tests were being conducted on the load capability of the road surface that will be used to transport the steam generators.

The most significant activity was the setup and operation of the special cutting and milling tool for separating the upper portion of the steam generators from the lower tube portions and the preparation of the new steam generators for welding to the upper portion. The tool consisted of a structural ring, larger in diameter than the upper portion of the steam generators, consisting of a 1 ft. x 3" flat plate with two vertical lateral support plates welded to to it. The structural ring would be supported to the steam generator at the area of the cuts by cantilever brackets welded to the steam generator. Two beds, at 180 degrees apart for holding the cutting tools, traveled on the ring on two "v" shaped horizontal tracks. one inside the ring and one outside the ring. The tool beds were driven by a sprocket chain device at a speed of approximately 2.5 revolutions per minute. The chain encircled the ring and was guided by a bar welded to the side of the structural ring. The bed would hold the cutting tool (a 1" square bar) in a fixture that could be arranged to cut at different angles. The depth of the cut was controlled by a screw powered by a pneumatic device. The sprocket was driven by a pneumatic motor.

The upper section of a steam generator from one of the canceled WNP projects was being used for practice and training. It took approximately 2 days to separate the upper portion of the steam generator from the lower portion in the transition area with a horizontal cut. The cut was through about 9" of steel. The tool will be used to prepare the new steam generators for welding to the upper drum of the old steam generators. Two cuts will be made. One to separate the temporary plate from the steam generator and another to make a beveled cut for the girth weld. The old steam generators will be separated from the upper drum portion by a horizontal cut. This tool was especially designed for the Millstone 2 steam generator replacement project. After its use for Millstone, it is planned for use in cutting up the Shoreham reactor vessel.

Review of the Licensee's 10 CFR 50.59 Determinations of Modifications Made in 1991

Introduction

The licensee's annual report for January 1, 1991 to December 31, 1991 was reviewed. The report identified 20 plant design changes, 10 plant design change evaluations, 30 procedure changes, 17 jumper lifted lead-bypass changes, one set point change and 13 tests. The report provided a summary of each change including a description of each change, a reason for the change, an a short safety evaluation that concluded in every case that the change did not constitute an unreviewed safety question per criteria of 10 CFR 50.59. A sample of 10 plant change requests (PDCRs) (included 3 PDCRs that were identified as plant design change evaluations (PDCEs) in the annual report) and 3 procedure changes were reviewed in depth to determine if acceptable determinations were performed:

| PDCR Number | Title |
|---|---|
| 2- ,80-86 | Millstone Unit 2 Fire Shutdown System |
| 2-006-89 | Annunciator, Radiation Monitoring and Stack Flow Instrument Modifications |
| 2-015-91 | Millstone Unit 2 Reactor Building Closed Cooling Water Heat Exchanger Cover Monorail |
| 2-018-91 | Auxiliary Steam Detection/Isolation System |
| 2-023-91 | Replacement of Solenoid Valve on 2-SW-3.1A (HV6399) RBCCW "A" Header Heat Exchanger Inlet Valve |
| 2-093-91 | Feedwater Block Valves Auto Closure |
| 2-101-91 | Delete The Fire Water Cross-Tie to Service Water for the "A" EDG |
| 2-88-086 | RBCCW Heat Exchanger Channel Head Coating |
| 2-88-108 | EDG Exhaust Piping Support Modifications |
| 2-90-051 | RPS High Power Pretrip Test Point |

| Procedure Number | Title |
|---|---|
| EOP 2530, Rev. 0 | Station Blackout |
| GC-SE-13, Rev. 0 | Installation of Raychem Type NCBK Breakout Kits |
| MP 2720x2, Rev. 0 | Raychem NPKV Low Voltage Kit removal, Selection and Installation (EQ) |

Other aspects of the 10 CFR 50.59 determination review included the review of procedures and training.

Procedures

The licensee has established formal procedural guidance and controls to evaluate each change, test or experiment for which 10 CFR 50.59 is applicable, and determine whether an unreviewed safety question (USQ) exists. The licensee bases the determination on an assessment of the impact of the proposed change, test or experiment according to the criteria of 10 CFR 50.59. Formal procedural guidance is contained in Nuclear Engineering Operations (NEO) Procedure 3.12, Rev. 6, Safety Evaluations. Rev. 6 went into effect in August 1991 and was a major revision to Rev. 5 which went into effect in August 1989. The preparer of the safety evaluation (SE) follows the SE format (Figure 7.2 of NEO 3.12). Both revisions follow the guidance recommended in NSAC-125 and enables the preparer of a SE to address the 10 CFR 50.59 criteria by asking the seven questions from NSAC-125, Section 3.1 and to determine if an URQ exists. Most of the evaluations done in 1991 followed the guidance of NEO-3.12, Rev. 5. If the NEO 3.12 guide or format is not used, the preparer's manager is required to sign the form indicating concurrence.

NEO 8.06, Safety Evaluations for Station Procedures, provides procedural guidance on the requirements governing USQ determinations and SE's of proposed revisions or changes to nuclear safety procedures. NEO 8.06 references NEO 3.12 for the performance of SEs.

NEO 3.03, Plant Design Change Records (PDCRs), provides procedural guidance on proposed modifications and provides check off forms and criteria for determining if a SE is required and if an integrated SE is required.

ACP-QA-3.02, Station Procedures and Forms, provides procedural guidance on proposed procedure modifications and provides a cover sheet with check off area and criteria for determining if a SE is required. An integrated SE is determined by the Manager of Safety Analysis if a SE is determined to be required.

The licensee has a well structured set of procedures in place governing the determination of USQ per the criteria of 10 CFR 50.59. NEO 3.12, Rev. 6 is an excellent procedure which, if followed, would assure proper determinations.

Training

The licensee's training department provides training on preparation of SEs per NEO 3.12, Rev. 6, which is a section of the training provided on NEO 3.03, PDCRs. Both instructors for SEs and PDCRs appear to have a good understanding of the subject matter. Both a lecture section and a workshop session is provided in the course work. The course for NEO 3.12, Rev. 6, has recently been implemented. It was noted that of 32 qualified to do SEs only 6 have received the training. The remaining 26 have been validated and excused from the training by "grandfathering" having been employed at Millstone since before 1983 or having experience in preparing SEs. The instructor was preparing the continuing instruction program and he was encouraged to have the managers commit to have all qualified SE preparers take the unit of instruction. It was concluded that the training course was adequate but more emphasis could be placed on having all qualified individuals participate in the training.

Reviews of 10 CFR 50.59 Determinations

All SEs that were reviewed complied with the requirements 10 CFR 50.59. Following is a discussion of significant findings:

Two PDCRs (2-080-86 and 2-093-91) provided integrated SEs. These were high quality SEs that followed the guidelines and format of NEO 3.12 and thoroughly addressed the seven questions to satisfy the 10 CFR 50.59 criteria for determining an USQ. It was noted that there is no distinguishing notation on the title or the body of an integrated SE that identifies it as an integrated SE. Other integrated SEs may have been missed for that reason.

Three other PDCRs (2-006-89, 2-018-91 and 2-101-91) provided excellent SEs that followed the guidelines and format of NEO 3.12 and thoroughly addressed the seven questions to satisfy the 10 CFR 50.59 criteria for determining an USQ.

Three PDCRs (2-023-91, 2-88-086 and 2-90-051) followed the format of NEO 3.12 and the seven questions were addressed; however, thorough responses to the seven questions were lacking. The brevity of these SEs was due to the simplicity of the modifications. One modification was a one for one replacement of a component. Also, two of the SEs were prepared prior to 1991, before the implementation of NEO 3.12, Rev. 6.

Two PDCRs (2-015-91 and 2-88-108) did not follow the NEO 3.12 format but provide a general discussion that concluded that there was no USQ. They briefly addressed the three criteria of 10 CFR 50.59 with little explanation. These changes were structural modifications and the SEs primarily addressed the addequacy of the supports.

Two SEs for procedures (GC-SE-13 and MP 2720x2) were short but addresses adequately the seven criteria for determining that there was no USQ. The SE for the Station Blackout procedure followed the format of NEO 3.12 but fell short in responding to the seven questions for determining if there was an USQ. This was an integrated SE and it did not measure up to the quality of other integrated SEs.

Conclusion

The licensee complies with the evaluation process of 10 CFR 50.59 for changes. The licensee has excellent procedures for implementing changes and for determining if there is an USQ. The licensee has a good training program available to those that need it. Supervisors of those qualified preparers of SEs should not continue the practice of excusing those preparers from initial training and continuing training on the basis of their experience. Corrective actions to continue to improve SEs will be reviewed as part of future 10 CFR 50.59 reviews.

/s/

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cc: Douglas Dempsey, RI
Gene Kelly, RI

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