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BYR 94-011

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attention: Mr. Morton Fairtile
Senior Project Manager
Nonpower Reactors and Decommissioning Project Directorate
Division of Operating Reactor Support

Reference: License No. DPR-3 (Docket No. 50-29)

Subject: 1993 ANNUAL REPORT

Dear Mr. Fairtile:

Enclosed is the Yankee Atomic Electric Company Annual Report for 1993. This report is submitted in accordance with 10CFR50.59(b)(2) and the requirements of Yankee Nuclear Power Station (YNPS) Defueled Technical Specification 6.8.1.

The Yankee Nuclear Power Station has been shut down since October 1, 1991, and the reactor defueled since February 15, 1992. The plant has been operating under a Possession Only License (POL) since August 5, 1992. The Decommissioning Plan for the facility was submitted to the Nuclear Regulatory Commission on December 20, 1993.

The attached report briefly describes the facility changes, tests, and experiments implemented without NRC prior approval under the provisions of 10CFR50.59.

We trust this information is satisfactory; however, if you have any questions, please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

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N. N. St. Laurent
Plant Superintendent

cc: USNRC, Region I

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YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION

(DOCKET NO 50-29)

1993 ANNUAL REPORT

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INTRODUCTION

The Yankee Nuclear Power Station, a pressurized water reactor plant previously rated at 185 MW electrical capacity, has been shut down since October 1, 1991, and the reactor permanently defueled as of February 15, 1992. The plant has been operating under a Possession Only License(POL) since August 5, 1992. The Decommissioning Plan was submitted to the NRC on December 20, 1993.

This report is submitted in accordance with 10CFR50.59(b)(2) and the YNPS Defueled Technical Specification 6.8.1. The changes identified in this report have been reviewed for, and were determined not to constitute, an unreviewed safety question as described in 10CFR50.59(a)(2).

A. Design Changes

Engineering Design Changes Requests (EDCRs)

- **EDCR 93-301 - Vapor Container Jib Crane**

  The purpose of this design change was to install a new jib crane in the Vapor Container (VC). The new jib crane is installed on the inner east face of the Reactor Support Structure (RSS) outer biological shield concrete wall above the charging floor. The jib crane is used to lift equipment (5 ton or less) from the ground through the equipment hatch, to a laydown area on the charging floor.

  Electrically, the jib crane is powered by the 480 V, three-phase distribution panel, "VC Electrical Support Panel No. 2".

  This change has no effect on plant operation except that installation of the new jib crane reduced the demand on the polar crane.

  Laydown loads on the charging floor did not increase from the original design basis. Also, stress levels in the concrete biological shield wall remained within allowable limits even when the most severe combination of simultaneous jib crane and polar crane loading configurations were postulated. The original design bases for existing plant systems and components are not adversely impacted by the installation and operation of the new jib crane.

  The new jib crane, its anchorage, and all electrical equipment are Non-Nuclear Safety (NNS) components. The RSS is also classified as NNS.

  Installation and operation of the new jib crane satisfactorily meet all original design criteria and design intent. There was no vital equipment that had to be removed from service or placed in a degraded mode of operation to accommodate installation.

- **EDCR 93-302 STEAM GENERATORS AND PRESSURIZER REMOVAL**

  This EDCR covered the removal of the four (4) steam generators and pressurizer from the Vapor Container, and the on-site preparation of these components for transport. Chem-Nuclear Systems, Inc. prepared the steam generators/pressurizer packages for transport off-site (beyond the owner-controlled boundary) in accordance with 10CFR71 requirements. An NRC approved

Safety Analysis Report (SAR) and a Certificate of Compliance (CoC) were required prior to steam generator package release for off-site transport.

The safety evaluation, addresses all aspects of the above component removals and preparation for shipment as they relate to: (1) potential impact on Yankee Plant safety-related systems, structures, and components; and (2) environmental impacts.

Both the Steam Generators and the Pressurizer were classified Non-Nuclear Safety (NNS). To ensure proper control and quality for the component removal work, however, a specified level of Quality Assurance was implemented. The packaging and transportation of the steam generators and pressurizer were classified as Requires Quality Assurance (RQA).

The modifications implemented by this EDCR had no impact on the SFP structure or pool water inventory. During steam generator and pressurizer removal, precautions were taken to preclude any interaction with the SFP and the SFP Cooling System.

The steam generator drop accident is the same general type as the fuel handling accident analyzed in the FSAR in that it is a radiological material handling accident. Both involve the handling of components with significant internal inventories of radioactive material. In addition, the doses from the steam generator drop would be well below the EPA PAG's at the Industrial Area Fence for any credible release from a steam generator drop accident.

The pressurizer drop accident is also the same general type as a fuel handling accident. However, the potential radiological consequences of a pressurizer drop accident are substantially less than the current licensing basis limits.

- **EDCR 93-305 Shield Tank Cavity Modifications**

The purpose of this EDCR was to prepare the shield tank cavity (STC) for reactor internals removal. The modifications consisted of: 1) providing a leak tight barrier around the reactor cavity T-slot seal area, 2) installing cask liner laydown pads over the STC moat area, and 3) removing drive shaft/guide tube storage racks, and removing miscellaneous components from the shield tank cavity liner.

Additionally, removal of the storage racks, control rod cutting stands, miscellaneous cavity liner attachments, and installation of liner laydown pads increased laydown space necessary to facilitate reactor internals removal work.

The modifications performed within this EDCR are classified as NNS.

To provide a leak tight cavity seal, a split and rolled pipe has been installed over the STC T-slot, and attached to the reactor cavity seal flange and the NST cavity seal flange by welding. This new seal was designed with consideration of both hydrostatic and bending loads and acts like a bellows to accommodate this loading.

The cask liner laydown pads are temporary components placed in the STC moat area to provide additional storage space for cask liners. The pads have been designed to accommodate the full payload weight of a 3-55 or 8-120 cask liner.

The original function of the drive shaft/guide tube storage racks was to provide a storage location for vessel components during refueling outages. Removal of the racks, control rod cutting stands and miscellaneous threaded rod attachments were necessary to facilitate reactor internals removal.

B. Non-Nuclear Safety Changes (NNS Changes)

- **NNS Change 92-004 - YNPS Auxiliary Service Water Pump**

This change removed the Service Water Booster Pump from the PAB and relocated the pump as the Auxiliary Service Water Pump in the Screenwell House.

The Pump was installed parallel to the existing main service water pump and allows the use of a 25 hp pump in lieu of the 125 hp service water pump.

The new pump is better suited to meet the reduced cooling demands of the permanently shutdown nuclear plant, however, installation of this pump was not designed to replace the service water pumps in their safety related function.

C. Temporary Design Changes (TCR)

- **TCR #93-46 - Removal of # 2 Emergency Diesel Generator (EDG) Output Breaker**

  This TCR provided the means to remove the 480 volt EDG output breaker. EDG #2 is isolated from other plant equipment both electrically and mechanically. EDGs #1 & #3 remain in service. Only one EDG is required to meet the needs of the Spent Fuel Pit Cooling system with the plant in a permanently shutdown condition.

- **TCR #93-47 - Removal of High Pressure Safety Injection Pump Breakers**

  This TCR provided the means to remove the High Pressure Safety Injection Pump 480 volt Breakers, which are no longer required due to the plant shutdown and the plant being in a permanently shutdown condition.

- **TCR #93-51 - Removal of # 2 EDG Breakers**

  This TCR provided the means to remove the 480 volt Bus Tie breaker (BT2B) and both Alternate Feed breakers for EMCC #1 and EMCC #2. EDG #2 was isolated from other plant equipment both electrically and mechanically by TRC #93-46.

- **TCR #93-54 - Removal of the Blank Flange for the Reactor Vent Line.**

  This TCR provided the means to remove a blank flange from the reactor cavity vent line. The flange was removed to investigate leakage from the Neutron Shield Tank telltale. Following the investigation, the TCR was removed and the flange re-installed.

- **TCR #93-66 Removal of Security Diesel**

  The plant's Security Diesel Generator (SDG) was evaluated and determined to be no longer required as a backup power supply. This TCR provided the means to disconnect the SDG as the backup power source and replaced with the Mass Electric line as a substitute backup supply to the gatehouse.

D. Other Changes

- **Winter Operation of Turbine Building Fire Suppression System**

  The safety evaluation provides the basis and justification for isolating the following fire suppression systems: Turbine Building column sprays, CREACS spray, and the deluge for the transformer oil cooler and hydrogen seal oil. This action was done in conjunction with the plant energy conservation program, reducing the Turbine Building heating demand by only heating the volume below the turbine floor. The fire system header supplying these systems was isolated and drained to prevent freezing. Actuation of the system requires manual action.

  The charcoal in the CREACS filter was removed from the filter assembly, since the system is no longer required in the permanently defueled condition. The system was not credited in the Defueled Safety Analysis.

- **Battery Chargers: Simplification of the Surveillance Requirement**

  The safety evaluation provides the basis and justification for a simplified battery charger functional test since the full load tests are no longer required. The simplified functional test (1) verifies that the chargers can respond to a sudden current demand and (2) adequately demonstrates the operating status of the chargers. Only one of four plant chargers is required in the permanently defueled condition.

- **Spent Fuel Pit (SFP) Cooling Operation**

  The safety evaluation provides the basis and justification for intermittent operation of the SFP cooling system. The thermal cycling caused by intermittent operation was determined to have no adverse impact on the integrity of the SFP liner or the water inventory maintenance capability. Intermittent operation cycles the pool temperature from 70 to 100 degrees Fahrenheit.

- **Contaminated Pipe Sampling and Processing**

  The safety evaluation provides the basis and justification for the removal of a section of Component Cooling pipe for processing at a decontamination facility. The section of pipe was removed from the non-nuclear safety section of the system and is not required to support decommissioning activities or plant operation. The results of the decontamination effort will be used to evaluate decommissioning options.