Docket File

Safety Evaluation Report

U. S. Nuclear Regulatory Commission

related to operation of

Office of Nuclear Reactor Regulation

Salem Nuclear Generating Station, Units 1 and 2

Docket Nos. 50-272 and 50-311

August 1976

Public Service Electric and Gas Company, et al.

Supplement No. 2

August 13, 1976

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

PUBLIC SERVICE ELECTRIC AND GAS COMPANY,

PHILADELPHIA ELECTRIC COMPANY,

DELMARVA POWER AND LIGHT COMPANY, AND

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION,

UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-272 AND 50-311

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1.0 INTRODUCTION

On October 11, 1974, the Nuclear Regulatory Commission (the Commission) issued its Safety Evaluation Report regarding the application by the Public Service Electric and Gas Company, et al (applicants) for licenses to operate the Salem Nuclear Generating Station, Units No. 1 and No. 2 (Salem facility). On June 28, 1976, the Commission issued Supplement No. 1 to the Safety Evaluation Report providing evaluations of matters not included in the Safety Evaluation Report, and identifying certain outstanding issues that required resolution prior to the issuance of an operating license.

This supplement is in support of our conclusions relative to the decision for issuing an operating license for Salem Nuclear Generating Station, Unit No. 1. This supplement therefore includes a discussion of those matters identified in Supplement No. 1 as requiring resolution prior to the issuance of an operating license and other matters relevant to issuing an operating license.

By letter dated July 14, 1976, and a subsequent letter dated July 20, 1976, the applicants have identified certain items for which they request a deferment of completion until after core fuel loading and other items until after initial criticality has been attained. We have reviewed the safety considerations associated with the deferment of each item, and we are conditioning the operating license for the Salem Nuclear Generating Station, Unit No. 1 to allow the deferment of completion of those items whose completion has been demonstrated to not be required for safety purposes. These deferments notwithstanding, power operation shall not be undertaken until each deferred item has been acceptably completed and verified by our Office of Inspection and Enforcement.

Each section of this supplement is numbered the same as the section of the Safety Evaluation Report and Supplement No. 1 to the Safety Evaluation Report that is being updated. The material contained in this report, therefore, is supplementary to and not in lieu of the discussion in the Safety Evaluation Report. Appendix A is a continuation of the chronology of the principal actions related to the processing of the application.

Our Office of Inspection and Enforcement has informed us that the axial flux difference monitor alarms used to show compliance with Section 3.2.1 of the Technical Specifications, and which we consider necessary to the acceptable implementation of constant axial offset control, have not been provided. Therefore, until such time as these alarms are acceptably implemented, we are conditioning the operating license for the Salem Nuclear Generating Station, Unit No. 1 to a power operation level not to exceed 50 percent of rated power, the power level permitted by Section 3.2.1 of the Technical Specifications without axial flux difference monitor alarms.

In light of the recommendations of our Office of Inspection and Enforcement and in conjunction with the deferring of certain items until after initial core fuel loading and other items until after initial criticality, and considering the status of the axial flux difference alarms, we are conditioning the operating license for the Salem Nuclear Generating Station, Unit No. 1 to permit the applicants to proceed as follows:

1. The applicants may at the license issue date proceed directly to Operational Mode 6 (initial fuel loading) and subsequently proceed to Operational Mode 5 (cold shutdown) and Operational Mode 4 (hot shutdown). Operation beyond these modes shall not proceed until the prerequisite items delineated in the operating license and the applicable subcritical tests have been acceptably completed and verified by our Office of Inspection and Enforcement.

2. Upon completion of the prerequisite items mentioned above, the applicants may proceed to Operational Mode 3 (hot standby) and Operational Mode 2 (initial criticality or startup). Operation beyond these modes shall not proceed until the prerequisite items delineated in the operating license and the applicable initial criticality and startup tests have been acceptably completed and verified by our Office of Inspection and Enforcement.

3. Upon completion of the prerequisite items mentioned above, the applicants may proceed to Operational Mode 1 (power operation). Operation in this mode shall be initially limited to 50 percent of rated thermal power until the axial flux difference alarms required for constant axial offset control have been acceptably implemented and verified by our Office of Inspection and Enforcement. Upon the acceptable implementation of the axial flux difference alarms, power operation may proceed to 100 percent of rated thermal power.

We have reviewed the recommendations of our Office of Inspection and Enforcement and have concluded that all items of construction and testing necessary for fuel loading and subcritical testing have been acceptably completed. In addition, subject to the acceptable completion of the deferred items and the acceptable implementation of the axial flux difference alarms, we also conclude that the Salem Nuclear Generating Station, Unit No. 1 can be operated at up to 100 percent of rated thermal power without undue risk to the health and safety of the public.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.9 Mechanical Systems and Components

3.9.4 Supports - Steam Generators, Reactor Coolant Pumps and Pressurizer

Sun Shipbuilding and Dry Dock Company (Sun Shipbuilding), the fabricator of the supports for the steam generators and reactor coolant pumps used at North Anna, Units No. 1 and No. 2 (Docket Nos. 50-338 and 50-339) has made allegations concerning the structural acceptability of these supports with respect to materials and design. Since the supports for the steam generators, reactor coolant pumps and pressurizer used at the Salem facility are somewhat similar to those used at North Anna with respect to configuration and materials of construction, and since Sun Shipbuilding fabricated the supports for the steam generators used at the Salem facility, the staff has been pursuing the applicability of these allegations to the Salem facility.

Briefly, the North Anna situation is as follows. Subsequent to delivery of the supports to the plant site, the supports were found to have numerous weld cracks and eventually all welds were cut out and repaired. The allegations made by Sun Shipbuilding assert that the problems encountered in fabrication stemmed from the inherent restraint of the welds in the supports, and that the unavoidable presence of cracks, coupled with a lack of fracture toughness and a propensity for lamellar tearing in the ASTM A-36 material specified, combine to result in an unacceptable structure for the intended service due to the potential for brittle fracture.

The staff investigated the design and fabrication history of the supports for the steam generators, reactor coolant pumps, and pressurizers used at the Salem facility. As a result of the investigation, the staff has identified the following significant conditions and factors that alleviate the concern for potential brittle fracture of these supports. The material of construction for the supports of the steam generators and reactor coolant pumps used at the Salem facility was A441-68, whereas the material of construction for these supports at North Anna was ASTM A-36 and A-572. A supplementary requirement for Charpy V notch testing (20 foot pounds minimum at 20 degrees Fahrenheit) was imposed on the support material used at the Salem facility and the material used met this requirement with ample margin, whereas no supplementary testing requirements were originally imposed on the North Anna supports.

Onsite inspections of the Salem facility supports by the magnetic particle method have shown only minor surface defects, none of which were critical from the standpoint of structural integrity. Despite the non critical aspects of these surface defects, all such indications were removed. Other phenomena, aside from the effect of significant flaws that could lead to concern of failure due to brittle fracture, are not present at the temperatures of service conditions for the supports during plant operation. Since the minimum service temperature of the supports is about 90 degrees Fahrenheit, the probability of brittle behavior at this temperature is very remote because adequate fracture toughness is available. Additionally, since the design of the supports at the Salem facility results in the supports being loaded almost entirely in compression during normal plant operation, fatigue crack initiation and propagation would not occur. The supports for the Salem facility have been designed to a conservative stress limit of 90 percent of the minimum required yield strength for the design basis accident, i.e., the postulated instantaneous rupture of a primary coolant pipe.

In light of the markedly different fabrication history with respect to the North Anna supports, the absence of significant flaws as demonstrated by inspections, the compressive loading to be experienced in service, the related absence of significant fatigue growth forcing functions and the relatively high service temperature, the extent of the toughness testing performed and the adequacy of the results of that testing, we conclude that the supports for the steam generators and coolant pumps installed at the Salem facility are acceptable for service.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.6 Steam Generator Head Cladding

We stated in Supplement No. 1 to the Safety Evaluation Report that the applicants had provided information regarding metallurgical indications in the stainless steel cladding in the heads of the Salem Nuclear Generating Station, Unit No. 1 steam generators. We also stated that the applicants were to verify that the cracks are confined to the cladding, and that we would report on our evaluation of this matter in a future supplement.

By letter dated June 30, 1976, the applicants submitted a report which provides additional information regarding these metallurgical indications. The submitted report includes (1) a description of the ultrasonic examinations of the cladding cracks in the No. 14 steam generator, (2) results of the examinations and conclusions, and (3) a proposed inservice inspection program to evaluate crack propagation. The cladding cracks were examined from the outside diameter surface, and the applicants conclude that the cracks examined do not penetrate into the base metal.

Although evidence of crack extension into the base metal was not detected, we have conservatively postulated, as we did for Indian Point Unit No. 3 (Docket No. 50-286), that a corrosion of 0.075 inch into the base metal would exist after forty years of service life. Since this conservatively postulated corrosion of 0.075 inch is considerably less than one-tenth the critical flaw, and since it is reasonable to expect that the postulated corrosion penetration into the base metal would be in the form of rounded pitting rather than as a sharp discontinuity, we conclude that the integrity of the channel heads will not be affected as a result of corrosion assisted fatigue.

Based on our review of the information submitted by the applicants, we conclude that operation of the Salem Nuclear Generating Station, Unit No. 1 with the cladding on the steam generator channel heads in their present condition is acceptable. To ensure that the integrity of the channel heads is not affected by plant operation, we require that an inservice inspection program be implemented to monitor the cladding cracks on a schedule consistent with the first three refueling outages as proposed by the applicants in their letter of June 30, 1976, i.e., the area to be monitored, and the equipment and procedures to be used will be the same as those used to generate the baseline data submitted with the report of June 30, 1976. In addition to monitoring the condition of the cladding after plant operation, we require that the augmented inservice inspection program include provisions to videotape the 100 percent visual inspection of the steam generator channel heads with a television camera. The technical specifications have been revised to require the performance of the above inservice inspection program in a manner acceptable to the staff.

Based on our evaluation, we conclude that operation of the Salem Nuclear Generating Station, Unit No. 1 with steam generator channel heads in their present conditions will not create undue risk to the health and safety of the public, and is therefore acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3.2 Single Failure Criterion

In Supplement No. 1 to the Safety Evaluation Report, we identified several motor operated valves which required design modifications to meet the single failure criterion. The principal modification was to incorporate the ability to restore power to the following valves from the control room: 11SJ44, 12SJ44, 11SJ49, 12SJ49, 1SJ67 and 1SJ68. We stated in Supplement No. 1 to the Safety Evaluation Report that our Office of Inspection and Enforcement would verify that these modifications had been implemented prior to our approval of plant startup. Our Office of Inspection and Enforcement has confirmed that the above modifications have been implemented; therefore, we consider this matter resolved.

6.3.4 Tests and Inspections

We stated in Supplement No. 1 to the Safety Evaluation Report that the functional flow capability of the emergency core cooling system would be considered acceptable after demonstrations by the applicants of proper actuation and flow delivery of all components, and that the actuation times of components and flow deliveries meet or exceed the values assumed in the Final Safety Analysis Report. We further stated that our Office of Inspection and Enforcement would verify that the requirements have been satisfied prior to approval of plant startup as defined in the technical specifications. Our Office of Inspection and Enforcement has confirmed that the above requirements have been met; therefore, we consider this matter resolved.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth in the Safety Evaluation Report, and in Supplement No. 1 and this supplement to the Safety Evaluation Report, we reaffirm our conclusions as stated in the Safety Evaluation Report.

In addition, we conclude that the prerequisite items of construction and testing have been acceptably completed such that fuel loading and subcritical testing of the Salem Nuclear Generating Station, Unit No. 1 can be conducted without undue risk to the health and safety of the public. We further conclude that, subject to the acceptable completion of the deferred items identified in the operating license and the acceptable implementation of the axial flux difference monitor alarms, that the Salem Nuclear Generating Station, Unit No. 1 can be operated at up to 100 percent of rated power without undue risk to the health and safety of the public.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW OF SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2

June 25, 1976	Letter from applicant transmitting Reactor Containment Building Integrated Leak Rate Test Report and Structural Test Report
June 28, 1976	Supplement No. 1 to Safety Evaluation Report issued
June 30, 1976	Letter from applicant transmitting report regarding ultrasonic examination of steam generator clad cracking
July 14, 1976	Letter from applicant transmitting list of items that will not be completed prior to scheduled core load date
July 15, 1976	Letter from applicant concerning participation in augmented startup test program
July 19, 1976	Letter from applicant requesting exemption for Unit No. 1 to certain code requirements of 10 CFR Part 50.55a
July 20, 1976	Letter from applicant providing additional information concerning incomplete items
July 20, 1976	Letter from applicant transmitting Revision 2 to Security Plan
July 28, 1976	Letter from applicant transmitting Annual Reports
July 30, 1976	Submittal of Amendment No. 39, consisting of revised and additional information
July 30, 1976	Letter from applicant providing list of additional items to be deferred until after core loading