North Atlantic

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

LICENSE AMENDMENT REQUEST NO. 97-01
"DESIGN FEATURES - FUEL ASSEMBLY RECONSTITUTION"
(TAC M97929)

This License Amendment Request is submitted by North Atlantic Energy Service Corporation pursuant to 10CFR50.90. The following information is enclosed in support of this License Amendment Request:

- Section I - Introduction and Safety Assessment for Proposed Changes
- Section II - Markup of Proposed Changes
- Section III - Retype of Proposed Changes
- Section IV - Determination of Significant Hazards for Proposed Changes
- Section V - Proposed Schedule for License Amendment Issuance and Effectiveness
- Section VI - Environmental Impact Assessment

Sworn and Subscribed
before me this
18TH day of FEBUARY, 1997

Robert D. Erola
Notary Public

Ted C. Feigenbaum
Ted C. Feigenbaum
Executive Vice President and Chief Nuclear Officer

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THE FOLLOWING INFORMATION IS PROVIDED IN SUPPORT OF LICENSE AMENDMENT REQUEST - LAR 97-01

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

Refer to the enclosed introduction and safety assessment of the proposed changes to the Technical Specifications addressed by this license amendment request.

II. MARKUP OF PROPOSED CHANGES

Refer to the attached markup of the proposed changes to the Technical Specifications. The attached markup reflects the currently issued revision of the Technical Specifications listed below. Pending Technical Specifications or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

The following Technical Specification is included in the attached markup:

Technical Specification	Title	Page(s)
TS 5.3.1	Fuel Assemblies	5-9

III. RETYPE OF PROPOSED CHANGES

Refer to the attached retype of the proposed changes to the Technical Specifications. The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

Refer to the enclosed determination of significant hazards for the proposed changes to the Technical Specifications addressed by this license amendment request.

V. PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

North Atlantic requests NRC review of License Amendment Request 97-01 and issuance of a license amendment by May 1, 1997, having immediate effectiveness and implementation required within 60 days. Issuance of a license amendment by the proposed date will support the upcoming refueling outage, OR05, presently scheduled for May 10, 1997.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

North Atlantic has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor increase the types and amounts of effluent that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, North Atlantic concludes that the proposed change meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

Section I

Introduction and Safety Assessment for the Proposed Changes

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

License Amendment Request (LAR) 97-01 proposes a change to the reactor core fuel assembly design features requirements contained in Technical Specification 5.3.1, "Fuel Assemblies". The proposed change will allow the use of solid stainless steel or zirconium alloy filler rods in fuel assemblies to replace failed or damaged fuel rods, provided that the new fuel assembly configurations are analyzed with applicable NRC approved codes and methods. This submittal is in accordance with the guidance provided in Generic Letter 90-02, Supplement 1, "Alternative Requirements For Fuel Assemblies In The Design Features Section of Technical Specifications".

B. Safety Assessments of Proposed Changes

BACKGROUND

During the current operating cycle, Cycle 5, Seabrook Station has recently detected an increase in primary coolant activity indicative of one or more fuel rod cladding defects. This is the first time in Seabrook Station's operating history of indication of "leaking" fuel rod(s) within the reactor core. Leaking fuel rods are not uncommon occurrences at operating nuclear power stations and on occasions failure of individual fuel rods may occur during routine operation of the plant or during movement of fuel assemblies. Because fuel rod failures occur on occasion the nuclear industry has developed methods to effectively deal with such occurrences, within currently established regulatory guidelines. Current industry practice is (usually) not to reload fuel assemblies which are known to contain failed fuel rods, as the cladding defects of such rods allow fission products to be released to the primary coolant. An increase of fission products within the primary coolant has the potential of creating radiation hot spots as well as increasing the general radiation levels throughout the plant. Long term operation with failed fuel rods is contrary to ALARA goals and principles. However, fuel assemblies which have failed fuel rods can be reconstituted and safely be reused, by replacing failed fuel rods with solid filler rods made from stainless steel or a zirconium alloy.

Because the Design Features section of the Technical Specifications currently precludes the use of fuel assemblies containing solid filler rods, a change to the Design Features section of the Technical Specifications is being requested. The change would permit timely removal of fuel rods that are either found to be leaking during a refueling outage or are determined to be probable sources of future leakage. In place of the damaged fuel rods, either solid stainless steel or zirconium alloy filler rods will be substituted. The proposed change provides flexibility in the repair of fuel assemblies containing damaged and leaking fuel rods by reconstituting the assemblies.

This request is consistent with the model technical specification outlined in NRC Generic Letter 90-02, Supplement 1, "Alternative Requirements For Fuel Assemblies In The Design Features Section Of Technical Specifications", and is generally consistent with the format and content of the improved Standard Technical Specifications for Westinghouse plants provided in NUREG-1431. In addition, the Technical Specifications (TS) will reflect fuel rod geometry which is consistent with the wording in 10CFR 50.46, and retain the original part of TS 5.3.1 specifying the fuel enrichment limit, which indirectly quantifies the amount of special nuclear material in use consistent with Section 182.a of the Atomic Energy Act of 1954, as amended.

Similar proposed Technical Specifications changes have been granted to Catawba Units 1 & 2.

GENERAL

In accordance with Supplement 1 to Generic Letter 90-02, the requested change to Seabrook Station Technical Specification (TS) 5.3.1 will allow the use of solid stainless steel or zirconium alloy filler rods in fuel assemblies to replace failed or damaged fuel rods, provided that the new fuel assembly configurations are analyzed with applicable NRC approved codes and methods. The Safety Significance portion of this evaluation identifies the mechanical, neutronic and thermal-hydraulic analyses which are affected by the presence of the solid stainless steel and zirconium alloy filler rods. In addition, a description is provided of the methodology which will be used to assess the impact of such filler rods on the affected analyses, and to demonstrate that all applicable design criteria and pertinent licensing basis acceptance criteria are satisfied.

The request for a change to the Technical Specifications for Seabrook Station Unit 1 is consistent with the guidance in Supplement 1 to NRC Generic Letter 90-02. No specific numerical limits are placed on the number of solid stainless steel or zirconium alloy filler rods which may be used in a single assembly, or on the number of reconstituted assemblies which may be used in a core. However, the use of filler rods will be limited to configurations which have been analyzed with NRC approved codes and methods. The methodology for analyzing reconstituted fuel assemblies is discussed in the Safety Significance section of this evaluation.

The use of solid stainless steel or zirconium alloy filler rods in fuel assemblies is further constrained by other requirements such as the fuel assembly mechanical design criteria. Technical Specifications which limit parameters potentially affected by the use of reconstituted fuel, such as core operating power and peaking factors, also inherently place constraints on the locations and numbers of solid stainless steel and zirconium alloy filler rods which can be used. As discussed in the Safety Significance section, all fuel design and performance criteria will continue to be satisfied, and all pertinent licensing basis criteria will be met for the use of reconstituted fuel at Seabrook Station.

SAFETY SIGNIFICANCE

1.0 Fuel Reconstitution

1.1 Mechanical Design

Access to the failed rods in a fuel assembly will be gained by removal of either the top or bottom fuel assembly fitting. The removed end fitting will be reattached using an attachment design (e.g., locking cup thimble screws on the bottom nozzle or insert lock tubes in the top nozzle) which is similar to that used on the fitting when the assembly was originally manufactured.

From the perspective of compliance with the fuel assembly mechanical design and fuel coolability requirements specified in Section 4.2 of the Standard Review Plan (NUREG-0800, Rev. 2), the two criteria which are impacted by fuel reconstitution are the fuel assembly hold-down force and the fuel assembly structural response to seismic and LOCA loads. The design basis for the fuel assembly hold-down forces requires that the fuel assembly not be allowed to lift due to flow during normal operating conditions. The basis for the fuel assembly structural response to seismic and LOCA loads is given in Appendix A to Section 4.2 of the Standard Review Plan.

The impact of fuel reconstitution on the fuel mechanical design has been assessed by the fuel vendor, Westinghouse. It was demonstrated that even for replacement of a significant fraction of the fuel rods in an assembly with solid stainless steel or zirconium alloy filler rods, the mechanical effects of reconstitution are acceptable for all Westinghouse fuel assembly designs. Details of this evaluation are documented in Reference 1.

The fuel rod mechanical design criteria are indirectly affected by fuel reconstitution, because the displacement of active fuel rods by solid stainless steel or zirconium alloy filler rods slightly increases the core average linear heat generation rate. Those fuel rod design criteria which could be affected by this change will be evaluated as part of the cycle specific fuel rod design evaluation.

All reconstituted fuel assemblies scheduled for further use will continue to satisfy the minimum fuel assembly and fuel rod mechanical design requirements.

1.2 Nuclear Design

A nuclear design evaluation, consistent with the approved methodology described in References 4 and 5, is performed on a cycle specific basis to demonstrate that a fuel reload will meet all applicable design criteria. The core configuration is reviewed each cycle to confirm that either existing safety analyses are applicable or a cycle specific analysis is performed. This safety evaluation process is discussed in more detail in Section 1.4 below.

Reconstituted assemblies will be incorporated into core loading plans as normal assemblies. Once a decision is made on which reconstituted assemblies will be used in a specific reload, the appropriate core physics models will be applied to reflect the actual geometry of the reconstituted assemblies in that reload cycle.

Fuel assembly reconstitution affects reactivity and local peaking due to the redistribution of power within the assembly. In the nuclear design analysis for each reload, reconstituted assemblies will be explicitly modeled in a conservative manner on a pin-by-pin basis to evaluate the effect on local power peaking and core-wide reactivity parameters (i.e., critical boron concentration and boron coefficient). If the effect is significant, it will be reflected in all phases of the design and safety analysis by either explicit calculations or additional uncertainties, as appropriate, to ensure that the reconstituted assemblies are treated in a conservative manner.

1.3 Thermal and Hydraulic Design

The thermal-hydraulic evaluation of reconstituted fuel assemblies is performed in accordance with the thermal-hydraulic methodology described in References 2 and 3, using NRC approved codes and methods. The evaluation will assess the safety significance of fuel rod reconstitution and will ensure that a core containing reconstituted fuel meets the same design criteria which are applicable to existing fuel designs.

Fuel rod reconstitution affects predictions of departure from nucleate boiling (DNB) in hot channels due to a local power reduction and the resultant effects on enthalpy and flow. The DNB effects of fuel reconstitution will be evaluated on a cycle specific basis to account for the local power reduction which occurs when fuel rods are replaced with solid stainless steel or zirconium alloy filler rods. Reference 1 shows that it is conservative to model a reconstituted assembly as a regular fuel assembly in DNB analyses.

The results of the evaluation in Reference 1 are applicable to the methods that will be used for Seabrook Station. The thermal-hydraulic evaluation methodology for Seabrook Station, performed by Yankee Atomic Electric Company (YAEC), is based on the Westinghouse WRB-1 DNB correlation methodology (Reference 2). The Westinghouse WRB-1 correlation has been used with the YAEC VIPRE models for Seabrook Station to perform DNB safety analyses for the use of Westinghouse fuel at Seabrook Station.

The DNB design basis defined in Reference 2 remains applicable for reconstituted cores. As discussed in Reference 1, cycle specific evaluations will be performed for future reload cores using reconstituted fuel. These evaluations, which will consider the exact configuration and associated core power distribution of the reconstituted assemblies, will confirm the conservative assumption that a reconstituted assembly is bounded by a regular fuel assembly in DNB analyses.

1.4 Non-LOCA Evaluation

Reload safety analyses are performed for each reload core to determine if the reload core configuration is bounded by existing safety analyses. The inputs and assumptions used in each accident analysis, documented in Chapter 15 of the UFSAR, are reviewed for continued applicability for the reload core. When a key analysis parameter is not bounded, further analysis is considered necessary to ensure that the required safety margin is maintained. This determination is made either through a complete reanalysis of the accident, or through a simpler, though conservative, evaluation process using parameter sensitivities. Any reanalysis will follow standard procedures and employ models and methods which have been used and approved in previous submittals to the NRC.

The cycle specific non-LOCA safety evaluations are designed to identify any changes in the fuel design which may invalidate existing safety analyses. Therefore, the cycle specific Reload Safety Evaluation process will ensure that the impact of the use of reconstituted fuel assemblies is evaluated for the appropriate non-LOCA analyses.

1.5 LOCA Evaluation

As discussed in Reference 5, the normal reload design methodology for Seabrook Station assures that fuel parameters which are significant to analyses of the Emergency Core Cooling System either remain applicable to or bounded by the parameter values used in the LOCA analysis. LOCA analyses for Seabrook Station are performed by Westinghouse. The discussion of the impact of fuel reconstitution on LOCA evaluations in Reference 1 is therefore directly applicable to the use of reconstituted fuel at Seabrook Station.

The impact of fuel reconstitution on the LOCA methodology is defined in detail in Reference 1, and may be summarized as follows:

1) If the failed fuel rods are replaced by fuel-bearing rods, the enrichment of the replacement rods is tailored to approximate the power generation of the rods being replaced. Typically, there will be no effect on the LOCA analysis of record as the result of this type of fuel reconstitution.

2) For replacement of failed fuel rods with solid stainless steel or zirconium alloy filler rods:

    a. A conservative peak clad temperature penalty will be assessed to account for the possible steady state effects on the Large Break LOCA only. The magnitude of this penalty is defined in Reference 1, and is proportional to the number of reconstituted rods in the fuel assembly.

    b. Based on the number of failed fuel rods to be replaced, the potential increase in linear heat rate to retain total core power constant at its rated value will be calculated. This change will be incorporated into both the Large Break and Small Break LOCA analyses.

Evaluations of the change in linear heat rate will be performed to assess the Peak Clad Temperature (PCT) penalty for the number of rods to be reconstituted.

3) The total PCT effect will be determined by adding the contributions described above. This total effect will then be added to the sum of the evaluations performed to date for the current plant configuration. This reconstitution penalty will be tracked throughout the core residence of the affected assemblies, and will be removed from the plant assessment against PCT margin when the reconstituted assemblies are removed from the core.

Presuming ongoing tracking of the reconstituted assemblies and available margin, the use of reconstituted fuel can be addressed on a cycle specific basis under the auspices of 10 CFR 50.59 with respect to the LOCA analyses.

No changes are being made to any design or safety related limit in conjunction with this Technical Specifications change. Confirmation that the limits are satisfied will continue to occur as part of the Reload Safety Evaluation process.

C. References

1. Slagle, W. H. (Editor), "Westinghouse Fuel Assembly Reconstituted Evaluation Methodology," WCAP-13060-P-A (Proprietary), July 1993.

2. YAEC-1849PA, "Thermal Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications", October 1992

3. YAEC-1854PA, "Core Thermal Limit Protection Function Setpoint Methodology for Seabrook Station", October 1992

4. YAEC-1363-A, "CASMO-3G Validation", April 1988

5. YAEC-1659-A, "SIMULATE-3 Validation and Verification", September 1988

Section II

Markup of Proposed Changes