5.2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet.

b. Annulus nominal volume = 543,000 cubic feet.

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet.

d. Nominal inside diameter = 148 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.5 feet.

g. Dome inside radius = 112 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

Fuel assemblies shall be limited to those designs that have been analyzed using NRC approved methodology and shown by tests or analyses to comply with fuel design and safety criteria.

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of between 134.1 and 136.7 inches, ~~and contain a maximum total weight of 2250 grams uranium.~~ Individual fuel assemblies shall contain fuel rods of the same nominal active fuel length. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

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Alternative Requirements for Fuel Assemblies

ATTACHMENT 2

SAFETY ANALYSIS

Introduction

This proposed license amendment revises the St. Lucie 1 Technical Specifications Design Features Section 5.3.1 to permit the use of fuel assembly designs that are not limited to a maximum uranium weight. Reload fuel assemblies will be limited to those designs that have been analyzed using an NRC approved methodology and shown by tests or analyses to comply with all applicable design and safety criteria.

The fuel assembly description in the Design Features Section of the Technical Specifications provides a description of the required characteristics of reload fuel. The St. Lucie 1 Technical Specification Design Features Section 5.3.1 requires that each fuel rod in a fuel assembly shall contain a maximum total weight of 2250 grams uranium. However, the Cycle 11 reload was designed, analyzed and fabricated to contain approximately 2275 grams of uranium per fuel rod. The purpose of this request is to revise this section of the Technical Specifications to permit the use of fuel assembly designs that are of similar physical design to the initial core loading, but are not limited by an unnecessary maximum fuel rod uranium weight requirement.

The requirement of a maximum fuel rod uranium weight is unnecessary because changes to the characteristics of the fuel rod (including uranium weight) that can impact design and safety criteria are specifically analyzed during the reload evaluation process. These evaluations, using NRC approved methodology, assure that applicable design and safety analysis criteria are met. Additionally, compliance of the design with the Limiting Safety System Settings and the Limiting Conditions for Operation in the Technical Specifications is demonstrated during the reload evaluation process. Therefore, the proposed amendment will not adversely impact the safe operation of St. Lucie Unit 1.

TECHNICAL DISCUSSION

Changes to the characteristics of the fuel rod/assembly that can impact design criteria, safety analysis criteria or safety limits are specifically analyzed for each reload, using NRC approved methodology, to assure that applicable criteria or limits are not violated. These analyses also assure that plant operation with the reload fuel assemblies comply with the Safety Limits and Limiting Conditions For Operation in the Technical Specifications.

An example of this design process is the St. Lucie Cycle 11 reload where several changes to the fuel rod design were incorporated into the reload fuel assemblies. The fuel rod design changes consisted of the following:

1. The pellet diameter was increased from 0.370 to 0.377 inches.

2. The pellet density was increased from 94% to 95% theoretical UO_2 density.

3. The clad thickness was reduced from 0.031 to 0.028 inches (identical to the initial core).

4. The pellet-clad gap was reduced from 0.0080 to 0.0070 inches.

5. The active fuel height was increased from 134.1 to 136.7 inches (identical to the initial core). This is accomplished by increasing the top natural uranium axial blanket from 6.0 to 8.64 inches.

6. The plenum spring length was reduced from 8.800 to 5.206 inches to accommodate the increased active fuel length.

7. Fuel rod helium fill gas pressure was increased from 290 to 330 psig.

Although the design of the Cycle 11 reload fuel was similar in physical characteristics to that of the fuel initially loaded into the reactor, the changes resulted in an increased fuel rod uranium weight (approximately 1% increase in the fuel rod weight). Significant aspects of the changes were evaluated to show compliance with applicable design and safety criteria. Other secondary aspects such as: structural impact on the reactor internals, vessel supports and spent fuel pool, and spent fuel heat load, were qualitatively evaluated and deemed to be insignificant. The key results and conclusions are discussed below:

a) The reduced gap width, the decrease in cladding thickness, the increase in fuel theoretical density and the increase in fill gas pressure necessitated a re-analysis/evaluation of the Large Break and Small Break LOCA events. The results demonstrated that all 10 CFR 50.46(b) criteria were met.

b) The increase in the heated length of the fuel rod and its impact on the Minimum Departure from Nucleate Boiling Ratio (MDNBR) was explicitly evaluated in the Thermal Margin/Low

Pressure and the DNB/LCO (Limiting Condition For Operation) verification analyses for Cycle 11. The results demonstrated that the current setpoints provide sufficient margin to DNB.

c) The impact of the reduction in gap width on the hot rod gap conductance throughout the cycle and its effect on Anticipated Operational Occurrences (AOO) was evaluated. Evaluation of the limiting DNB AOO, Loss of Flow, demonstrated that the reference analysis remains bounding for Cycle 11.

d) The impact of the design changes on the core physics parameters were explicitly modeled. The results demonstrated that the key parameters met applicable design and safety criteria, and Technical Specifications. For example, peak linear heat rate and radial peaking factor values of 13.4 kw/ft and 1.59, respectively, were calculated. The corresponding Technical Specification limits are 15.0 and 1.70. Excess shutdown margin of 1406 pcm was calculated. The Moderator Temperature Coefficient was calculated to be within the Technical Specification limits at all times during Cycle 11 operation.

e) Integrity of the new fuel rod design during normal operation and Anticipated Operational Occurrences was confirmed by a detailed mechanical performance analysis. It was concluded that:

- the maximum steady-state cladding strain was well below the 1% design limit,
- the maximum steady-state cladding stresses met the ASME Boiler and Pressure Vessel Code Requirements,
- the transient circumferential strain was within the 1% design limit,
- the transient stress calculated during power ramps (up to the maximum allowable peaking factor) was within the 56 ksi design limit,
- cladding creep collapse was precluded,
- the fuel rod pressure remained below the design criteria of system pressure plus 800 psi throughout life,
- the maximum local cladding oxidation was below the 130 micron limit,
- the cladding fatigue usage factor was below the 0.67 design limit,

- the fuel temperature remained below the melting temperature and the clad total uniform strain remains below 1% for the AOO condition.

f) Radiological consequences for each limiting event were evaluated against 10 CFR 100 criteria and found to be bounded by the results of previous analysis.

In conclusion, the deletion of the maximum rod weight in the Design Features Section 5.3.1 of the Technical Specifications on Fuel Assemblies will permit changes in rod uranium weight while maintaining similarity in physical design to that of the initial core. Any changes in the characteristics of the reload fuel assemblies will be limited to those designs that have been analyzed using an NRC approved methodology and shown by tests or analyses to comply with all applicable design and safety criteria.

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ATTACHMENT 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92. 10 CFR 50.92 states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The amendment will not increase the probability of an accident because it does not change the plant operating modes or the requirement that the reload fuel be similar in physical design to the initial core loading. This requirement ensures that the fuel assembly outside dimensions and interface with core internals and other plant equipment remain the same. This results in no change in the handling and operation of the fuel assemblies that would increase the probability of an accident. Additionally, the consequences of any previously analyzed accident will not be significantly increased since any changes to the fuel assembly design will continue to be evaluated using NRC approved methodology to demonstrate compliance with applicable design and safety criteria.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The amendment will not create the possibility of a new or different accident not previously analyzed, since the operating modes and plant configuration will not be changed from those previously analyzed in the Final Safety Analysis Report.

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

This amendment will not reduce the margin of safety since the plant operating and safety limits will remain unchanged. All cycle designs have been and will continue to be analyzed using NRC approved methods to demonstrate that existing design limits and safety analysis criteria are met in advance of cycle operation.

In addition, the NRC has provided examples of amendments that are considered not likely to involve significant hazards considerations (48 Fed. Reg. at 14870). This proposed amendment matches example (iii):

> "a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to acceptance criteria for the Technical Specifications, that the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that the NRC has previously found such methods acceptable."

This particular amendment for a less restrictive fuel rod uranium weight matches this example since Technical Specification 5.3.1 will continue to require reload fuel assemblies which are similar in physical design as that previously approved for St. Lucie Unit 1.

When compared to the standards set in 10 CFR 50.92(c), this proposed amendment does not involve a significant safety hazards consideration. This is further verified by comparing this change with the example given in the Federal Register, where in, this is a change that will result in the reactor core being reloaded with fuel assembly designs that have been analyzed with applicable NRC approved methodology to verify compliance with applicable design and safety criteria. Therefore, it is concluded that operation of St. Lucie Unit 1 in accordance with the proposed amendment will not pose a threat to the public health and safety.

Based on the above, we have determined that the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.