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Electric and Gas
Company

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United States Nuclear Regulatory Commission
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Gentlemen:

REQUEST FOR LICENSE AMENDMENT, REVISION 2
SALEM GENERATING STATION UNITS 1 AND 2
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

In accordance with the requirements of 10 CFR 50.90, Public Service Electric and Gas Company (PSE&G) hereby submits a request for amendment of Facility Operating Licenses DPR-70 (Unit 1) and DPR-75 (Unit 2) of the Salem Generating Station. Pursuant to the requirements of 10 CFR 50.91(b)(1), a copy of this submittal has been sent to the state of New Jersey as indicated below.

On November 19, 1990, PSE&G submitted a proposed amendment request which would facilitate the maximum allowable fuel assembly enrichment in the reactor, new fuel storage racks and spent fuel storage racks. On May 20, 1991, PSE&G submitted Revision 1 to the amendment request to specify the maximum enrichment of the fuel assemblies for the new and spent fuel storage racks.

Based on discussions with Mr. J. Stone, NRR Project Manager for Salem Generating Station, we are now transmitting Revision 2 of this amendment request. This revision specifies, within the revised Technical Specifications, the calculational uncertainties which have been included in the spent fuel storage rack analysis thereby eliminating the Technical Specification reference to the UFSAR.

This revision does not affect the justification for the proposed change or the Significant Hazards Consideration as stated in the previous submittal. This information, which includes a detailed description of the calculational uncertainties, is contained in attachment 1. Section I and II of this attachment have been revised to delete the administrative reference to the UFSAR. Attachment 2 provides a markup of the applicable Technical Specification pages to reflect the requested change.

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Upon NRC approval, please issue a License Amendment which will be effective upon issuance and shall be implemented within 60 days of issuance.

Should you have any questions on this submittal, please do not hesitate to contact us.

Sincerely,

Attachments (2)

C Mr. J. C. Stone
Licensing Project Manager - Salem

Mr. T. Johnson
Senior Resident Inspector

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Administrator - Region I

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REQUEST FOR AMENDMENT, REVISION 2
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FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
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NLR-N91087

ATTACHMENT 1

I. Description of Change

This amendment change submittal requests that: (1) Technical Specification sections 5.3.1, 5.6.1, and 5.6.3 be revised by removing the maximum enrichment limit and replacing it with the requirement that the reference Kinf (Infinite Multiplication Factor) for fuel assemblies be less than or equal to 1.453 in unborated water at 68 F° in core geometry; and (2) Technical Specification section 5.6.1 be revised to include an additional reactivity uncertainty for fuel assemblies containing Integral Fuel Burnable Absorber (IFBA) pins.

II. Reason for Proposed Change

This amendment is requested in support of Salem Unit 1, Cycle 11, and Salem Unit 2, Cycle 8. In these cycles, fuel enrichments may be increased to 4.55 w/o U-235 to allow design and operational flexibility associated with near term, projected, high energy cycle requirements. These license amendments will permit storage of Vantage 5H (V5H) fuel, containing Integral Fuel Burnable Absorbers (IFBAs), with a maximum initial enrichment of 4.55 w/o U-235. V5H fuel is currently loaded in Cycle 9 of Salem Unit 1 and Cycle 6 of Salem Unit 2, and will be utilized for future reloads.

The justification supporting this amendment request provides the bases for concluding that the proposed changes are consistent with the licensing bases of the spent fuel pool and verify that spent fuel pool criticality safety limits are not violated. The evaluation demonstrates that an increase in maximum initial assembly enrichment up to 4.55 w/o U-235, with sufficient IFBAs, will not cause a significant reduction in the margin of safety. Since the criticality safety analysis confirm the original criteria and the V5H assembly structural integrity is equivalent to that of the standard assembly, the possibility of a new or different kind of accident or condition outside of previous evaluations is not credible.

No additional analyses were required to be performed for the new fuel storage racks. The previous analyses (Reference 3) supporting the license change which increased enrichment to 4.05 w/o U-235 in the spent fuel pool and new fuel storage racks actually accounted for enrichments of up to 4.50 w/o U-235 in the new fuel storage racks.

This 4.50 w/o U-235 fuel did not include IFBA pins. Since the current proposed license change will assure that fuel assemblies are less reactive than 4.05 w/o U-235 with no IFBA, the previous new fuel storage rack analyses remain bounding.

III. Justification for the Proposed Change

The Westinghouse Vantage 5H Fuel (V5H) option was introduced as a mix with the standard fuel in Salem Unit 1, Cycle 9 and Unit 2, Cycle 6. PSE&G plans to continue the use of V5H fuel for all future operating cycles. In order to achieve PSE&G's economic goals, fuel strategies, being evaluated for Unit 1 , Cycle 11, may include the use of enrichments of 4.50 w/o U-235, which exceeds the current enrichment limit of 4.05 w/o U-235 for storage in the spent fuel pool. Therefore, supplemental criticality analyses have been performed to support the storage of 4.55 w/o U-235 fuel. This enrichment value includes 0.05 w/o U-235 allowance for manufacturing tolerances on enrichment.

The analyses and evaluations performed to support the storage of higher enriched fuel have concluded that spent fuel criticality limits are maintained when storing fuel with a maximum enrichment of 4.55 w/o U-235 provided, that fuel with enrichments greater than 4.05 w/o U-235 have sufficient Integral Fuel Burnable Absorbers (IFBA) to maintain an unborated reference fuel assembly Kinf less than or equal to 1.453 at 68 F° in core geometry.

Figure 1 shows the plot of enrichment versus number of IFBA pins per fuel assembly above which the unborated fuel assembly Kinf is less than 1.453 at 68°F. The Kinf remains less than 1.453 provided at least the number of IFBA pins shown at each enrichment point on Figure 1 are loaded in symmetric patterns within the fuel assembly at the specified initial enrichment.

Description of the Salem Spent Fuel Pool

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (Keff) of the fuel assembly array will be less than 0.95.

Criticality of the fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies.

The fuel assemblies, assumed for this analysis, contain the highest enrichment allowed without any control rods or any removable burnable poison and are at their most reactive point in life. The assemblies are conservatively modeled with water replacing the assembly grid volume. The storage cell nominal geometry is shown in Figure 2. The biases and uncertainties determined in the original storage rack analyses were based on the following assumptions:

1) The moderator is pure water at the temperature within design limits of the pool which yields the largest reactivity. No dissolved boron is included in the water.

2) The nominal case calculation treats the storage rack as an infinite lattice.

3) Credit is taken for the neutron absorption in full length structural materials and in solid materials added specifically for neutron absorption.

4) A bias is included in the reactivity calculation to account for Boral self shielding.

5) A bias is included to account for random variations in the minimum gap between storage cells.

Description of the Salem Fuel Designs

Both the Standard and Vantage 5H fuel types are Westinghouse 17x17, twelve foot long assemblies which utilize a 0.374 inch OD fuel rod and zircaloy cladding. The primary mechanical difference between the fuel types is the V5H design employs zircaloy, rather than inconel, spacer grids. Either assembly design can incorporate IFBA pins which consist of a thin zirconium diboride (ZrB_2) coating on the outside of the fuel pellet. As a result, the IFBA is a non-removable and hence an integral part of the fuel assembly once it is manufactured. For this analysis, a nominal poison loading of 0.00157 grams B10 per

inch of IFBA pin was used. Within the assembly, the IFBA pin distributions were arranged in symmetric patterns, which followed standard Westinghouse loadings of up to 128 IFBA pins. It was determined that the results of the analysis are valid for any arrangement of IFBA and non-IFBA fuel pins provided that there is a symmetric distribution within the assembly geometry.

Criticality Analysis

Extensive analyses have previously been performed to support the storage of standard fuel assemblies with a maximum initial enrichment of up to 4.05 w/o U-235 under both normal and postulated accident conditions. In order to increase the maximum initial enrichment for the Salem spent fuel pool storage, the concept of reactivity equivalencing has been used. This concept is predicated upon the reactivity decrease associated with the addition of IFBA fuel rods offsetting the reactivity increase associated with the increase in initial enrichment. A series of reactivity calculations have been performed to generate a set of ordered pairs, consisting of IFBA rod number per assembly versus initial enrichment, which yield a Keff equivalent to the 4.05 w/o U-235 standard fuel Keff when the V5H IFBA fuel is stored in the spent fuel racks.

The data points in the core geometry for the reactivity equivalence curve were calculated with a transport theory computer code, PSCPM. PSCPM is a depletable, two-dimensional, multigroup, transport theory code. The PSCPM code is based on the EPRI code CPM-2 which has been benchmarked against isotopic experiments and reactor critical state point data during the ARMP package development (Reference 1). CPM-2 has also been compared to other industry standard cross-section generation codes such as EPRI-CELL and CASMO (Reference 2). For the purposes of reactivity comparisons, all these codes give essentially the same results. PSCPM is a version of CPM-2 modified by PSE&G to permit IFBA fuel assembly calculations. Since PSCPM is based on CPM-2, it will reproduce the CPM-2 results from the EPRI study, and therefore is as reliable as the other industry standard codes. In addition, the PSCPM code has been validated by comparisons to Westinghouse IFBA and enrichment calculated data.

A second code, PDQ7/HARMONY (hereafter referred to as PDQ), was used to model the spent fuel pool geometry. PDQ is a depletable, few-group, diffusion theory code. In this analysis, PDQ uses cross-sections generated by PSCPM as input. Each PDQ case was checked by comparing the Kinf edited over the assembly region only to the Kinf from the identical PSCPM. In addition, PDQ results were checked against PSCPM results in the spent fuel pool geometry with no Boral for the current 4.05 w/o U-235 enrichment limit.

The criticality analysis performed to support this amendment request submittal consisted of five parts. In the first part, PSCPM and PDQ cases were executed for the maximum enrichment of 4.05 w/o U-235 with no IFBA rods, in both core geometry and in spent fuel pool geometry, unborated at 68 F°. PSCPM was used to generate nuclear data for the fuel assembly in core geometry. This data was then transferred to PDQ for use in the SFP geometry cases. PDQ was used in the spent fuel pool geometry to establish a common Keff calculation for the current maximum enrichment. PSCPM was used in the spent fuel pool geometry to determine water gap cross section data.

During this first part, the Boral absorption cross-sections in the PDQ were adjusted such that the nominal Keff duplicated the 0.92 in the spent fuel pool geometry, as determined in the original rack analysis.

The second part of the analysis consisted of a series of core geometry, hot full power, 600 ppm soluble boron depletions using the PSCPM code. These depletions were for enrichment/IFBA combinations. The enrichment ranged from 4.05 w/o to 4.55 w/o U-235. The number of IFBA rods ranged from 0 to 128 rods configured in standard Westinghouse IFBA arrangements. These depletion cases were then restarted at 68 F° and 0 ppm boron. The core geometry Kinf values calculated from these restarts are used in the reactivity equivalencing procedure.

Figure 1 shows the constant Kinf line corresponding to the number of IFBA rods as a function of enrichment determined from this reactivity equivalencing procedure. The endpoints in the figure are 4.05 w/o U-235 with 0 IFBA and 4.55 w/o with 60 IFBA. The interpretation of the data is that the reactivity of a 4.55 w/o fuel assembly with 60 IFBA rods is equivalent to the reactivity of a 4.05 w/o assembly with 0 IFBA rods. This figure will be incorporated into section 9.1.2.1 of the Salem UFSAR.

The third part of the analysis consisted of a series of PDQ cases with spent fuel pool geometry using the cross-sections determined by the PSCPM restarts described above. These PDQ cases verified that the relative reactivities of the various enrichment/IFBA combinations did not change between the core or spent fuel pool geometries. Note that the original analyses, supporting 4.05 w/o U-235 fuel storage (Reference 3), assumed no grids in the fuel (replaced grid volume with water volume). Therefore, storage of the V5H and standard fuel assemblies will result in equivalent Keffs.

The forth part of the analysis was a study of the axial cutback of the boron coating on the IFBA rods from the ends of the active fuel. This study was performed using one-dimensional PDQ cases.

These cases determined additional reactivity to be added to the nominal Keff of the various enrichment/IFBA combinations starting with 4.55 w/o U-235 and full length IFBA fuel. A cutback of up to 18 inches on each end of the IFBA rods resulted in Keff values less than 0.92 in the spent fuel pool geometry. The plot in Figure 1 which defines the minimum number of IFBA pins for enrichments greater than 4.05 w/o U-235 includes the effect of an 18 inch IFBA cutback on each end of the fuel rod.

The fifth part of the analysis examined the Kinf of the enrichment/IFBA combinations as a function of assembly burnup. The maximum reactivity occurs at zero burnup for all cases at the constant Kinf line. For cases off the line with more than 60 IFBA rods, the maximum reactivity may occur at some higher burnup. However, in those cases, the maximum Kinf remains less than the reactivity of the constant Kinf line for U-235 enrichments less than 4.55 w/o.

To simplify verification of acceptability for storage of fuel in the spent fuel pool, a Kinf for a fresh 4.05 w/o U-235 fuel assembly was determined using PSCPM. The Kinf is used as a reference criticality point which eliminates the need to specify an acceptance enrichment versus number of IFBA rods correlation. Calculation of the Kinf resulted in a reference value of 1.453 at 68 F° in the reactor core geometry. This value, 1.453, is consistent for the varying enrichment/IFBA combinations including biases added for IFBA loading tolerance, IFBA loading pattern variation, IFBA modeling uncertainties and original analysis uncertainties.

Calculational Uncertainty Allowances

The combination of biases due to mechanical tolerances in rack geometry (water gap between poison plates), benchmark comparisons, poison particle self-shielding, and uncertainties due to methodology result in a total uncertainty allowance of 2.2% delta k/k.

The addition of IFBA rods in an assembly for enrichments greater than 4.05 w/o U-235 result in the following additional uncertainties:

- The IFBA model uncertainty is evaluated to be 0.2% delta k/k for a fuel assembly containing 60 IFBA rods. This value is consistent with PSE&G studies and the result of comparing PSCPM Kinf values against Westinghouse generated results for the range of enrichment/IFBA combinations depicted in Figure 1.

- The IFBA loading tolerance was taken to be 5%, such that in the criticality analysis the IFBA loading was assumed to be 0.95 times the nominal value of 0.00157 grams of B10 per inch. This results in a 0.3% reactivity uncertainty.

- Potential variation in the symmetric IFBA loading pattern within the assembly results in a maximum 0.2% reactivity uncertainty.

With linear addition of these uncertainties, the resultant 2.9% delta k/k total uncertainty has been included in the spent fuel pool rack Keff determination.

References

1) Advanced Recycle Methodology Program (ARMP) documentation. EPRI CCM-3

2) "Evaluation of Discrepancies in Assembly Cross-Section Generator Codes", EPRI NP-6147

3) License change request submitted by PSE&G on September 2, 1983. Subsequently issued as Amendment No. 55 to Facility Operating License No. DPR-70 (Unit 1) and Amendment No. 22 to Facility Operating License No. DPR-75 (Unit 2) dated November 22, 1983.

IV. Significant Hazards Consideration

In accordance with 10CFR50.92, PSE&G has reviewed the proposed changes and concluded that they do not involve a significant hazards consideration. Extensive analyses were previously performed to support storage of V5H fuel up to a maximum enrichment of 4.05 w/o U-235 (refer to Reference 3 above). Increasing the maximum assembly enrichment limit to 4.55 w/o U-235, including IFBA rods, for allowed storage in the spent fuel pool does not compromise the three criteria of 10CFR50.92(c). The proposed changes do not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed. Because of the conservative methods and assumptions used to evaluate the maximum possible assembly multiplication factor, there is more than reasonable assurance that no significant hazard based on criticality safety is involved in storing fuel assemblies with enrichments of up to 4.55 w/o U-235, with sufficient IFBAs, in the spent fuel storage racks under both normal and postulated accident conditions. The calculations used to determine the minimum number of IFBA rods required as a function of enrichment assured an assembly Kinf less than or equal to that of a fresh 4.05 w/o U-235 assembly with no IFBA under 0 ppm soluble boron conditions. The criticality accidents for 4.05 w/0 U-235 fuel have been analyzed previously and there will be no increase in assembly Kinf.

   Additionally, evaluations of reload core designs (using any enrichment) will be performed on a cycle by cycle basis as part of the Reload Safety Evaluation (RSE) process to ensure that the reactor operation is consistent with the current safety analysis. Therefore, there is no increase in the probability or consequences of any accident previously analyzed.

2. Create the possibility of a new or different kind of accident. The increase in enrichment to 4.55 w/o U-235 involved the performance of evaluations to envelope the corresponding changes in reactivity. Use of the reactivity equivalencing procedures ensures that the spent fuel pool criticality limits are not exceeded. Additionally, there are no proposed changes to the spent fuel rack geometry.

3. Involve a significant reduction in a margin of safety. As discussed above, for worst case assumptions the assembly Kinf values for a maximum enrichment of 4.55 w/o U-235, with a sufficient number of IFBA rods do not exceed those for the previously analyzed 4.05 w/o U-235. Therefore there are no reductions in any margin of safety.

V. Conclusions

Based on the above discussions and those presented in the Justification Section, it has been determined that the proposed Technical Specification revisions do not involve a significant increase in the probability or consequences an accident over previous evaluations, create the possibility of a new or different kind of accident, or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

Reanalysis of the spent fuel pool heat loads and radiological consequences of potential fuel handling accidents were not required to be performed as part of this submittal. These are issues which are affected by extended fuel burnups and operational history. Although it is possible to achieve extended cycle burnups using assemblies with increased enrichment, actual assembly burnups will depend on core reload designs and integrated power history.

Therefore, the impact of extended burnups will be addressed as a separate issue, and any required safety analysis and FSAR updates will be performed as necessary.

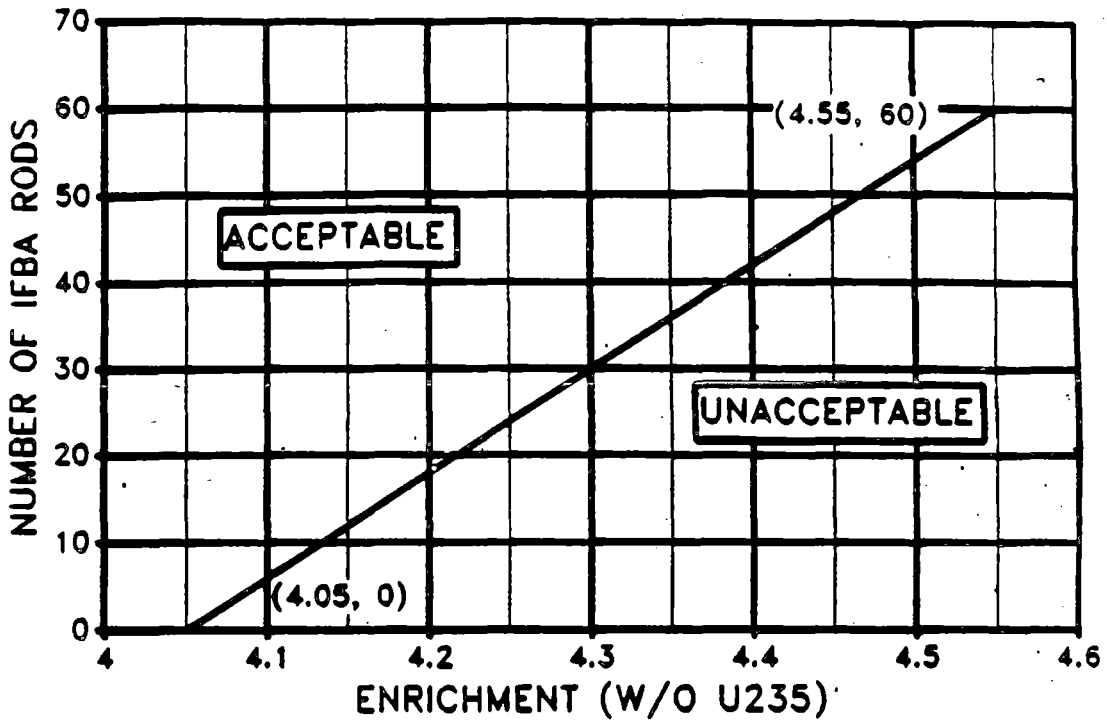

Figure 1

MINIMUM NUMBER OF IFBA RODS PER ASSEMBLY VERSUS ENRICHMENT

Figure 2

SALEM SPENT FUEL STORAGE CELL NOMINAL DIMENSIONS