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SUBJECT: Application for amend to license NPF-22,requesting to use 24-month operating cycle & advance design SPC ATRIUM-10 fuel.

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SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 166 TO LICENSE NPF-22:
UNIT 2 TECHNICAL SPECIFICATION CHANGES FOR
ATRIUM-10 FUEL
PLA-4527 FILES R41-2/A17-2 Docket No. 50-388

The purpose of this letter is to propose changes to the Susquehanna Steam Electric Station Unit 2 Technical Specifications. These changes stem from the scheduled use of Siemens Power Corporation (SPC) ATRIUM-10 fuel. The ATRIUM-10 fuel design is a 10x10 lattice design which has been analyzed according to SPC's NRC approved methodology and meets the applicable safety criteria. The proposed change entails changes to the Definitions (Section 1.0), the MCPR Safety Limit values (Sections 2.1 and 3/4.4), the Design Features (Section 5.3), and the Administrative Controls (addition of methodology references to Section 6.9.3.2). Associated BASES changes are also included.

Enclosure A to this letter is the "Safety Assessment" supporting this change. Enclosure B to this letter is the "No Significant Hazards Considerations" evaluation performed in accordance with the criteria of 10 CFR 50.92. The proposed changes have been approved by the Susquehanna SES Plant Operations Review Committee and reviewed by the Susquehanna Review Committee.

Enclosure C to this letter is the current pages of the Susquehanna SES Unit 2 Technical Specifications marked to show the proposed changes.

PP&L plans to implement the proposed changes in March 1997 to support Cycle 9 operation. Therefore, we request NRC complete the review of this change request by March 31, 1997 to support our scheduled implementation dates.

A001

Any questions regarding this request should be directed to Mr. A. K. Maron at (610) 774-7727.

Very truly yours,

R. G. Byram

Enclosures

copy: NRC Region I
Mr. C. Poslusny, Jr., NRC Sr. Project Manager - OWFN
Mr. K. M. Jenison, NRC Sr. Resident Inspector - SSES
Mr. W. P. Dornsife, Pa. DEP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PENNSYLVANIA POWER & :
LIGHT COMPANY

Docket No. 50-388

PROPOSED AMENDMENT NO. 166
FACILITY OPERATING LICENSE NO. NPF-22
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 2

Licensee, Pennsylvania Power & Light Company, hereby files proposed Amendment No. 166 to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment contains a revision to the Susquehanna SES Unit 2 Technical Specifications.

PENNSYLVANIA POWER & LIGHT COMPANY
BY:

R. G. Byram
Sr. Vice President - Nuclear

Sworn to and subscribed before me
this 18th of December 1996.

Notary Public

Notarial Seal
Martha C. Sedora, Notary Public
Allentown, Lehigh County
My Commission Expires Jan. 15, 1998
Member, Pennsylvania Association of Notaries

SAFETY ASSESSMENT

Unit 2 Technical Specification Changes for ATRIUM-10 Fuel

BACKGROUND

Susquehanna Steam Electric Station Unit 2, Cycle 9 will utilize a 24 month operating cycle and the advanced design SPC ATRIUM-10 fuel. The ATRIUM-10 fuel design is a 10x10 lattice design which contains 83 full length fuel rods, 8 part length fuel rods, and a central water channel to enhance neutron moderation. The mechanical design has been analyzed according to SPC's NRC approved generic mechanical design criteria (Reference 2). PP&L has reviewed the SPC mechanical design calculations (performed according to SPC's QA program), and the results demonstrate that ATRIUM-10 complies with NRC approved criteria.

The ATRIUM-10 design and analyses using the NRC approved codes and methodologies added to the Technical Specifications will be used to support Unit 2 cycles starting with cycle 9. This proposed change to the Susquehanna SES Unit 2 Technical Specifications supports the use of ATRIUM-10 fuel.

Description of the Proposed Change

The Unit 2 Technical Specification changes consist of:

(1) changes to two definitions in Section 1.0 to make them applicable to ATRIUM-10 fuel (i.e., to reflect the ATRIUM-10 design's part length fuel rods).
(2) inclusion of the U2C9 MCPR Safety Limits in Sections 2.1.2 and 3.4.1.1.2,
(3) changes to Section 5.3.1 to reflect the ATRIUM-10 design, and
(4) inclusion of Siemens Power Corporation (SPC) methodology topical reports (References 2 to 17) in Section 6.9.3.2,

Changes to the BASES sections to reflect the ATRIUM-10 design and methodology are also included. A summary of the Technical Specifications changes is provided below.

Definitions (Section 1.0)

The ATRIUM-10 fuel design is a 10x10 lattice design which contains 83 full length fuel rods and 8 part length fuel rods. In Definitions 1.2 and 1.3, the definitions for AVERAGE BUNDLE EXPOSURE, AVERAGE PLANAR EXPOSURE, and AVERAGE PLANAR LINEAR HEAT GENERATION RATE are changed to apply to fuel assemblies containing part length rods, as well as assemblies containing only full length rods (e.g., 9x9-2).

MCPR Safety Limits (Sections 2.1.2 and 3.4.1.1.2)

Excessive overheating of the fuel rod cladding can result in cladding damage and the release of fission products. In order to protect the cladding against overheating due to boiling transition, the THERMAL POWER, High Pressure and High Flow SAFETY LIMITs (Sections 2.1.2 and 3.4.1.1.2 of the Susquehanna SES Unit 2 Technical Specifications) were established.

NUREG-0800, Standard Review Plan Section 4.4, specifies an acceptable, conservative approach to define this SAFETY LIMIT. Specifically, a Minimum Critical Power Ratio (MCPR) value is specified such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. Boiling transition is predicted using a correlation based on test data (i.e., a Critical Power Correlation). Currently, the ANFB Critical Power Correlation is used.

The specific SAFETY LIMIT MCPR values (two-loop and single-loop) are being revised using NRC approved licensing methods. The SAFETY LIMIT MCPR calculation combines various uncertainties such as feedwater flow, feedwater temperature, pressure, power distribution uncertainties, and uncertainty in the Critical Power Correlation.

Design Features (Section 5.3.1)

Section 5.3.1 is revised to reflect the fact that ATRIUM-10 contains a central water channel. Reference to a 150 inch active fuel length is removed. Also, the maximum enrichment is increased from 4.0 to 4.5 weight percent U_{235}. Criticality analyses were performed to assure that the reactivity requirements of Technical Specification 5.6 are met.

Addition of Siemens Methodology References (Section 6.9.3.2)

Included in the revised Technical Specifications via reference (Section 6.9.3.2) are additional NRC approved methodology reports. The NRC approved topical reports added (References 2 to 17) contain methodology used to assure safe operation of Unit 2 with ATRIUM-10 fuel.

BASES Changes

BASES Section 2.1.1 (THERMAL POWER, Low Pressure or Low Flow) was revised to be applicable for both 9x9-2 and ATRIUM-10 fuel. Specifically, the amount of flow in an ATRIUM-10 assembly was calculated and specified.

BASES Section 2.1.2 was changed to remove a reference to the XN-3 correlation which is no longer used.

SAFETY ANALYSIS

This section discusses the safety implications of the proposed action.

Definitions (Section 1.0)

The change to the definitions for AVERAGE BUNDLE EXPOSURE, AVERAGE PLANAR EXPOSURE, and AVERAGE PLANAR LINEAR HEAT GENERATION RATE allow them to be applicable to all types of fuel assemblies. There are no safety implications of this change.

MCPR Safety Limits (Sections 2.1.2 and 3.4.1.1.2)

General Design Criterion 10 requires that the specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). The fuel cladding integrity Safety Limit is set such that no significant fuel damage from cladding overheating is calculated to occur if the limit is not violated. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The MCPR Safety Limit helps ensure sufficient conservatism in the operating MCPR limit such that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR=1.0) and the MCPR Safety Limit is based on a statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the Safety Limit is the uncertainty inherent in the critical power correlation.

The critical power correlation is based on a significant body of practical test data, providing a degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation, the assumed reactor conditions used in defining the Safety Limit introduce conservatism into the limit because bounding, high radial power factors and bounding, flat local peaking distributions are used to estimate the number of rods in boiling transition.

A cycle specific MCPR Safety Limit analysis was performed for PP&L by SPC. This analysis used NRC approved methods described in Technical Specification Reference 13 (ANF-524(P)(A), Revision 2 and Supplement 1 Revision 2). The SAFETY LIMIT MCPR calculation statistically combines uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and the uncertainty in the Critical Power Correlation. The SPC analysis used cycle specific power distributions and calculated a value of MCPR such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or

anticipated operational occurrences. The resulting two-loop and single-loop values (Technical Specification Sections 2.1.2 and 3.4.1.1.2) are included in the proposed change.

Design Features (Section 5.3.1)

The description of a fuel assembly (Section 5.3.1) is revised to reflect the fact that ATRIUM-10 contains a central water channel. Since the active fuel length of ATRIUM-10 is different from that of 9x9-2, reference to an active fuel length of 150 inches was deleted.

In addition, the maximum allowed enrichment is increased from 4.0 to 4.5 weight percent U_{235}. Criticality calculations were performed to ensure that ATRIUM-10 fuel with a lattice average enrichment of 4.5 weight percent U_{235} can be safely stored in both the new fuel vault and the spent fuel storage pool at Susquehanna. These SPC analyses used the KENO Monte Carlo code which is part of the SCALE 4.2 Modular Code System. These calculations demonstrate that the maximum k-effective of both the new fuel vault and spent fuel storage pool will not exceed 0.95 under the worst credible storage array conditions or under accident conditions. The calculations included allowances for statistical uncertainty associated with the analytical method, computer code benchmark calculations, and both fuel and rack manufacturing tolerances. The analyses demonstrate that maximum fuel lattice average enrichments up to and including 4.50 weight percent U_{235} can be allowed.

Addition of Siemens Methodology References (Section 6.9.3.2)

Included in the revised Technical Specifications via reference (Section 6.9.3.2) are additional NRC approved methodology reports. The NRC approved topical reports added (References 2 to 17) contain methodology which is used to assure safe operation of Unit 2 with ATRIUM-10 fuel. PP&L will continue to use their NRC approved reload analysis methods (Reference 1) to analyze Unit 2 cores containing ATRIUM-10 fuel. The subsections below describe how the methodologies in the references to be added to Section 6.9.3.2 will be used for SSES:

Mechanical Design

The ATRIUM-10 mechanical design has been analyzed according to SPC's NRC approved generic mechanical design criteria (Reference 2). PP&L has reviewed the SPC mechanical design calculations (performed according to SPC's QA program), and the results demonstrate that ATRIUM-10 complies with the NRC approved criteria. References 4 through 7 describe mechanical design methodologies supporting the Reference 2 methodology.

A requirement of SPC's generic mechanical design criteria (Reference 2) is that fuel designs have stability characteristics that are equivalent to or better than a previously approved SPC design. ATRIUM-10 has been shown to have roughly equivalent stability characteristics to 9x9-2 and ATRIUM-9 fuel.

Anticipated Operational Occurrences and ASME Overpressure

PP&L's NRC approved methodology (Reference 1) specifies that AOOs and the ASME overpressure analyses are performed assuming two-loop operation. To assure continued conformance to PP&L's Current Licensing Basis, SPC is performing analyses of AOO and ASME overpressure events in two-loop and single-loop operation (both full cores and mixed cores containing ATRIUM-10 fuel). These analyses using their NRC approved methodology (References 3 and 12 through 17) will be used to demonstrate that the results of these events in two-loop operation bound the results of these events in single-loop operation.

Loss of Coolant Accident

ATRIUM-10 LOCA analyses are being performed using SPC's NRC approved LOCA methodology (References 8 through 11). The MAPLHGR and LHGR restrictions derived from these analyses will be applied to the ATRIUM-10 assemblies in Unit 2.

Single Loop Operation (SLO) Pump Seizure

Reference 20 documents NRC approval of a generic pump seizure analysis applicable to 9x9-2 assemblies at SSES. To address the SLO pump seizure event for ATRIUM-10 cores, the Reference 20 approach will be used, and the analysis will be performed using the current SPC NRC approved codes. Specifically, the XTG, XFYRE, and COTRANSA codes and the XN-3 Critical Power correlation used in the Reference 20 approach are replaced with the MICROBURN, CASMO, and COTRANSA2 (Reference 15) codes and the ANFB correlation. The number of fuel rods in boiling transition is calculated using SPC's MCPR Safety Limit methodology. Bounding SPC analyses of both full cores and mixed cores containing ATRIUM-10 fuel are being performed to demonstrate that the radiological consequences of a pump seizure event will not exceed a small fraction (i.e., 10%) of 10CFR100 guidelines.

Fuel and Equipment Handling Accidents

Reference 19 documents NRC approval of the analytical methods used for fuel and equipment handling accidents for SPC 9x9-2 fuel at SSES. The methodology conforms to the requirements of Regulatory Guide 1.25. Using the same methodology, SPC performed fuel and equipment handling accident calculations involving ATRIUM-10 fuel. The conservatively calculated doses were "well within" 10CFR100 guidelines (i.e., 25% of 10CFR100 doses).

Control Rod Drop

SPC has confirmed that their parametric control rod drop accident also applies to ATRIUM-10 fuel. PP&L will continue to calculate input to the parametric analysis using its NRC approved methodology.

In summary, the analytical approaches to be used for Unit 2 to analyze ATRIUM-10 fuel are consistent with previously approved approaches and utilize NRC approved codes and methods.

BASES Changes

The BASES for Section 2.1.1 (THERMAL POWER, Low Pressure or Low Flow) was revised to be applicable for both 9x9-2 and ATRIUM-10 fuel. SPC performed evaluations to calculate the mass flux in an ATRIUM-10 bundle when the downcomer level is above the top of active fuel and reconfirmed that the critical power is above 3.35 MW. Thus, the THERMAL POWER, Low Power or Low Flow Safety Limit is valid for both 9x9-2 and ATRIUM-10.

CONCLUSIONS

The proposed change to the Susquehanna SES Unit 2 Technical Specifications supports the use of ATRIUM-10 fuel. NRC approved methods are used to compute the MCPR Safety Limits and Core Operating Limits. The analytical approaches to be used for Unit 2 to analyze ATRIUM-10 fuel are consistent with the approaches previously approved by the NRC and will utilize NRC approved codes.

REFERENCES

1. PL-NF-90-001, Supplement 2-A, "Application of Reactor Analysis Methods to BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation," July 1996.

2. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

3. XN-NF-81-58 (P)(A) Supplements 1 and 2 Revision 2, "RODEX 2 Fuel Rod Thermal-Mechanical Response Evaluation Model," May 1986.

4. XN-NF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Response Evaluation Model," August 1986.

5. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5 Revision 1, "Qualification of Exxon Nuclear Fuel for Extended Burnup," October 1986.

6. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity," November 1986.

7. ANF-90-082(P)(A) Revision 1 and Revision 1 Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," May 1995.

8. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

9. ANF-CC-33(P)(A) Supplement 2, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.

10. ANF-CC-33(P)(A) Supplement 1 Revision 1, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," August 1986.

11. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

12. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

13. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

14. ANF-1358(P)(A), Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," September 1992.

15. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," August 1990.

16. XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T : A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.

17. XN-NF-84-105(P)(A), Volume 1 Supplement 4, "XCOBRA-T : A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data," June 1988.

18. "A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," SCALE 4.2, Oak Ridge National Laboratory, revised December 1993.

19. Letter from James J. Raleigh (NRC) to H. W. Keiser (PP&L), "Cycle 7 Reload Amendment, Susquehanna Steam Electric Station, Unit 1(TAC No. M82356)", May 7, 1992.

20. Letter from Mohan C. Thadani (NRC) to H. W. Keiser (PP&L), "Cycle 6 Reload, Susquehanna Steam Electric Station, Unit 1(TAC No. 77165)", November 2, 1990.

NO SIGNIFICANT HAZARDS CONSIDERATIONS

Unit 2 Technical Specification Changes for ATRIUM-10 Fuel

Pennsylvania Power & Light Company has evaluated the proposed Technical Specification change in accordance with the criteria specified by 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration. The criteria and conclusions of our evaluation are presented below.

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The applicable sections of the FSAR are Chapters 5, 6.3, 9, and 15 of the FSAR. Chapter 5 discusses the results of the ASME overpressure analysis for the reactor pressure boundary. Chapter 6.3 discusses the LOCA. Chapter 9 discusses fuel storage and handling. Chapter 15 describes the transient and accident analyses, a majority of which have been generically dispositioned to be non-limiting. A discussion of the impact of the Technical Specification changes is provided below.

   The change to Definitions 1.2 and 1.3 makes the definitions applicable to ATRIUM-10. There are no effects on safety functions from this change.

   A cycle specific MCPR Safety Limit analysis was performed for PP&L by SPC. This analysis used NRC approved methods described in Technical Specification Reference 13 (ANF-524(P)(A), Revision 2 and Supplement 1 Revision 2). The SAFETY LIMIT MCPR calculation statistically combines uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and the uncertainty in the Critical Power Correlation. The SPC analysis used cycle specific power distributions and calculated MCPR values such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The resulting two-loop and single-loop values (Technical Specification Sections 2.1.2 and 3.4.1.1.2) are included in the proposed change. Thus, the cladding integrity and its ability to contain fission products is not adversely affected.

   The change to the Design Features (Section 5.3) increases the allowable enrichment. Analyses have demonstrated that the ATRIUM-10 fuel will remain subcritical (k-effective < 0.95) in both the spent fuel pool and the new fuel vault. Thus, the change to allowable enrichment has no impact on safety functions. The description of a fuel assembly (Section 5.3) is also revised to reflect the ATRIUM-10 central water channel, and reference to an active fuel length of 150 inches was deleted. This change reflects the physical characteristics of the ATRIUM-10 fuel and has no impact on the probability or consequences of an event.

Included in the revised Technical Specifications via reference (Section 6.9.3.2) are additional NRC approved methodology reports. The NRC approved topical reports contain methodology which is used to assure safe operation of Unit 2 with ATRIUM-10 fuel. These methodologies assure that the core meets appropriate margins of safety for all expected plant operational conditions ranging from refueling and cold shutdown of the reactor through power operation. Thus, the results obtained from the analyses will provide assurance that the reactor will perform its design safety function during normal operation and design basis events.

The BASES changes for Section 2.1.1 (THERMAL POWER, Low Pressure or Low Flow) reflect that the Safety Limit is valid for both 9x9-2 and ATRIUM-10.

Therefore, the proposed action does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The changes to the Unit 2 Technical Specifications (Definitions, MCPR safety limits, Design Features, and inclusion of methodology references) to allow use of ATRIUM-10 fuel do not require any physical plant modifications, physically affect any plant components, or entail significant changes in plant operation. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure. The referenced methodology added to Section 6.9.3.2 contains NRC approved acceptance criteria. The consequences of transients and accidents will remain within the criteria approved by the NRC. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

   The applicable Technical Specification Sections include 1.0, 2.0, 3/4.4, 5.3, and 6.9.3.2.

   The changes to the Unit 2 Technical Specifications discussed in Item 1 above (Definitions, MCPR Safety Limits, Design Features, and inclusion of methodology references) to allow use of ATRIUM-10 fuel do not require any physical plant modifications, physically affect any plant components, or entail significant changes in plant operation. Therefore, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The NRC approved methods detailed in the references added to Section 6.9.3.2 maintain an equivalent margin of safety as currently defined in the bases of the applicable Technical Specification sections.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL CONSEQUENCES

An environmental assessment is not required for the proposed change because the requested change conforms to the criteria for actions eligible for categorical exclusion as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. The proposed change does not involve a significant hazards consideration as discussed above. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in the individual or cumulative occupational radiation exposure.