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SUBJECT: Submits methodology used to perform ATWS analysis on Unit 2
Cycle 9.

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SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 9 ATWS EVALUATION
PLA-4480 FILE R41-2 **Docket No. 50-388**

References:

1) GENE-637-024-0893, "Evaluation of Susquehanna ATWS Performance for Power Uprate Conditions," September 1993.

2) PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," March 24, 1992.

3) Advisory Committee on Reactor Safeguards Thermal Hydraulic Phenomena Subcommittee Meeting, Tuesday, October 31, 1995, Rockville, Maryland.

4) NRC Safety Evaluation: Modifications to the Boiling Water Reactor (BWR) Emergency Procedure Guidelines to Address Reactor Core Instabilities, June 6, 1996.

5) NRC Safety Evaluation: "Licensing Topical Report for Power Uprate with Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788)(TAC Nos. M83426 and M83427)," November 30, 1993.

The purpose of this letter is to inform the NRC that PP&L intends to perform the ATWS analysis for Susquehanna SES Unit 2 Cycle 9 (U2C9) using in-house methods. This letter discusses the methodology we plan to utilize.

Background

The latest ATWS analysis for Susquehanna was performed by GE for power uprate conditions (Ref. 1). This analysis also accounted for the 8x8 to 9x9 fuel change. In the U2C9 refueling (Spring of 1997) outage, 10x10 fuel will be loaded in the core. Because of the change in fuel type, it is necessary to re-evaluate the performance of Susquehanna under ATWS conditions.

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Methodology

The scope of the ATWS analysis is defined in our current evaluation for Susquehanna (Ref. 1). This analysis considers the impact of the following seven initiating events with scram failure:

- Main Steam Line Isolation Valve (MSIV) Closure,
- Pressure Regulator Failure - Open (PREGO),
- Loss of Normal Feedwater Flow (LOFW),
- Inadvertent Opening of a Safety/Relief Valve (IORV),
- Feedwater Controller Failure - Open (FWCFO),
- Turbine Trip (TT), and
- Loss of Normal AC Power (LACP).

For these ATWS events, the reactor and containment response is evaluated against the following criteria:

- **Reactor Pressure Vessel (RPV) Integrity**
  The peak RPV pressure must be less than 1500 psig (Service Level C).

- **Fuel Integrity**
  The maximum fuel cladding temperature cannot exceed 2200 °F, and the local cladding oxidation must be less than 17%.

- **Containment Integrity**
  The peak pressure must remain below the design pressure of 53 psig, while the peak suppression pool bulk temperature must remain less than 190 °F.

The analysis in Ref. 1 shows that there are two limiting events for peak values of vessel pressure, clad temperature, and suppression pool temperature. These two events are MSIV closure and Pressure Regulator Failure - maximum demand (PREGO). The results of Reference 1 indicate that peak vessel pressure and peak clad temperature are higher in events where the pressurization transient is caused by an MSIV closure as opposed to a turbine trip. The most severe ATWS events with regard to suppression pool heating are those which involve closure of the MSIVs and which also have feedwater injection available for a significant time following MSIV closure. Therefore, in the U2C9 ATWS evaluation for Susquehanna, only these two events will be considered.

Since ATWS is a beyond-design-basis event, we plan to use nominal set points for equipment actuation. In certain cases, however, Upper Analytical Limits (UALs) are used (e.g., S/RV and RPT set points) and an additional equipment failure (failure of one S/RV to open) is considered in order to add conservatism and to provide agreement with the GE methodology used in the Susquehanna power uprate ATWS study (Refs. 1,5).

Peak vessel pressure for the MSIV-closure and PREGO events will be calculated using the RETRAN02 MOD 5.1 computer code to determine if it remains below the allowable limit of 1500 psig. Each scenario will be run with the U2C9 core which is partially loaded with ATRIUM-10 fuel. These ATWS analyses will be carried out with the same model which is used to perform the nuclear fuel reload analysis that has been accepted by the NRC (Ref. 2).

The increase in the ratio of clad surface area to fuel volume associated with the change to 10x10 fuel should result in lower PCTs for ATWS as long as the power response is similar to that for 9x9 fuel. The peak neutron flux will be obtained from the RETRAN calculations. This peak power will be compared with the peak power computed by GE in Ref. 1. If the peak power calculated by RETRAN for U2C9 is reasonably close to the GE-calculated value, then we can conclude, based on the large available margin (737 °F) and the larger clad-surface-area-to-fuel-volume ratio associated with 10x10 fuel, that the peak clad temperature (PCT) limit of 2200 °F will not be exceeded.

The final parameter of interest, Peak Suppression Pool Temperature (PSPT), is governed by the Hot Shutdown Boron Weight and the Hot Shutdown Reactivity as well as the power shape and reactivity coefficients under natural circulation conditions. These parameters will be calculated for U2C9 in order to evaluate the impact of the 10x10 fuel load on suppression pool temperature response for ATWS. The PP&L SABRE code will be used to evaluate the U2C9 PSPT for the MSIV-closure and PREGO ATWS events. SABRE results have been used by the NRC in evaluating reactor water level control strategies for ATWS mitigation (Ref. 3), and the NRC has concluded that SABRE predictions for ATWS scenarios are comparable to results obtained with TRAC-BF1 (with 1-D neutronics) and RAMONA-4B (Ref. 4). The Hot Shutdown Boron Weight and Hot Shutdown Reactivity are input data for the SABRE calculation. Additional SABRE input data, axial power shape and nodal reactivity coefficients for U2C9, will be computed using the SIMULATE and SIMTRAN computer codes. If the PSPT calculated by SABRE for U2C9 is less than the 190 °F limit specified in the GE analysis, then we can again conclude that the peak suppression pool temperature limit remains satisfied.

PP&L plans to complete this study by September 30, 1996. The applicability of this study will be determined for cycles beyond U2C9 if the fuel design or fuel mix changes. Any questions on this letter should be directed to Mr. R. Sgarro at (610) 774-7552.

Very truly yours,

R. G. Byram

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