Kevin Cimorelli
Site Vice President

Susquehanna Nuclear, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3795 Fax 570.542.1504
Kevin.Cimorelli@TalenEnergy.com

June 2, 2020

Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10 CFR 50.90

SUSQUEHANNA STEAM ELECTRIC STATION RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED LICENSE AMENDMENT REQUESTING REVISION TO THE DOSE CONSEQUENCE ANALYSIS FOR A LOSS OF COOLANT ACCIDENT
PLA-7865

Docket No. 50-387 and 50-388

References:
1) Susquehanna letter to NRC, "Proposed Amendment to Licenses NPF-14 and NPF-22: Revision to the Dose Consequence Analysis for a Loss of Coolant Accident (PLA-7823)," dated January 2, 2020 (ADAMS Accession No. ML20002B254).

2) NRC email to Susquehanna, "Request for Additional information for the Revision of the Susquehanna Steam Electric Station, Units 1 and 2, Updated Final Safety Analysis Report and Technical Specifications to Modify the Design Basis Accident Loss of Coolant Accident (EPID: L-2020-LLA-0000)," dated April 21, 2020.

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), submitted, in Reference 1, a request for an amendment to the Updated Final Safety Analysis Report (FSAR) and Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22. The proposed amendment would modify the Current Licensing Basis for the Design Basis Accident (DBA) Loss of Coolant Accident (LOCA) analysis described in the SSES FSAR, as previously reviewed by the NRC. The proposed changes would utilize an updated version of the ORIGEN code, introduce a new source term to account for the introduction of ATRIUM 11 fuel, use new inputs/assumptions that decrease the assumed Emergency Safety Feature leakage into secondary containment, increase the assumed maximum allowable Standby Gas Treatment System exhaust flow rate from secondary containment, and increase the allowed control structure unfiltered inleakage that is assumed in the DBA LOCA dose analysis.

The NRC provided a Request for Additional Information (RAI) in Reference 2. This RAI was discussed with members of the NRC staff during a clarification call on April 21, 2020. Enclosure 1 to this letter provides Susquehanna's response to the NRC's RAI.

Susquehanna has reviewed the information supporting a finding of No Significant Hazards Consideration and the Environmental Consideration provided to the NRC in Reference 1 and determined the information provided herein does not impact the original conclusions in Reference 1.

There are no new or revised regulatory commitments contained in this submittal.

Should you have any questions regarding this submittal, please contact Ms. Melisa Krick, Manager – Nuclear Regulatory Affairs, at (570) 542-1818.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 6/2/2020

K. Cimorelli

Enclosure:

1. Response to Request for Additional Information

Copy: NRC Region I
Mr. M. Rossi, NRC Resident Inspector
Ms. S. Goetz, NRC Project Manager
Mr. M. Shields, PA DEP/BRP

Enclosure 1 of PLA-7865

Response to Request for Additional Information

Response to Request for Additional Information

On January 2, 2020, Susquehanna Nuclear, LLC (Susquehanna), submitted a license amendment request (LAR) for the Susquehanna Steam Electric Station (SSES). The proposed amendment would modify the Current Licensing Basis (CLB) for the Design Basis Accident (DBA) Loss of Coolant Accident (LOCA) analysis described in the SSES Updated Final Safety Analysis Report (FSAR), as previously reviewed by the NRC. By email dated April 6, 2020, the NRC provided a draft request for additional information (RAI). The draft RAI was discussed with members of the NRC during a clarification call on April 21, 2020. Subsequently, on April 21, 2020, the NRC provided its final RAI. The response to this request for additional information (RAI) is provided below.

NRC RAI 1.A

Provide additional information expanding on the technical basis provided for the removal of the non-ESF [Emergency Safety Feature] leakage from the current licensing basis dose consequence analysis. The requested information should focus on the technical basis from a dose consequence perspective as opposed to referencing assumptions included or excluded in regulatory guidance.

Susquehanna Response

Discussion of the technical basis for removal of the non-ESF leakage from the CLB DBA LOCA analysis requires an understanding of the flow path and design of the non-ESF leakage pathways – Control Rod Drive (CRD) insert/withdrawal lines and Scram Discharge Volume (SDV). Each are discussed in the following paragraphs.

The SDV is part of the CRD System. As is typical of Boiling Water Reactor (BWR) CRD systems, control rod motion is ultimately caused by movement of a drive piston internal to the drive mechanism. The motive force for movement of the drive piston is pressurized water on either side of the drive piston. The underside of the piston receives pressurized water from the Hydraulic Control Unit (HCU) during a scram. The over side of the drive piston is filled with normal reactor coolant prior to a postulated scram. During the scram, the pressurized HCU water provides the motive force for the upward movement of the drive piston which displaces reactor coolant on the over side of the piston into the SDV. The SDV is vented during normal plant operation and is isolated on a scram signal. Therefore, the activity in the SDV post-LOCA is equivalent to the reactor coolant activity during normal operation.

The CRD insert and withdrawal lines are the pipes connected to the containment penetrations associated with the flow to the CRD mechanism (e.g., CRD Charging Water, CRD Drive Water

Header, CRD Cooling Water Header) and the return flow (e.g., CRD Exhaust Water, SDV). SSES FSAR, Section 6.2 describes these lines as:

> The control rod drive system insert and withdraw lines penetrate the drywell.
>
> The CRD insert and withdrawal lines are not part of the reactor coolant pressure boundary since they do not directly communicate with the reactor coolant. The classification of these lines is quality group B, and they are designed in accordance with ASME [American Society of Mechanical Engineers] Section III, Class 2. The basis on which the CRD insert and withdrawal lines are designed is commensurate with the safety importance of maintaining the pressure integrity of these lines.
>
> It has been accepted practice not to provide automatic isolation valves for the CRD insert and withdrawal lines to preclude a possible failure mechanism of the scram function. The control rod drive insert and withdrawal lines can be isolated by the solenoid valves outside the primary containment. The lines that extend outside the primary containment are small and terminate in a system that is designed to prevent out-leakage. Solenoid valves are normally closed, but open on rod movement and during reactor scram. In addition, a ball check valve located in the control rod drive flange housing automatically seals the insert line in the event of a break. Finally, manual shutoff valves are provided.

The CRD insert and withdrawal lines are filled with water from the Condensate Storage Tank (CST) during normal operation. This is clean water from a radioactivity standpoint. The only change to the water in these systems post-scram is the above discussed SDV.

The BWR CRD system is not unique to SSES. A review of NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," Revision 4, determined that the Standard TS do not include the SDV in TS 5.5.2, "Primary Coolant Sources Outside Containment." TS 5.5.2 minimizes leakage from systems outside primary containment that contain highly radioactive fluid after a transient or accident. As discussed above, the SSES CRD insert and withdrawal lines and the SDV will not contain highly radioactive fluid post-LOCA. Therefore, leakage from these systems need not be included in the dose consequence analysis or included in TS 5.5.2. This change aligns with current industry dose consequence modeling and meets the requirements of Regulatory Guide 1.183.[1]

NRC RAI 1.B

Provide information to demonstrate, by sensitivity or scoping evaluations, that the impact on the TEDE [Total Effective Dose Equivalent] criteria as a result of removing the non-ESF leakage

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[1] The proposed change to eliminate the SDV from TS 5.5.2 does not impact Susquehanna's response to address the concerns of NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping."

would be negligible. Provide a summary or discussion of such an analysis performed and the conclusions drawn.

Susquehanna Response

As stated in the response to Question 1.A, the water contained in these systems is typically water from the CST. Under post-accident conditions, the line may contain activity at normal operation reactor coolant conditions which is limited by TS 3.4.7, “RCS Specific Activity.” Due to the small amount of activity in normal reactor coolant compared with post-LOCA fluid, the dose contribution is considered negligible.