NRR-PMDAPEm Resource

From: Singal, Balwant
Sent: Wednesday, April 02, 2014 9:30 AM
To: 'Sterling, Lance'
Subject: Request for Additional Information - License Amendment Request to Revise Fire Protection Program - TACs MF2477 and MF2478
Attachments: MF2477-RAI.docx

Lance,

By letter dated July 23, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13212A243), STP Nuclear Operating Company (STPNOC, the licensee) submitted license amendment request to revise South Texas Project (STP), Units 1 and 2, Fire Protection Program related to alternate shutdown capability. Specifically, STPNOC requested crediting the performance of certain operator actions in the control room, including one automatic operation, in the event a fire requires evaluation of the control room. The NRC staff requires the attached additional information to complete its review.

Draft Request for Additional information (RAI) was transmitted on March 25, 2013 and a RAI clarification call was held on April 1, 2014. It was agreed that STPNOC will provide the response within 30 days from the date of this e-mail. However, if more time is needed to respond to the RAI request, STPNOC will co-ordinate the revised date with the NRC staff.

Please treat this e-mail as formal transmittal of RAIs from our reactor systems branch. As discussed, additional RAIs from the other review branches are also expected.

Please advise if you have any questions.

Thanks.

Balwant K. Singal
Senior Project Manager (Comanche Peak and STP)
Nuclear Regulatory Commission
Division of Operating Reactor Licensing
Balwant.Singal@nrc.gov
Tel: (301) 415-3016
Fax: (301) 415-1222

Hearing Identifier: NRR_PMDA
Email Number: 1199

Mail Envelope Properties (192BB59294514E41B5305C3C828676080149B17A508B)

Subject: Request for Additional Information - License Amendment Request to Revise Fire Protection Program - TACs MF2477 and MF2478
Sent Date: 4/2/2014 9:30:27 AM
Received Date: 4/2/2014 9:30:00 AM
From: Singal, Balwant

Created By: Balwant.Singal@nrc.gov

Recipients:
"'Sterling, Lance'" <lsterling@STPEGS.COM>
Tracking Status: None

Post Office: HQCLSTR02.nrc.gov

Files	**Size**	**Date & Time**
MESSAGE	1475	4/2/2014 9:30:00 AM
MF2477-RAI.docx	30464	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received: ZZZ

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST

SOUTH TEXAS PROJECT, UNITS 1 AND 2

FIRE PROTECTION PROGRAM

ALTERNATIVE SHUTDOWN CAPABILITY

By letter dated July 23, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13212A243), STP Nuclear Operating Company (STPNOC, the licensee) submitted license amendment request to revise South Texas Project (STP), Units 1 and 2, Fire Protection Program related to alternate shutdown capability. Specifically, STPNOC requested crediting the performance of certain operator actions in the control room, including one automatic operation, in the event a fire requires evaluation of the control room. The NRC staff requires the following additional information to complete its review:

1. General

The analyses indicate that spurious actuations are assumed to occur at the time of reactor trip. Spurious actuations can occur at any time during the fire, or not at all. Please state how was it determined that spurious actuations should be assumed to occur at the time of reactor trip?

2. Enclosure 1, Page 9

The licensee states, in part, that:

> [C]ertain actions within the control room must be successful to assure that RCS [reactor coolant system] process variables do not exceed the limits predicted for a loss of normal ac [alternating current] power (i.e., [Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50,] Appendix R, Section III.L requirement) until control is successfully transferred.

10 CFR 50, Appendix R, Section III.L states, During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, [or] rupture of the containment boundary.

Among the South Texas Project Electric Generating Station (STPEGS), Updated Final Safety Analysis Report (UFSAR), Chapter 15 accident analyses, loss of normal ac power is classified as an anticipated operational occurrence (AOO). Acceptable analysis results for this event resemble the acceptance criteria of Appendix R, Section III.L. For example, AOO analysis results must show that there is no fuel clad damage, and that the RCS pressure boundary remains intact.

Please compare the acceptance criteria of Appendix R, Section III.L to the UFSAR acceptance criteria for AOO’s, and explain any differences that are identified.

Please explain how is the AOO design requirement, which prohibits an AOO from developing into a more serious event, considered in the fire hazards analyses?

3. Enclosure 1, Page 9

UFSAR, Chapter 15.2.6, does not report the results of a loss of normal ac power analysis. It states, "Plant specific analysis has shown that a loss of normal FW [feedwater] with a subsequent loss of AC power is the most limiting Condition II event in the decrease in secondary heat removal category with respect to the pressurizer overfill criterion and is analyzed in Section 15.2.7. Therefore, detailed analytical results for a loss of AC power transient will not be presented here."

As a Chapter 15 accident analyses, loss of normal ac power is less limiting than the loss of normal feedwater without offsite power. However, the loss of normal ac power could be the more limiting event when used as an Appendix R, Section III.L requirement. Please provide the following:

- Predicted values, from the loss of normal ac power analysis, that are to be used as limits.
- Identify the RCS process variables that must be maintained within those predicted for a loss of normal ac power and please explain how these RCS process variables were selected.
- Provide a comparison of the most limiting fire hazard analysis results against the loss of normal ac power analysis results, as per Appendix R, Section III.L.

4. Enclosure 1, Page 9

In UFSAR, the more limiting Condition II event (i.e., the loss of feedwater without offsite power event) is chosen because it comes closer to challenging the pressurizer overfill criterion. The pressurizer overfill criterion is not among the criteria of Appendix R, Section III.L.

Please explain the role (if any) that the pressurizer overfill criterion plays in the fire hazard analyses. Refer to Case 2, "Spurious Opening of One Pressurizer PORV [power operated relief valve]" (Section A2.2.3, Enclosure 1, Attachment 2, Page 10).

5. Enclosure 1, Attachment 2, Page 10

In Case 2, the pressurizer fills and the PORVs relieve water. It is stated that the PORVs are qualified to pass water. Please state how the PORVs have been qualified to pass water.

6. Enclosure 1, Attachment 2, Page 13

In Case 2, Figure A2.3.3 depicts a period (until about 2,000 seconds) of repeated opening and closing of the PORV. Please explain how the PORV automatic control system circuitry has been qualified to reliably open and close the PORV, as needed.

7. Enclosure 1, Attachment 2, Pages 10 and 11

An American Nuclear Society (ANS) design requirement for AOOs[1] states, "Condition II events (i.e., anticipated operational occurrences) shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action." In Case 2,

---

1 ANS-N18.2-1973, *"Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*, August 6, 1973

the pressurizer fills and one PORV relieves water. Figure A2.3.3 depicts about 20 minutes of water relief through the PORV. Please explain how the plant will remain capable of returning to power after having experienced 20 minutes of water relief.

8. Enclosure 1, Attachment 2, Page 13

Stable plant conditions are maintained until excess letdown is placed in service (7,210 seconds), and pressurizer water level returns to the indicating range (16,928 seconds). The pressurizer is water-solid for about 4-1/2 hours.

Please explain how would the pressurizer pressure be controlled while the pressurizer is water-solid? Also, please define “stable plant conditions”, and explain how stable conditions would be maintained for 4-1/2 hours.

9. Enclosure 1, Attachment 2, Page 13

In Case 2, Figure A2.3.3 depicts a period (between 3,000 seconds and 14,000 seconds) falling and rising pressure. Please explain this pressure variation. (charging pumps secured at 610 seconds and the shutoff head of the high head safety injection pumps is stated as approximately 1,700 psia.)

10. Enclosure 1, Attachment 2, Pages 25-27

For Cases 1a and 1b, in which offsite power is assumed to be lost, please justify the assumption that pressurizer backup heaters are available.

11. Enclosure 1, Attachment 2, Page 29

For Case 1a, in which offsite power is assumed to be lost, please justify the assumption that (1) the PORV control system circuitry is qualified to cycle the PORV until 2,950 seconds, and (2) that power can be supplied to the PORV for that length of time.

12. Enclosure 1, Attachment 1, Section A1.8

This analysis is geared toward minimizing pressurizer level; to show that subcooling margin is not lost. Subcooling margin, as depicted in Figure A1.8.9, reaches a minimum at about the time the indicated pressurizer level drops to 7.1%; but remains positive. Please state the minimum value of subcooling margin? Also, please estimate the uncertainties associated with this value and what causes the subcooling margin to increase after reaching its minimum?

Table A1.8 and Figure A1.8.2 indicate that pressurizer water level is 7.1% and constant for approx. 608 seconds. Figures A1.8.1, A1.8.3, A1.8.4, A1.8.5, A1.8.6, and A1.8.9 depict a change in trend between 600 and 650 seconds. Please explain what causes this change?