NRR-PMDAPEm Resource

From: Regner, Lisa
Sent: Wednesday, August 10, 2016 1:40 PM
To: Lance Sterling (lsterling@stpegs.com)
Cc: Drew Richards (amrichards@STPEGS.COM)
Subject: DRAFT RAI question - CRDM LAR
Attachments: Draft RAI MF7577.docx

Lance,
Here are the draft RAI questions for the permanent removal of D-6 CR LAR. I will process these formally for issuance later next week. Let me know if there are any surprises.
Lisa

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Sent Date: 8/10/2016 1:40:19 PM
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From: Regner, Lisa

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Recipients:
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Options
Priority: Standard
Return Notification: No
Reply Requested: No
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<u>REQUEST FOR ADDITIONAL INFORMATION</u>
<u>SOUTH TEXAS PROJECT UNIT 1</u>
<u>REQUEST TO OPERATE WITH 56 CONTROL RODS</u>

SRXB – RAI 1

The licensee states, in part, that the required Reactor Coolant System (RCS) Shutdown Margin (SDM) boron concentrations for Operating MODES 3, 4, and 5 will be higher with Control Rod D-6 removed. Additionally, Table 8 of the LAR provides a summary of the SDM calculated at the End of Cycle (EOC) for the four representative cycles performed for the limiting Hot Zero Power (HZP) main steam line break accident. It is unclear to the NRC staff how the SDM was determined. The calculated SDM is evaluated for each core reload design to satisfy General Design Criteria (GDC) 28, "Reactivity Limits." The NRC staff requests that the licensee provide a discussion on how the SDM was calculated for Cycle 20, and the multi-cycle assessment.

SRXB – RAI 2

Table 7 of the LAR discusses the impact of Control Rod D-6 on the key safety parameters related to UFSAR Chapter 15 safety analyses. The discussion only references to the bounding UFSAR Chapter 15 analyses with no further discussion of how the removal of the control impacts the inputs/assumptions of the analyses. The NRC staff requests that the licensee provide the following:

a) For each Chapter 15 analysis:

   a. Please discuss how consideration of Control Rod D-6 was previously incorporated into each accident analysis (e.g., control rod D-6 was part of the shutdown bank that was inserted into the core following reactor trip initiated by a turbine trip).

   b. Please discuss how the removal of the control rod impacts the key safety parameters (e.g., since the shutdown banks are assumed to insert during this event, the overall trip reactivity is decreased with the removal of Control Rod D-6).

   c. Please provide the basis for events that are not impacted by removal of control rod D-6.

   d. If there is an impact on the key safety parameters, please provide an estimate of the magnitude of the change to the key safety parameter.

b) For the Control Rod Ejection Accident (CREA), Table 7 identifies "various" as the sections of the reload methodology. Please specify these locations and please identify the key safety parameters related to this accident (i.e., those impacted and not impacted by the Control Rod D-6 removal).

c) The staff identified several discrepancies between the key safety parameters impacted identified in Table 7 and the key safety parameters identified for each accident in the reload methodology. Please discuss why there are differences between the documents as identified below (Note that for this RAI, Table 7 of the LAR is abbreviated as Table 7):

   - Feedwater System Malfunctions (reduction in FW temperature and increased FW Flow): Table 7 identifies Trip Reactivity as impacted while the reload methodology does not identify Trip Reactivity as a key safety parameter.

   - Loss of External Load and Turbine Trip: Table 7 identifies Trip Reactivity as impacted while the reload methodology does not identify Trip reactivity as a key safety parameter. The reload methodology identifies moderator density coefficient (MDC) as a key safety parameter while Table 7 does not identify MDC as impacted.

   - Feedwater System Pipe Break: Table 7 identifies Trip Reactivity as impacted while the reload methodology does not identify Trip Reactivity as a key safety parameter. The reload methodology identifies Shutdown Margin (SDM) as a key safety parameter while Table 7 does of the LAR does not identify SDM as impacted.

   - Partial Loss of Forced Reactor Coolant Flow, Complete Loss of Forced Reactor Coolant Flow, and Rod Cluster Control Assembly (RCCA) Misoperation: The reload methodology identifies MDC as a key safety parameter while Table 7 does not identify MDC as impacted.

   - Startup of Inactive Reactor Coolant Loop at an Incorrect Temperature: The reload methodology identifies MDC and Shutdown Margin as a key safety parameter while Table 7 does not identify MDC and Shutdown Margin as impacted.

d) In the column in Table 7, several comments state that an analysis is bounded by another. Please discuss the basis for why these analyses are bounded by another and confirm that these bounding assumptions are unchanged with the removal of the control rod.

SRXB – RAI 3

The licensee states, in part, in Table 3 of the application that the total rod worth is evaluated on a cycle-specific basis to ensure that the SDM and trip reactivity limits are met. It is unclear to the NRC staff whether the licensee has evaluated the influence and impact of total rod worth on CR D-6 in relation to the UFSAR Chapter 15 analyses. The calculated total rod worth is evaluated for each core reload design to satisfy GDC 28, "Reactivity Limits." The NRC staff requests that the licensee provide clarification of the removal of CR D-6 on total rod worth and the parameters in relation to the UFSAR Chapter 15 analyses.

SRXB – RAI 4

Table 5 of the LAR contains additional nuclear design key safety parameters that are not part of the reload methodology. Have these additional parameters in Table 5 been analyzed in past reloads? Have these additional parameters in Table 5 been incorporated into the reload guidance such that they will be analyzed in future reloads?

SNPB – RAI 1

GDC 10, "Reactor Design," states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Such margin is commonly demonstrated using computational models to simulate how the system would behave during normal operation and anticipated operational occurrences. Because the results of these simulations are used to confirm that such margin exists, the simulations themselves and the computer models which are used to perform them must be trustworthy. The Advanced Nodal Code (ANC) is used to perform analyses for these scenarios and will have a change in inputs due to the removal of the D-6 control rod. Provide justification for the continued use of ANC with the removal of the D-6 control rod. This should include a demonstration that any change to the simulations considered (i.e., N-1 to N-2 rods out) are within the capabilities of ANC and the scope of the initial approval of ANC.

SNPB – RAI 2

GDC 10, "Reactor Design," states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Such margin is commonly demonstrated using computational models to simulate how the system would behave during normal operation and anticipated operational occurrences. Because the results of these simulations are used to confirm that such margin exists, the simulations themselves and the computer models which are used to perform them must be trustworthy. One such simulation, the Hot Zero Power Main Steam Line Break (HZP MSLB), uses multiple computer codes to simulate the scenario. Provide further details on the methodology for performing the HZP MSLB analysis. Specifically address how the analysis of record, generated by RETRAN, was used in conjunction with ANC and VIPRE to ensure that there was margin to DNB.