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June 9, 1992

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Subject: **Docket No. 50-206**
Documents Referenced in Amendment Application No. 205
Technical Specification Changes Related to Moderator Temperature Coefficient
San Onofre Nuclear Generating Station Unit 1 (SONGS 1)

References:
1. Letter, Harold B. Ray, SCE to NRC, "Amendment Application No. 205, Moderator Temperature Coefficient Change Request," May 1, 1992
2. Letter, R. M. Rosenblum, SCE to NRC, "Main Steam Line Break Analysis with Revised Moderator Temperature Coefficient," May 18, 1992

Enclosed is a copy of Westinghouse letter no. SCE-91-611, from Mr. S. A. Pujadas to Mr. T. Yackle of SCE, dated November 6, 1991. This letter was referenced in our recent amendment application (Reference 1) concerning changes to Technical Specification limits for the end-of-cycle Moderator Temperature Coefficient (MTC), safety injection line minimum boron concentration, and shutdown margin. In a recent telephone conversation, the NRC staff requested a copy of the references in that application.

The amendment application also referenced Westinghouse letter no. SCE-92-518, from Mr. S. A. Pujadas of Westinghouse to Mr. T. Yackle of SCE, dated April 6, 1992. This letter contained results of Main Steam Line Break (MSLB) reanalysis performed by Westinghouse to support the amendment application, and included information proprietary to Westinghouse. The results of the MSLB reanalysis have subsequently been issued by Westinghouse as WCAP-13346 (Westinghouse Proprietary) and WCAP-13347 (Westinghouse Non-Proprietary) reports. We submitted both versions of the WCAP reports to the NRC by letter dated May 18, 1992 (Reference 2). These WCAP reports supersede the

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April 6, 1992 Westinghouse letter referenced in our amendment application, and therefore, that letter is not being submitted.

If you have any questions concerning this matter, or if you require any additional information, please contact me.

Very truly yours,

R M Rosenblum

Enclosure

cc: J. B. Martin, Regional Administrator, NRC Region V
George Kalman, NRC Senior Project Manager, San Onofre Unit 1
J. O. Bradfute, NRC Project Manager, San Onofre Unit 1
C. W. Caldwell, NRC Senior Resident Inspector, San Onofre Units 1, 2&3

Enclosure

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Mr. T. Yackle, Discipline Manager
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SCE-91-611
November 6, 1991
ET-NSL-OPL-I-91-635

SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1
MTC - Technical Specification

Dear Mr. Yackle:

Provided in the attached text is our response to Southern California Edison's request for information regarding the basis for Technical Specification 3.9, Moderator Temperature Coefficient, as it relates to the safety analyses for San Onofre Nuclear Generating Station - Unit 1 (SONGS-1).

If an update to the SONGS-1 UFSAR is made by SCE, please be advised the variation of Keff with core average temperature at a constant pressure of 1000 psia provided in Figure A1-2 of Reference 6 to the attachment should be used to replace UFSAR Figure 15-2.5. This figure is applicable to the current licensing basis Steamline Break Analyses provided in References 6, 7, and 8 of the attachment.

The current UFSAR Figure 15-2.6 is appropriate as shown to illustrate the effect of power generation in the core on overall reactivity. This effect is consistent with the analysis assumptions for Reference 8 of the attachment and conservatively bounds the analysis assumption used in References 6 and 7 of the attachment. The equivalent analysis assumption used in the Reference 6 and 7 analyses is that shown in Figure A1-3 of Reference 6 to the attachment.

Please also note that other UFSAR changes appear to be needed in Section 6.2.1.3.2 of the SONGS-1 UFSAR since the text in this section states that the SLB mass and energy releases were calculated using the MARVEL computer code. As indicated in References 7 and 8 of the attachment, the LOFTRAN computer code is used to perform these calculations.

If there are any questions on the information provided or if you need any additional information, please contact Gary Ament at (412) 374-4897.

Very truly yours,

S. A. Pujadas

S. A. Pujadas, Manager
Western Area
Domestic Customer Projects

/slf
Attachment

ATTACHMENT

Provided herein is the response to Southern California Edison's request for information regarding the basis for Technical Specification 3.9, *Moderator Temperature Coefficient*, as it relates to the safety analyses for San Onofre Nuclear Generating Station - Unit 1 (SONGS-1).

In Reference 1, Southern California Edison (SCE) provided background information and documented concerns related to the basis for MTC Technical Specification 3.9, in particular, as it relates to the safety analyses supporting evaluation of the various steamline break (SLB) events.

SCE specifically requested that the following items be addressed by Westinghouse:

1. The correct basis for the MTC technical specification for SCE.
2. An explanation of the relationship between the MDC value used in the bounding accident and the value used in the SLB events.
3. Verification that the SLB event MDC value is properly bounded or not impacted by the surveillance performed in accordance with Technical Specification 3.9.

The correct basis for the SONGS-1 Technical Specification 3.9 is that previously provided by Westinghouse (Reference 2) supporting the change from a specification on burnup to a specification on negative MTC. This correct basis is not the same as that contained in the current SONGS-1 Technical Specification.

While it is accurate in a physical sense to state that a more negative MTC can result in more limiting conditions resulting from excessive cooldown events (e.g., SLB), it is inappropriate to cite the analyses of the SLB events within the basis of this End-of-Life (EOL) MTC specification. The reason for this is directly related to the application of the Westinghouse SLB analysis methodology (Reference 3) in conjunction with the Westinghouse Reload Safety Evaluation Methodology (Reference 4) in initially performing and then confirming the applicability of the SLB analyses on a cycle-by-cycle basis.

Specifically, the Westinghouse Reload Safety Evaluation Methodology is applied in combination with the Westinghouse SLB analysis methodology in evaluating the SLB events. Although not all of the discussions in Reference 3 apply to SONGS-1, the reactivity feedback modeling assumptions (which includes consideration for moderator density / temperature effects) described in Reference 3 are consistent with those used in the SLB analyses for SONGS-1. These SLB modeling assumptions conservatively reflect end-of-cycle conditions with the most reactive RCCA stuck in its fully withdrawn position. The following summarizes the application of the Westinghouse SLB and Reload Safety Evaluation Methodologies in analyzing the SLB events for SONGS-1.

The moderator, boron, and Doppler coefficients are initially computed using a multidimensional, full-core, neutron-diffusion nuclear model. The coefficients are computed with the most reactive control rod stuck out of the core and all other rods fully inserted. Therefore, the neutron flux distribution used in calculating the coefficients is representative of that which is present during the SLB transient and the moderator and boron coefficients are appropriately calculated over the SLB transient temperature range. The Return-to-Power coefficient (typically referred to as the Doppler coefficient) is calculated at a constant volume-average moderator density at several power levels. Since this coefficient is computed with a stuck control rod, the power distribution is tilted toward the stuck rod and because of power effects, the moderator density in the vicinity of the stuck rod is lower than the core average moderator density. Thus, the coefficient is a combination of moderator feedback effects, and Doppler temperature effects with the Doppler feedback term dominating. The boron coefficient is computed at several moderator densities using the same calculation model as was used for the moderator and return-to-power coefficients.

These coefficients are then used in the transient code (LOFTRAN, Reference 5) to compute the integrated reactivity effects. In LOFTRAN, the moderator coefficient is used as a function of moderator density, and the reactivity insertion due to the moderator density change is obtained by integrating the coefficient from Hot Zero Power (HZP) to the conditions at the SLB statepoint. The Doppler Power defect gives the reactivity change due to the core returning to power. This defect is given as a function of core relative power and only adds negative reactivity when the core is at a power level other than zero.

The boron coefficient is used to give the negative reactivity insertion as boron is added to the core (i.e., via the Safety Injection System).

For SLB events occurring while the reactor is at power (e.g., the SONGS-1 SLB events for mass and energy releases inside and outside containment), the reactivity feedback model used is the same as that applied in the SLB event initiated from HZP conditions. For SLB events occurring from power, the transient takes place in four general phases as described in Reference 3; Phase I - before reactor trip, Phase II - immediately following reactor trip, Phase III - short period following reactor trip, and Phase IV - shutdown phase. Although Reference 3 deals with DNB concerns and not mass and energy releases, the categorization of phases in Reference 3 is useful for the following discussion.

In Phase I, positive reactivity insertion begins due to the core cooldown in the presence of a negative moderator temperature coefficient. During this period of time, the concept of a "stuck rod" has no relevance. For the SONGS-1 HFP SLB cases, Phase I has a duration of two seconds from the time of the break. During this period, the core cooldown has progressed less than 0.2°F. Therefore, it is acceptable to use the stuck rod coefficients throughout this phase of the transient since the reactivity change associated with a 0.2°F cooldown is negligible, regardless of the coefficients used.

In Phase II, negative reactivity is inserted by the rods which includes an allowance for the most reactive rod fully withdrawn (i.e., "stuck rod") from the core. A rapid decrease in core power occurs and the plant approaches hot shutdown conditions. The rate at which core power decreases is unaffected by the reactivity coefficients during this phase because of the large negative reactivity insertion associated with the reactor trip.

In Phase III, the continued cooldown caused by the break results in a return to power as in the case initiated from zero power (and modeling the aforementioned EOL "stuck rod" reactivity coefficients).

Finally in Phase IV, reactor power ultimately decreases to decay heat levels as boron concentration in the reactor coolant reaches the core, steam blowdown decays, and the cooldown is terminated.

For all SLB cases analyzed, reactivity checks at various SLB statepoints are made to confirm the transient modeling of the coefficients. These reactivity checks are performed using multidimensional calculations to verify that the integrated reactivity insertion applied in the system transient is conservative. In addition, these confirmatory calculations are performed assuming the maximum allowable burnups for the reload cycle of interest and for the previous cycle. As described in Reference 3, it is intended that the reactivity variation modeled in the system code (i.e., LOFTRAN) be conservative relative to the time-independent, spatial diffusion code calculations. These reactivity checks are performed whenever there is a new system transient analysis (i.e., new SLB statepoints) or when core changes result in potential core reactivity balance changes (e.g., new fuel reload). The application of the core reload design methods used in the Westinghouse reload evaluation process is described in detail in Reference 4. It is this application of the design methods which places restrictions on the range of parameters and conditions important to the reload design (e.g., burnup) and which is used to confirm the continued applicability of the aforementioned SLB analysis assumptions on a cycle-by-cycle basis. Note that while the Westinghouse reload evaluation process described in Reference 4 only specifically addresses SLB for core response (i.e., DNB), all the SONGS-1 SLB analyses, including those performed for SLB mass and energy releases inside and outside containment, are considered in the reload evaluation process due to their dependency on the SLB reactivity modeling in the transient analyses.

Because of the application of this process which directly confirms the conservatism contained in the SLB reactivity assumptions on a cycle-by-cycle reload design basis, the Technical Specification basis provided in Reference 2 refers only to the UFSAR safety analyses in general terms and intentionally does not cite any particular accident analysis. However, there are other UFSAR licensing basis events (e.g., Increase in Feedwater Flow, Control Rod Bank Withdrawal at Power, Feedline Break) which do assume maximum moderator reactivity feedback conditions in their respective safety analyses. For these events, which are not evaluated in the same manner as SLB, the most negative MTC defined in Technical Specification 3.9 is of importance. Hence, a most negative MTC (most positive MDC) is assumed for these events and the analyses of these type events are the real basis for Technical Specification 3.9 on Moderator Temperature Coefficient.

The moderator and Doppler reactivity coefficient modeling assumptions for all the UFSAR Chapter 15 licensing basis events are also defined in Table 15.0-3 of the SONGS-1 UFSAR. The MTC in Technical Specification 3.9 can be derived from the values in this table.

For the current licensing basis SLB analyses for SONGS-1 as documented to SCE via References 6, 7, and 8, the variation of K_{eff} with core average temperature at a constant pressure of 1000 psia is that shown in Figure A1-2 of Reference 6. The effect of power generation in the core on overall reactivity as modeled in the SLB analyses is conservatively bounded by that currently shown in Figure 15-2.6 of the SONGS-1 UFSAR. These figures are appropriate to illustrate the most limiting moderator and Doppler reactivity coefficient modeling assumptions supporting the licensing basis analyses of the SLB events as described in Sections 6.2.1.3.2 and 15.2 of the SONGS-1 UFSAR and as referenced in Table 15.0-3 of UFSAR section 15.0.

Finally, while it is theoretically possible to correlate the ARO (All-Rods-Out) HFP MTC to the N-1 HZP MTC at 1000 psia, it is not practical nor would it be desirable to do this on a cycle-by-cycle basis. The ARO HFP MTC is primarily a function of core average burnup and core average neutron leakage. The steamline break MTC is also a function of these parameters, but in addition is a strong function of the N-1 control rod worth, worst stuck rod location, and worst stuck rod worth. Therefore, since the relationship between the ARO HFP MTC and the HZP N-1 rodded MTC changes with each cycle, attempting to place a Technical Specification on the MTC component of the SLB reactivity model while maintaining a realistic return-to-power model would be an administrative burden that is not considered necessary. The current burnup limitation placed on each Reload Safety Evaluation is considered by Westinghouse to be sufficient to assure that the reactivity modeling for the SLB is technically acceptable and remains within the bounds of the licensing basis safety analyses.

REFERENCES

1) Letter dated September 6, 1991, *Modification to Moderator Density Coefficient Input for Steam Line Break (SLB) Accident Analysis of Record (AOR), SONGS-1*, Mr. A. J. Eckhart (SCE) to Mr. S. A. Pujadas (W).

2) 86SC*-G-0001, January 3, 1986, *SONGS-1 Technical Specification Change*, B. D. McKenzie (W) to Mr. J. Rainesverry (*sic*) (SCE).

3) WCAP-9226, Rev. 1, (Proprietary) and WCAP-9227 (Non-Proprietary), January, 1978, *Reactor Core Response to Excessive Secondary Steam Releases*, S. D. Hollingsworth, D. C. Wood.

4) WCAP-9272-P-A (Proprietary), July, 1985, *Westinghouse Reload Safety Evaluation Methodology*, S. L. Davidson (Ed).

5) WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984, *LOFTRAN Code Description*, T. W. T. Burnett.

6) SCE-90-697, November 2, 1990, *Steamline Break Core Response Analysis for SONGS-1*, H. C. Calton (W) to Mr. Bernie Carlisle, (SCE).

7) SCE-90-693, October 29, 1990, *Main Steamline Break M&E Releases for SONGS-1*, H. C. Calton (W) to Mr. Bernie Carlisle, (SCE).

8) SCE-91-579, August 12, 1991, *Upper Head Temperature Used in Safety Analysis*, S. A. Pujadas (W) to Mr. T. R. Yackle, (SCE).