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June 30, 1992

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

Subject: **Docket No. 50-206**
Revision 1 of Amendment Application No. 205, Moderator Temperature Coefficient Change Request
San Onofre Nuclear Generating Station Unit 1 (SONGS 1)

Reference: Letter, Harold B. Ray, SCE to NRC, "Amendment Application No. 205, Moderator Temperature Coefficient Change Request," May 1, 1992

This letter forwards Revision 1 to our Amendment Application No. 205 dated May 1, 1992 (See Reference). This revision to the amendment application is editorial in nature and does not result in a change to the safety analysis included in the amendment application. Amendment Application No. 205 requested changes to several SONGS 1 Technical Specifications including Specification 3.5.2, "CONTROL ROD INSERTION LIMITS." Following their review of the amendment application, the NRC identified several editorial changes needed to the Technical Specification pages submitted with the amendment application. The revised pages are included as an enclosure to this letter.

BACKGROUND

By Amendment Application No. 205, we requested a change to the end-of-cycle Moderator Temperature Coefficient (MTC) limit in Technical Specification 3.9. To accommodate the revision to the MTC value, changes were also requested to the Technical Specification limit for shutdown margin in Specification 3.5.2. The amendment application included all pages (3.5-6 through 3.5-9) of existing Specification 3.5.2, although the proposed changes only affected pages 3.5-7 and 3.5-8.

During their review of the amendment application, the NRC found an editorial error on the existing Technical Specification page 3.5-6. In addition, it was noted that Specification 3.5.2 listed several references that were not specifically called out in the text, and that one of the references in the list was not current. Technical Specification pages 3.5-6 through 3.5-8 have been revised to address these concerns. The necessary changes are described in detail in the following sections.

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CHANGE TO TECHNICAL SPECIFICATION PAGE 3.5-6

Existing Technical Specification page 3.5-6 contains an editorial error. Specifically, the word "power" is missing from the phrase "maximum specific power" in the last sentence on this page. This word is needed for accuracy of the sentence which describes the considerations in setting the control group insertion limits. The "maximum specific power" is intended to be one of the considerations.

Page 3.5-6 was previously issued by the NRC on August 21, 1989 as part of Amendment No. 130, and apparently, the word "power" was inadvertently omitted when this page was issued. The corrected page 3.5-6 is included here as part of the enclosure with a change bar denoting the addition of the word "power."

CHANGES TO REFERENCES

Proposed Specification 3.5.2 in the amendment application contains a list of 10 references, although only references 8 and 9 are specifically called out in the text. References 1 through 7 are general in nature, and are not intended to support any specific statements in Specification 3.5.2. Current information supporting plant design related to the Technical Specifications is included in the Updated Final Safety Analysis Report, and supersedes these references. Therefore, these references will be deleted from the list in accordance with the NRC Staff request, and references 8 through 10 will be renumerated as references 1 through 3. Reference 10 will also be specifically called out in the text since this reference contains the basis for the change to the shutdown margin limit requested by Amendment Application No. 205.

Reference No. 10 is Westinghouse letter no. SCE-92-518, dated April 6, 1992, which contains results of Main Steam Line Break (MSLB) reanalysis performed by Westinghouse to support Amendment Application No. 205. This letter also contains 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). Both of these versions of the WCAP reports were submitted to the NRC by a letter dated May 18, 1992. These WCAP reports supersede the April 6, 1992 Westinghouse letter referenced in Specification 3.5.2. Therefore, the reference to the Westinghouse letter will be replaced by reference to the WCAP reports. These reports will be designated as reference no. 3 in the revised Technical Specification page 3.5-8.

IMPACT ON PREVIOUS SAFETY ANALYSIS

The changes discussed in this letter are entirely editorial in nature and do not affect the safety analysis submitted in the referenced amendment application. The updated pages are enclosed to this letter.

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

Very truly yours,

RM 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

3.5.2 CONTROL ROD INSERTION LIMITS

APPLICABILITY: MODES 1 and 2

OBJECTIVE: This specification defines the insertion limits for the control rods in order to ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

SPECIFICATION:

A. Except during low power physics tests or surveillance testing pursuant to Specification 4.1.1.G, the Shutdown Groups and Control Group 1 shall be fully withdrawn, and the position of Control Group 2 shall be at or above the 21-step uncertainty limit shown in Figure 3.5.2.1.

B. The energy weighted average of the positions of Control Group 2 shall be at least 90% (i.e. > Step 288) withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all Control Group 2 positions during the core cycle.

ACTION:

A. With the control groups inserted beyond the above insertion limits either:

1. Restore the control groups to within the limits within 2 hours, or

2. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

3. Be in at least HOT STANDBY within 6 hours.

B. With a single dropped rod from a shutdown group or control group, the provisions of Action A are not applicable, and retrieval shall be performed without increasing THERMAL POWER beyond the THERMAL POWER level prior to dropping the rod. An evaluation of the effect of the dropped rod shall be made to establish permissible THERMAL POWER levels for continued operation. If retrieval is not successful within 3 hours from the time the rod was dropped, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% Δ k/k.

BASIS: During STARTUP and POWER OPERATION, the shutdown groups and control group 1 are fully withdrawn and control of the reactor is maintained by control group 2. The control group insertion limits are set in consideration of maximum specific power,

shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_\Delta H$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design value of specific power, $F^N_\Delta H$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_\Delta H$ and F_Q are 13.2 kW/ft, 1.57 and 2.78, respectively.(1) At partial power, the $F^N_\Delta H$ maximum values (limits) increase according to the following equation, $F^N_\Delta H\ (P) = 1.57\ [1 + 0.2\ (1-P)]$, where P is the fraction of RATED THERMAL POWER. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

2. The minimum shutdown capability required is 1.25% Δp at BOL, 2.05% Δp at EOL (2) and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available SHUTDOWN MARGIN is greater than the above values.

3. The worst case ejected rod accident (3) covering HFP-BOL, HZP-BOL, HFP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 220 cal/gm for irradiated fuel.

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

   Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% RATED THERMAL POWER. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed.

   It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available SHUTDOWN MARGIN.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

The presence of a dropped rod leads to abnormal power distribution in the core. The location of the rod and its worth in reactivity determines its effect on the temperatures of nearby fuel. Under certain conditions, continued operation could result in fuel damage, and it is the intent of ACTION B to avoid such damage.

References: (1) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989.

(2) Main Steamline Break Analysis for Revised Moderator Density Coefficients, WCAP-13346 (Westinghouse Proprietary Class 2) and WCAP-13347 (Westinghouse Proprietary Class 3), April 1992.

(3) An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January, 1975.