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MIRAGLIA,F. Licensing Branch 3

SUBJECT: Forwards responses to seven confirmatory SER items requested by Core Performance Branch, Instrumentation & Control Sys Branch & Auxiliary Sys Branch.

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add: Core Performance Br - 1cy
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4LAJ

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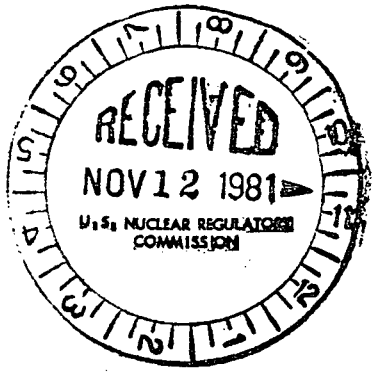
November 10, 1981



Director, Office of Nuclear Reactor Regulation
Attention: Mr. Frank Miraglia, Branch Chief
Licensing Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-361
San Onofre Nuclear Generating Station
Units 2 and 3

During the week of October 26, 1981 the NRC requested additional information in order to resolve seven confirmatory items. The information was requested by the NRC CPB, ICSB and ASB and was informally telecopied to SCE. Consistent with the NRC request, enclosed please find seven (7) copies of the responses to the seven confirmatory items.

If you have any questions or comments concerning this information please contact me.

Very truly yours,

K P Baskin

Enclosures

B001
S/1

Add: Core Performance Br - 1cy
Instrumentation & Control Br - 1cy
Auxiliary Sys Br - 1cy

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CPB ITEMS
(Items 1 and 2)

ITEM 1

Revised Analysis of Reactivity Initiated AOO for San Onofre Units 2 and 3

Amendment 24 to San Onofre Units 2 and 3 was submitted in April 1981. Included in this Amendment was a revised analysis of the part-length CEA subgroup drop event which is considered an anticipated operational occurrence. I do not know the reason for this revised analysis, particularly since the SONGS 2 and 3 SER and Supplement 1 have already been issued and this event was not an open item. The following are some specific questions which I believe should be addressed by Southern California Edison Company:

1. Why has the part-length CEA subgroup drop event been reanalyzed?

2. Why has the time to low DNBR trip signal for this event increased from 0.6 to 3.3 seconds?

3. Why were the moderator temperature coefficients for the part-length CEA subgroup drop cases assuming dropped CEA reactivity worths of +.04 percent and +.031 percent changed from the most positive values to the most negative values? Does not the most positive DNR result in the maximum core power level for positive initial reactivity insertions?

4. Should the maximum core power given in Table 15.4-15 be 108.2 percent instead of 113.2 percent?

5. Since the most rapid approach to the DNB SAFDL can be caused by the part-length CEA subgroup drop, what effect does the reanalysis have on Technical Specifications in regard to core thermal margins?

RESPONSE

The revised analysis of Reactivity initiated AOO for San Onofre Units 2 and 3 was reanalyzed to accommodate a longer allowable trip for the part length CEA Subgroup Drop. SCE is preparing responses to the above concerns which will be transmitted to the NRC by November 16, 1981.

ITEM 2

Loose Parts Monitoring System

As addressed in the SER, the applicant's loose parts monitoring system (LPMS) had been reviewed and found acceptable with regard to its sensor locations, sensor sensitivity, alarm setting and its functional capability for a seismic event, etc. However, since the issuance of the SER, the Regulatory Guide 1.133, "Loose Part Detection Program for the Primary System of Light Water Cooled Reactors," has been isused. The Regulatory Guide also specifies that a Technical Specification for the LPMS should be provided to specify the limiting conditions for operation and surveillance requirement.

In addition, each licensee is required to provide a report describing the training program for plant personnel that addresses operation of the system hardware and implementation of the loose part detection program. We have informed the applicant of these requirements. The applicant has committed to submit a report regarding the personnel training program and LPM system. With the inclusion of the Loose Part Detection instrumentation in the Technical Specification, we conclude that an acceptable LPMS and program will be implemented for SONGS Units 2 and 3.

RESPONSE

SCE is developing the training program relative to the operation of the Loose Part Monitoring System. The report outlining the training program will be available for NRC inspection 6 months prior to commercial operation of San Onofre Unit 2.

Technical Specification 3.3.3.10 provides the LCO and surveillance requirements for the loose part detection instrumentation.

ICSB ITEMS
(Items 3,4,5 and 6)

ITEM 3

Seismic Scram

Confirmatory Items: The applicant has agreed to provide the following formal documentation prior to fuel load:

1. A revision to the FSAR describing the seismic reactor trip and stating that the design complies with IEEE-279.

2. Drawings of the seismic trip instrumentation having the same level of detail as those instrumentation and control drawings currently referenced in Chapter 1.7 of the FSAR.

RESPONSE

The FSAR is being revised to indicate that the seismic reactor trip design complies with applicable portions IEEE-279-1971. Section 7.2.2.3.2 of the FSAR is being revised to describe the status of compliance. This is being incorporated in the San Onofre Units 2 and 3 FSAR as part of Amendement No. 27 which is forthcoming.

Consistent with the NRC's request, elementary drawings to the same level of detail as normally provided in Table 1-7 of the FSAR will be submitted to the NRC prior to installation of the seismic trip system.

ITEM 4

I&E Bulletin 80-06

Confirmatory Items: The applicant has committed to provide formal documentation prior to fuel loading to confirm the information provided to the staff through discussions. This information should confirm that:

1. Valve position indication exists in the control room for each valve affected by the safety signal reset,

2. Each valve affected by reset has the capability to be manually controlled from the control room,

3. The emergency procedures will contain caution statements to alter the operation that these components will change state upon reset of the associated USFAS, and

4. Testing of the installed instrumentaton and any necessary corrective action(s) will be completed prior to operation above 5% during plant start-up.

RESPONSE

1. The SIT interlock setpoint is 500 psia rather than 700 psia as conveyed previously.

2. Each valve identified in Table 222.42-1 has control and position indication in the control room.

3. The appropriate caution statements will be added to the emergency procedures prior to fuel load to alert operators to expect a change of position upon safety signal reset for the valves listed in Table 222.42-1.

4. Testing and any corrective action shall be completed prior to fuel load. (Amendment No. 27 which is forthcoming, revises page Q&R 15.0-19 of the San Onofre Units 2&3 FSAR to include this commitment).

ITEM 5

I&E Bulletin 79-02

There are two areas of unacceptable interaction with Non-Class IE components. The applicant has committed to modify the charging pump controls and the pressurizer Class IE heater controls to ensure that the charging pumps and the Class IE pressurizer heaters can be operated to reach cold shutdown.

The applicant states that Non-Class IE power supply inverters are of the type referenced in IE Circular 79-02. It was stated that, to date, some difficulty has been encountered due to spurious transfers of the static switch during voltage transients. The applicant has committed to take corrective action to either increase the transfer setpoints or replace the switch. This action will be taken prior to Fuel Load.

We conclude that, with the modifications described above, sufficient equipment for safe shutdown would remain available subsequent to loss of any Class IE or Non-Class IE electrical bus and that the applicant has adequate emergency procedures to deal with the resulting plant conditions. The license will be conditioned to require completion of the modifications prior to fuel load.

Confirmatory Item: The applicant has committed to formally confirm that the above modifications have been made and to indicate the final resolution of the problem with the spurious transfers of the static switch prior to fuel loading.

RESPONSE

The modifications to the control switches for the chagring pumps and for the pressurizer heaters as described in the response to NRC question 222.41 have been completed.

Spurious transfers experienced on the SCI power supply inverters for the non-safety related computer bus were resolved by increasing the transfer switch setpoint to 150%. This was accomplished in September, 1979 and no further problems have been experienced.

ITEM 6

II.F.2 Instrumentation for Inadequate Core Cooling

Review of the information provided by the applicant in Amendment 25 of the FSAR indicates that the hardware design of the instrumentation for detection of the inadequate core cooling to be provided during the first refueling period of Unit 2 and prior to initial fuel load in Unit 3 adequately complies with the guidance of NUREG-0737, "Clarification of TMI Action Plan Requirements".

Confirmatory Items: The applicant has agreed to provide the following formal documentation.

1. An FSAR change prior to January 1, 1982, indicating that the verification and validation of the software associated with the Qualified Safety Parameter Display System will be under a quality assurance plan that will include a test facility, integrated software testing, and static and dynamic test which cycles the software through design transients. The quality assurance plan will be available for staff audit.

2. Drawings at least six months prior to the first refueling on Unit 2 having the same level of detail as those instrumentation and control drawings currently referenced in Chapter 1.7 of the FSAR.

RESPONSE

1. Verification and validation of the QSPDS software for the ICC display will be performed which includes use of a designated test facility, integrated software tesing and static and dynamic tests which throughly test the software. The QSPDS verification testing procedures will be developed utilizing the experience gained from previous CPCS software verification.

2. The ICC electrical drawings to the same level of detail as normally provided in Table 1-7 of the FSAR will be submitted 6 months prior to the first refueling of San Onofre Unit 2.

ASB ITEM
(Item No. 7)

ITEM 7

FEEDWATER HAMMER TESTING

We conclude that the proposed test of automatic AFW initiation is acceptable subject to the following: conformatory formal submittal of the proposed test procedure; clarification of the method of test initiation; commitments to inspect the entire feedwater line prior to going to an appreciate power level (more than 25%) or providing adequate instrumentation with remote readout capability to detect pipe and support movements for the portion of the line inside containment; close monitoring of feedwater flow and primary hot and cold leg temperatures for any anomalies that indicate potential feedwater ring damage. We will require that a similar test be performed for Unit 3.

RESPONSE

In order to resolve this item SCE (Mr. F. R. Nandy) contacted the NRC (M. B. Mann) to discuss the clarification necessary on October 28, 1981. Consistent with the discussion an acceptable test abstract was prepared and is enclosed.

S/G FEEDRING/AUXILIARY FEED PIPING INTERGRITY TEST*

OBJECTIVE

To prove the adequacy of the steam generator feedring to withstand the introduction of auxiliary feedwater following exposure to a steam environment.

PREREQUISITES

a) Auxiliary Feedwater system is available.

b) Main steam system is available.

c) Appropriate ac and dc power sources are available.

TEST METHOD

a) Lower the generator water level to below the feedwater sparger without feedwater flow.

b) Initiate, or simulate initiation of, automatic auxiliary feedwater flow to reflood the steam generator by starting both the motor driven and turbine-driven pumps at EFAS flow rates.

ACCEPTANCE CRITERIA

a) No significant noise or vibrations are observed during test.

b) Visual inspection indicates that the intergrity of feed piping, pipe supports and feedring have not been violated.**

**Visual inspection of feedwater piping and supports will occur at a convenient time during the test program prior to going above 25% reactor power. Visual inspection of the feedring will occur at first cold shutdown.

* This test will occur prior to going above 25% reactor power.