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SUBJECT: Responds to 840827 request for addl info re response to NRC Question 222.43 concerning high energy line break effects on control sys.W/2 oversize drawings.Aperture cards available in PDR.

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December 17, 1984

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

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

References: A. SCE to NRC letter dated April 1, 1983
B. SCE to NRC letter dated September 15, 1983
C. SCE to NRC letter dated November 30, 1983

By letter dated August 27, 1984, SCE received an NRC request for additional information regarding SCE's response to NRC question 222.43, high energy line break (HELB) effects on control systems. The August 27, 1984 NRC request for additional information consisted of two basic concerns: A) steam bypass control system and reactor regulating system simultaneous malfunctions and B) the use of best estimate instead of FSAR calculational uncertainties in the HELB Analysis.

Included as Enclosure I to this letter is SCE's response to Part A of the NRC request. Enclosure II provides SCE's response to Part B of the NRC request. It should be noted, as indicated in Enclosure II, that SCE did not make use of best estimate calculational methods in the HELB Analysis. Only the FSAR calculational uncertainties that cover the postulated control systems malfunctions were removed from the HELB Analysis calculations. This is consistent with the FSAR Chapter 15 calculational methods.

If you have any questions concerning the enclosed information, please contact me.

Very truly yours,

M. O. Medford

cc: Harry Rood (to be opened by addressee only)
F. R. Huey, NRC Resident Inspector

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ENCLOSURE I

Response to Part "A" of the NRC Request for Additional Information Regarding the Effects of High Energy Line Breaks on Control Systems

QUESTION:

It is stated in the letter from M. Medford (SCE) to G. Knighton (NRC) dated September 15, 1983, that the steam bypass control system (SBCS) and reactor regulating system (RRS) malfunctions cannot occur simultaneously during a steam line break (SLB) event due to the presence of the Automatic Withdrawal Prohibit (AWP) signal. Upon review of documentation supplied to date, including the FSAR, the staff has concluded that insufficient information has been provided to describe the interaction (interlock) associated with the SBCS, the RRS, and the control element drive mechanism control system (CEDMCS). Therfore, please provide detailed information to describe the subject interface design which has been implemented to prevent simultaneous operation of the worst-case failures (those that exacerbate event consequences) for the SBCS and the RRS during a single HELB event. As a minimum -

1. Confirm that a single SLB event (one that could affect the input parameter channels to both the SBCS and RRS) will not also cause failure of the subject SBCS/RRS interlocks required to prevent their simultaneous worst-case malfunction.

2. Provide detailed elementary drawings and electrical schematics to show the interaction of the RRS and SBCS output signals with the CEDMCS. This should clearly show the interface of the SBCS AWP signal with CEDMCS which is to be used to block the RRS demand for withdrawal of control element assemblies (CEAs). The drawings should be highlighted and/or annotated as necessary for clarity.

3. Provide information to verify that should the RRS demand for withdrawal signal exist, a subsequent SBCS AWP signal will block the RRS demand signal and will result in the discontinuation of CEA withdrawal. Drawings to be provided as part of item 2 above should clearly show this. Again, please highlight and/or annotate where necessary.

The staff also recommends that the FSAR should be revised to describe the interaction of the SBCS with the CEDMCS and its blocking function associated with the RRS outputs to the CEDMCS.

RESPONSE:

1. A single SLB event (one that could affect the input parameter channels to both the SBCS and the RRS) will not also cause a failure of the SBCS/RRS interlock. The SBCS cabinet is located in the control area of the auxiliary building. The CEDMCS cabinet is located in the radioactive waste area of the auxiliary building. The electrical cables that transmit the AWP signal from the SBCS to the CEDMCS are located in the auxiliary building. There are no steam lines in the auxiliary building. Hence, a single SLB event cannot prevent the SBCS from generating an AWP signal when required. A SLB event cannot fail the cables that transmit the AWP signal from the SBCS to the CEDMCS, or fail the SBCS/RRS interlock in the CEDMCS.

2. An AWP signal from the SBCS will block a CEA withdrawal demand signal from the RRS, or terminate an automatic withdrawal if the withdrawal is already in progress. Enclosure (A) (Combustion Engineering drawing D-ICE-414-472 Rev. 01, which is applicable to SONGS Units 2 and 3) is a functional logic drawing that shows the interaction of the RRS and SBCS output signals with the CEDMCS. The override by the AWP signal of the RRS request for withdrawal of the CEAs occurs at the NAND gate labeled "T" on Attachment (A). Gate "T" on Attachment (A) corresponds to the gates circled on Attachment (B) (Combustion Engineering drawing 6022-35025, which is applicable to SONGS Units 2 & 3) which is an electrical schematic that shows that interaction of the RRS and SBCS output signals with the CEDMCS.

The SBCS generates an AWP signal (input "I" on Attachment (A)) whenever an automatic bypass valve opening demand signal exists. The RRS may generate an automatic raise (AR) signal (input "F" on Attachment (A)) due to normal or faulted inputs. The RRS generates other signals which do not impact this discussion since they do not result in the uncontrolled withdrawal of the CEAs. The CEDMCS also receives a manual raise (MR) signal (input "J" on Attachment (A)) when the operator actuates a CEA withdrawal. Input "J" is zero for this discussion since no operator action is assumed. Enclosure (A) also shows other signals input to the CEDMCS. These signals do not, however, impact the SBCS/RRS interlock.

If an AWP is generated and the RRS is requesting a CEA withdrawal, then the inputs to the CEDMCS will have the following values (note that inputs that inhibit motion of the CEAs are "0"): AR = "1", AWP = "0", and MR = "0" (no operator action). Tracing these inputs through Attachment (A) yields the following conclusions:

a) Since the AWP signal, input "I", is "0", the output of the NAND gate "T" will be "1", regardless of the values of the other inputs to gate "T".

b) Since the MR signal, input "J" is "0", the output of the NAND gate "V" will be "1", regardless of the values of the other inputs to gate "V".

c) The output of gate "T" is inverted to "0" by the inverter "U". The output of gate "V" is inverted to "0" by the inverter "W". The outputs of these two inverters are input at the Exclusive OR gate "X". The output of gate "X" will thus be "0".

d) The "0" output of gate "X" is input to the NAND gate "Y". Hence, the output of gate "Y" will be "1" regardless of the values of the other inputs to gate "Y".

e) The outputs of the CEDMCS logic for the "control group raise" portion of the diagram (amplifier/inverters "Z") are "0" (i.e., the demand for withdrawal of the CEAs is blocked).

3. The above response (2) applies to a CEA withdrawal in progress when an AWP signal is generated. The logic is not latched. When signal "I" is zero, the automatic withdrawal will be terminated.

ENCLOSURE II

Response to Part "B" of the NRC Request
for Additional Information Regarding the Effects
of High Energy Line Breaks on Control Systems

QUESTION:

The staff has reviewed the applicant's responses to NRC Questions 222.43 and 222.44. Respectively, these questions requested information on 1) high energy line break (HELB) effects on control systems and 2) the failure of any power sources, sensors, or sensor impulse lines which provide power or signals to two or more control systems. Further clarification is necessary to resolve the HELB issue (item 1 above).

It is stated in the letter from M. Medford (SCE) to G. Knighton (NRC) dated November 30, 1983 that the calculational uncertainties placed on the bounding moderator temperature coeficient (MTC), bounding Doppler Coefficient, and bounding CEA worth used in the FSAR analysis are not included in the steam line break (SLB) analyses performed for the HELB study. The MTC, Doppler Coefficient, and CEA worth used in the HELB analysis are best-estimate data. The staff considered the removal of the FSAR calculational uncertainties from the conservatisms used in the HELB analysis to be unacceptable. It is the staff's position that the HELB analysis should be performed using the same conservative assumptions as used in the FSAR analyses. Thus, the staff requests that the consequences of multiple and single control system failures which could result from each postulated HELB event be reanalyzed where necessary based on the staff's position stated above or provided additional information to support the current HELB analyses.

RESPONSE:

The HELB evaluations of the SLB with failures of control systems were conservative using input parameters, assumptions, and models that were conservative with respect to the expected values of the input parameters and behavior of the plant. They were not evaluations using best estimate data. They used the same input parameters, assumptions, and models as the FSAR Chapter 15 analysis with some of the conservatisms removed from the MTC, the Doppler coefficient, the CEA scram worth, and the response of the safety injection system.

The purpose of the HELB evaluations performed in response to NRC Question 222.43 was to demonstrate that the licensing analysis presented in the FSAR for the HELB events included adequate margin to cover the postulated control system failures. The main concern for the events in question (SLBs with failures of the main feedwater control system (MFWCS) and/or the steam bypass control system (SBCS)) is post reactor trip return-to-power (RTP). The SLB analysis presented in Chapter 15 of the FSAR include conservatism in the input parameters to which the SLB post reactor trip RTP consequences are sensitive. These conservatisms include but are not limited to those parameters and assumptions summarized in Table 1. In Table 1, the "Nominal Values" are those values of the input parameters that would be typical of normal plant operation.The "FSAR Chapter 15 Values" are those values of the input parameters that were conservatively selected for the SLB post reactor trip RTP analysis presented the FSAR. The conservatisms were included, in part, in the Chapter 15 analysis to provide margin to cover scenarios that complicate the event. The "HELBA Values" are those values used for the SLB with malfunctions of the MFWCS and/or the SBCS evaluations. The thermal-hydraulic models used for the SLB analysis also include many conservatisms for the SLB post reactor trip RTP consequences which were also utilized in the HELBA evaluations. These conservatisms include but are not limited to those summarized in Table 2.

The HELB evaluations could have taken credit for the conservatisms in many input parameters and modeling assumptions to demonstrate that the licensing analysis presented in the FSAR includes adequate margin to cover the postulated control system malfunctions. However, the only conservatisms removed for the HELB evaluations from the Chapter 15 SLB analysis are in the MTC, the Doppler coefficient, the number of high pressure safety injection (HPSI) pumps operating, and the CEA scram worth. The values of these parameters used for the HELB with malfunctions of control systems evaluations are not best estimate values. They are conservative with respect to the nominal values.

Hence, the HELB evaluations used input parameters, assumptions, and models that are conservative. The HELB evaluations show that the FSAR safety analyses include adequate margin to cover the postulated malfunctions of control systems.

TABLE 1

INPUT ASSUMPTIONS/PARAMETERS TO WHICH THE SLB RETURN-TO-POWER CONSEQUENCES ARE SENSITIVE

Assumption/ Parameter	Nominal Value	FSAR Chapter 15 Value	HELBA Value
1. Core Burnup	BOC to EOC	EOC	EOC
2. Doppler Multipler	< 1.00	1.15	1.00
3. MTC Multiplier	< 1.00	1.10	1.00
4. CEA Scram Worth (%$\Delta\rho$)	11.3 (no stuck CEA)	8.55 (stuck CEA)	9.41 (stuck CEA)
5. Break Size (Ft^2)	-	7.41 (most adverse)	7.41 (most adverse)
6. Safety Injection System	2 HPSI pumps, nominal flow	1 HPSI pump, minimum required flow	2 HPSI pumps, minimum required flow
7. ESFS Delay Times			
a. MSIV Closure Time (sec)	<5.0	5.0	5.0
b. Time from SIAS Generated Until SI Pumps Reach Full Speed (sec)	<11.0	11.0	11.0
8. ESFS Setpoints			
a. SIAS, low pressurizer pressure (psia)	1763	1560	1560
b. MSIS, low S.G. pressure (psia)	711	675	675
9. Initial RCS Pressure (psia)	2250	2000	2000
10. Inital Core Inlet Temperature (^oF)	553	560	560
11. Inital Core Mass Flow Rate (lbm/hr x 10^6)	143.0	132.2	132.2

TABLE 1

(CONTINUED)

Assumption/ Parameter	Nominal Value	FSAR Chapter 15 Value	HELBA Value
12. Initial Core Power (MWT)	3410	3478	3478
13. Boron Worth, (ppm/% $\Delta\rho$)	79 at 538^{o}F 62 at 68^{o}F	88 (assumed constant at all conditions)	88 (assumed constant at all conditions)

TABLE 2

MODELING ASSUMPTION TO WHICH THE SLB
RETURN-TO-POWER CONSEQUENCES ARE SENSITIVE

1. No credit is assumed for moisture carry over during the steam generator blowdown.
2. The moderator reactivity contribution to the core power is based on the core cold edge temperature.
3. The model used for determining the reactivity insertion due to borated safety injection flow assumes the maximum time delay for the safety injection pumps to reach full speed, unborated water must be swept from the safety injection lines before borated water enter the cold legs.
4. No reduction in heat transfer by the steam generators from the primary to the secondary is assumed until the liquid mass on the secondary side decreases to zero.
5. Frictionless choked steam flow from both steam generators through the break is assumed. The pressure used to calculate the steam flow rate is conservatively assumed to be that of the steam generator shell side pressure.