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SUBJECT: Forwards responses to 830401 & 20 submittals to NRC Questions 222.43, "Qualficiation of Control Sys" & 222.44, "Control sys Failure."

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November 30, 1983

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

SCE's letters dated April 1 and 20, 1983 transmitted responses to NRC Questions 222.43 "Qualification of Control Systems" and 222.44 "Control System Failures." Subsequently on June 30, 1983 a request for clarification was received by SCE to verify that the following considerations were included in the High Energy Line Break (HELB) analysis:

1. Multiple (as well as individual) control systems failure or malfunctions caused by the identified HELB.

2. Single failure of any safety related system used to mitigate the consequences of HELB.

Additionally, SCE was requested to provide information regarding the effect of pressurizer pressure signal on the Steam Bypass Control System and the Reactor Regulating System. The responses to these questions were transmitted to the NRC by letter dated September 15, 1983.

On September 29, 1983 SCE received a request for further clarification to verify that the following considerations were included in the HELB analysis:

A) Verify that simultaneous malfunctions of all impacted control systems had been covered. Also identify the specific FSAR evaluations which bound each HELB/Control System Interaction.

B) Identify any HELB analysis that is not bounded by the FSAR analysis, and provide information concerning the assumptions, input parameters and results of such an analysis.

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Enclosed please find seven (7) copies of the responses to the items above relative to the April 1 and 20, 1983 submittals to the NRC. The clarification provided by these responses serves to substantiate SCE's position that any impact of a HELB/Control Systems interaction due to either single or multiple control system failure is acceptable.

If you have any questions or comments concerning the enclosed information, please let me know.

Very truly yours,

M. O. Medford

cc: Harry Rood, NRC (To be opened by addressee only)
A. L. Chaffee (NRC Site Inspector)

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

QUALIFICATION OF CONTROL SYSTEMS
INFORMAL NRC REQUEST FOR ADDITIONAL INFORMATION

NRC REQUEST (Paraphrased)

(A) An evaluation of any HELB which may induce more than one control system malfunction should consider simultaneous malfunctions of all impacted control systems (whether or not the individual impacts are considered significant) to assure that the consequences of the combined malfunctions are acceptable.

Verify that simultaneous malfunctions of all impacted control systems has been covered. Also identify the specific FSAR evaluations which bound each HELB/Control System Interaction.

(B) For any HELB/control system interactions which were quantitatively reanalyzed, provide information concerning the assumptions, input parameters and results of such an analysis, relative to the bounding FSAR analysis.

RESPONSE:

(A) The analysis of high energy line break (HELB) interactions with control systems performed for San Onofre Nuclear Generating Stations 2 & 3 (SONGS 2 & 3) evaluated the impact of simultaneous malfunctions in all control systems affected by a common HELB and determined that the HELB consequences are bounded by those presented in Chapter 15 of the SONGS FSAR. A summary of the HELB/control system interactions is provided in Table 1. Each HELB/control system interaction is placed into one of the following categories: 1) the interaction is not possible due to physical location of the break and the control system; 2) the interaction results in control system behavior which is bounded by the control system disposition assumed in the FSAR analyses; or 3) the interaction results in control system behavior which may not be bounded by the actions assumed in the FSAR analyses, and therefore, requires further investigation.

The first category of HELB/control system interaction is easily removed from concern, both as an individual failure and as a multiple failure, since no interaction is possible. The interactions removed from concern in this manner are indicated as NI (no interaction) in Table 1 (e.g., failure of the PPCS during a RCSBOC).

The HELB/control system interactions which result in control system actions bounded by the control system disposition assumed in the FSAR analyses are of no concern individually. The interactions removed from concern in this manner are indicated as NMA (no more adverse) in Table 1. An example of this is the failure of the SBCS during a LOCA. In the FSAR analysis the SBCS is assumed to be in the manual mode of operation, and not to enhance RCS energy removal. If the SBCS were to fail during a LOCA it could either fail closed and remove no additional RCS heat, as assumed in the FSAR, or it could fail open and enhance RCS heat removal by improving the operation of the ECCS. It is easily concluded that the failure of the SBCS during a LOCA results in HELB consequences no more adverse than those presented in the FSAR. Any interaction which was determined to be NMA was included in the multiple failure analysis and the control system was assumed to be in the mode of operation postulated in the FSAR analyses.

The HELB/control system interactions which could result in control system behavior not bounded by the control system disposition assumed in the FSAR analysis are indicated as X in Table 1. These interactions were analyzed both individually and in combination with each other. The results of the individual analyses and a summary of the multiple failure analyses are presented in the SCE Response to NRC FSAR Question 222.43. A detailed discussion of the multiple failure analyses and the conclusions drawn from them is presented below.

Loss of Coolant Accident (LOCA) Evaluations

The combined impact of the PLCS, PPCS, SBCS, RRS, SGBS, RCP, MFWCS, and TGCS malfunctions during a LOCA were investigated. This failure combination results in an increase in peak clad temperature (PCT) of less than 163°F. This increase in PCT is small in comparison to the established PCT margin of 688°F [discussed in (B) 1.1 of this response], and is, therefore, bounded by the LOCA analyses presented in Chapter 15 of the SONGS FSAR (Section 15.6.3.3). Stable long term cooling operation following the event will be achieved, since the malfunctioning of the control systems listed above will not significantly increase the required condensate volume (<3500 gal increase), or delay entry into shutdown cooling beyond that presented in FSAR subsection 15.6.3.3. In addition, it was determined that the failure of these control systems would not cause further violation of the RCS pressure boundary integrity.

Steam Line Break (SLB) Evaluations

The failures of the PLCS, PPCS, SBCS, RRS, SGBS, MFWCS, and TGCS were evaluated as multiple failures with respect to each criterion (fuel performance, return-to-power). The results of the analysis demonstrated that the fuel performance and return to power consequences are bounded by the SLB event presented in subsection 15.1.3.1 of the FSAR. In addition, it was determined that the failure of these control systems would not violate the RCS pressure boundary integrity.

Feedwater Line Break (FWLB) Evaluations

The result of multiple failures of the PLCS, PPCS, SBCS, RRS, SGBS, MFWCS, and TGCS is an increase in the peak RCS pressure by <3 psi. This impact is negligible in comparison to the margin inherent in the conservatisms assumed in the FWLB analysis presented in subsection 15.2.3.1 of the FSAR.

Reactor Coolant System Break Outside Containment (RCSBOC) Evaluations

The combined failures of the SBCS, SGBS, RCP, and MFWCS during a RCSBOC were considered and resulted in event consequences which are bounded by the RCSBOC event presented in subsection 15.6.3.1 of the FSAR. Note that concurrent failures of the SGBS and MFWCS produce the same effect, that being a depletion of the steam generator inventory. Only a gradual depletion of inventory is of concern during a RCSBOC. Therefore, combining both system malfunctions produces an impact no more severe than if only one system malfunctioned.

The specific conclusions drawn from the multiple interaction analysis are:

1. The impact on peak clad temperature, entry into shutdown cooling, and RCS pressure boundary integrity for a LOCA with any combination of possible control system malfunctions is bounded by the LOCA analyses presented in subsection 15.6.3.3 of the FSAR,

2. The impact on pre-trip fuel performance, post-trip return to power, and RCS pressure boundary integrity for a SLB with any combination of possible control system malfunctions is bounded by the SLB analyses presented in subsection 15.1.3.1 of the FSAR,

3. The impact on peak RCS pressure for a FWLB with any combination of possible control system malfunctions is bounded by the FWLB analyses presented in subsection 15.2.3.1 of the FSAR, and

4. The impact on radiological releases for a RCSBOC with any combination of possible control system malfunctions is bounded by the RCSBOC presented in subsection 15.6.3.1 of the FSAR.

(B) All HELB/control system interaction combinations identified in Table 1 are bounded by the results of the FSAR analyses. The margin of conservatism available in the FSAR analysis was found to be greater than the worst case degradation in performance caused by the failed control systems. Information concerning the assumptions, input parameters and results of these margin analyses are provided below.

1. <u>Loss-of-Coolant Accident Analysis</u>
As identified in the SCE Response to the NRC request for additional information on HELB evaluations, only small break LOCAs are impacted by control system malfunctions. For small break LOCAs, the PCT margin was defined as the difference between the FSAR Chapter 15 analysis results and the results of a best estimate (BE) analysis for the limiting small break LOCA. For post-LOCA long term cooling, the margin was established as the difference between the seismically qualified condensate storage tank (CST) inventory available and the inventory required in the FSAR Chapter 15 analysis. The following calculations were performed to determine the margin in both cases.

1.1 Evaluation of Peak Clad Temperature

The limiting small break LOCA (0.05 ft^2) was analyzed with best estimate input and assumptions to establish the base case. The significant features of this analysis which differ from the licensing analysis include:

(a) Condensation of steam in the RCS cold legs due to subcooled safety injection (non-equilibrium effect).

(b) Improved hot rod heat transfer calculations which account for radiation to steam as well as convective steam cooling.

(c) 1.0 x 1971 ANS Decay Heat

The analysis assumed loss of off-site power on a reactor trip, which was generated by low pressurizer pressure (LPP). The analysis also assumed the worst single failure in the ECCS system, which is the failure of a diesel generator to start. Therefore, the analysis took credit for only one high pressure safety injection pump. Table 2 lists values of the significant input parameters and compares them with the values used in the FSAR Chapter 15 analysis.

Table 2 lists significant results of the analysis. Reactor trip and SIAS for the BE case is generated at 25 seconds as opposed to at 105 seconds in the FSAR case. The maximum core uncovery is reduced from 3.8 ft (in the FSAR case) to 1.7 ft. The peak clad temperature (PCT) calculated is 1044°F, which is 688°F less than the PCT calculated for the FSAR analysis. This difference of 688°F in PCT is established as the margin available to accommodate the impact of control system failures.

1.2 Delayed Entry Into Shutdown Cooling

In the long-term following a small break LOCA, stable long term decay heat removal is provided by RCS cooldown followed by use of the shutdown cooling system.

For the FSAR Chapter 15 analysis, feedwater inventory needed to cool the RCS below the shutdown cooling (SDC) entry temperature (<400°F) is less than 153,000 gallons. This value is calculated conservatively as the analysis assumes that only one SG is available for the cooldown. The combined consequences of the assumed control system malfunctions was determined to require at most an additional 3,500 gallons of condensate. The total seismically qualified condensate storage tank inventory available is approximately 350,000 gallons (see FSAR subsection 10.4.9.3). Therefore, the margin available to accommodate control system failures is much greater than required.

2. <u>Steam Line Break</u>

Both the steam line break (SLB) event presented in subsection 15.1.3.1 of the SONGS FSAR, and the SLB analysis performed for the HELB study meet the requirements of Standard Review Plan (SRP) 15.1.5. Both analyses assumed a stuck control element assembly (CEA), initial condition which maximizes event consequences, a single failure of an active component, and fuel conditions which yield the most limiting moderator temperature and Doppler reactivity coefficients. The SLB analyzed for the HELB/control system interaction analysis assumed the same stuck CEA worth and initial conditions as the SLB presented in subsection 15.1.5.1 of the FSAR. The HELB analysis also assumed a single failure in an active component. The failure of a main steam isolation valve to close given a main steam isolation signal was assumed for post-trip return-to-power concerns and the failure of one high pressure safety injection (HPSI) pump to start on SIAS was assumed for pre-trip fuel performance concerns. The moderator temperature coefficient (MTC), Doppler coefficient, and CEA worth on trip used in the HELB analysis bound the fuel conditions expected during the first cycle as do those presented in subsection 15.1.5.1 of the FSAR. However, the MTC, Doppler coefficient and CEA worth used in the HELB analysis are best-estimate data. That is, the calculational uncertainties placed on the bounding moderator temperature coefficient (±15%), bounding Doppler coefficient (±15%), and bounding CEA worth (±10%) in the FSAR analysis are not included in the HELB analysis. This difference between the FSAR and HELB analysis is not a reduction in the conservatisms required by the SRP but a reduction in the calculational uncertainties applied to the conservatisms required by the SRP.

The steam line break analyses performed for the HELB/control system interaction study remove certain calculational uncertainties on fuel performance data but still meet the requirements as described in SRP 15.1.5.

Table 1 Summary of HELB/Control System Interactions

HELB TYPE	HELB CRITERION	PLCS	PPCS	SBCS	RRS	SGBS	RCP	BCS	MFWCS	TGCS	Bounding FSAR Subsection
LOCA	PCT	X	X	NMA	X	NMA	NMA	NI	NMA	X	15.6.3.3
	SDC	NMA	NMA	NMA	NMA	NMA	NMA	NI	X	NMA	15.6.3.3
	RCPB	NMA	X	NMA	NMA	NMA	NMA	NI	NMA	NMA	15.6.3.3
SLB	FP	NMA	NMA	X	X	NMA	NI	NI	NMA	X	15.1.3.1
	RTP	NMA	NMA	X	NMA	NMA	NI	NI	X	NMA	15.1.3.1
	RCPB	NMA	X	NMA	NMA	NMA	NI	NI	NMA	NMA	15.1.3.1
FWLB	RCSP	X	X	NMA	NMA	NMA	NI	NI	NMA	NMA	15.2.3.1
RCSBOC	RR	NI	NI	NMA	NI	X	NMA	NI	X	NI	15.6.3.1

X = Indicates that a HELB/control system interaction is possible and/or assumed.
NI = No interaction (NI) between the HELB and control system is possible.
NMA = Potential failure modes are no more adverse (NMA) than the control system behavior assumed in the FSAR.
PCT = Peak Clad Temperature.
SDC = Delaying entry into shutdown cooling.
RCPB = Reactor coolant pressure boundary integrity violation due to pressurizer heater misoperation.
FP = Pre-trip fuel performance.
RTP = Post-trip return-to-power.
RCSP = Reactor coolant system pressure.
RR = Radiological releases.

TABLE 2

LOCA ANALYSIS

BEST ESTIMATE LIMITING SMALL BREAK ANALYSIS

INPUT PARAMETERS:	BEST ESTIMATE	FSAR ANALYSIS LICENSING
BREAK SIZE, FT^2	0.05	0.05
BREAK LOCATION	BOTTOM COLD LEG	BOTTOM COLD LEG
CORE POWER, MWT	3458*	3478
MULTIPLIER ON ANS DECAY HEAT	1.0	1.2
MTC, $\Delta\rho/°F$	0.0*	$+0.15 \times 10^{-4}$
AXIAL SHAPE INDEX, ASI	-0.3*	-0.3
HPSI PUMP FLOW	NOMINAL*	MINIMUM
RWT WATER TEMPERATURE, °F	70.0	120.0
RT & SIAS SETPOINT, PSIA	1763.	1560.**
RESULTS:		
TIME OF RT & SIAS, SEC	25	105
PEAK CLADDING TEMPERATURE, °F	1044	1732
MAXIMUM CORE UNCOVERY, FT	1.7	3.8
DURATION OF CORE UNCOVERY, SEC	826	1610

NOTES:

* Actual SONGS 2 & 3 Technical Specification Limits with monitoring uncertainty included.

** Actual setpoint is much higher. The FSAR analysis setpoint is conservatively based on a large break LOCA or large steam line break containment conditions.

LIST OF ACRONYMS

BCS	Boron Control System
ECCS	Emergency Core Cooling System
FWLB	Feedwater Line Break
HELB	High Energy Line Break
LOCA	Loss of Coolant Accident
MFWCS	Main Feedwater Control System
PLCS	Pressurizer Level Control System
PPCS	Pressurizer Pressure Control System
RCP	Reactor Coolant Pump
RCSBOC	Reactor Coolant System Break Outside Containment
RRS	Reactor Regulating System
SBCS	Steam Bypass Control System
SGBS	Steam Generator Blowdown System
SLB	Steam Line Break
TGCS	Turbine-Generator Control System