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SUBJECT: Forwards response to 830630 request for addl info re control sys failures & qualification of control sys.Info concerning effect of pressurizer pressure signal on steam bypass control sys & reactor regulating sys also encl.

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September 15, 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 failures or malfunctions caused by the identified HELB.

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

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.

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

Very truly yours,

T.D. Mercurio

for M. O. Medford
Supervising Engineer
San Onofre Units 2 & 3 Licensing

Enclosures
cc: H. Rood(to be opened by addressee only)

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ENCLOSURE

SAN ONOFRE NUCLEAR GENERATING STATION - UNITS 2 AND 3
CONTROL SYSTEMS FAILURES & QUALIFICATION OF CONTROL SYSTEMS
REQUEST FOR ADDITIONAL INFORMATION

In response to NRC questions 222.43 "Qualification of Control Systems" and 222.44 "Control Systems Failures", SCE Letters dated April 1 and 20, 1983 were provided. The intent of these Letters was to respond to License Conditions 2.c(12) and 2.c(10) of the San Onofre Generating Station Units 2 and 3 Operating Licenses respectively.

It is not clear from Enclosure 1 of the April 1, 1983 Letter that the following considerations were taken into account in performing the analysis for each HELB:

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

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

Please verify that these items were considered in performing the HELB analyses. Also, please identify those control systems impacted by each of the HELBs, provide a short description of the consequences and identify the FSAR events that bound these HELB scenarios.

In response to multiple control systems failures due to common sensors and sensing lines, the April 20, 1983 letter states that erroneous pressurizer pressure signals to the Reactor Regulating System and the Steam Bypass Control System will not affect the output of these systems because the pressurizer pressure signal is used only as a compensating or bias signal. Please provide additional detail in support of this argument. If the argument cannot be supported, please verify that the consequences of the transient are still bounded by a Chapter 15 analysis.

Response

A. Additional Information on HELBs

The high energy line break (HELB) evaluations considered the effects of the HELBs on non-safety grade or control systems and assessed the impact of any consequential failures on the course of the HELBs analyzed in Chapter 15 of the Final Safety Analysis Report (FSAR) for San Onofre Units 2 and 3. The failures included those resulting from either environmental or direct physical interactions (jets, pipe whip, and flooding) between the HELB and the non-safety grade or control systems. Individual as well as multiple failures were considered in the evaluations.

The following non-safety grade or control systems may impact the consequences of the HELBs.

1. Reactor Regulating System (RRS)
2. Pressurizer Level Control System (PLCS)
3. Pressurizer Pressure Control System (PPCS)
4. Main Feedwater Control System (MFWCS)
5. Boron Control System (BCS)
6. Reactor Coolant Pumps (RCP)
7. Steam Bypass Control System (SBCS)
8. Turbine Generator Control System (TGCS)
9. Steam Generator Blowdown System (SGBS)

The high energy line breaks analyzed in Chapter 15 of the FSAR are:

(a) Loss-of-Coolant-Accident (LOCA)
(b) Steam Line Break (SLB)
(c) Feedwater Line Break (FWLB)
(d) Reactor Coolant System Break Outside Containment (RCSBOC)

A brief description of the consequences of the failures (both individual and multiple) of the control systems mentioned above on the HELBs are provided below along with the resolution of each major concern.

1. Failure of the Reactor Regulating System

A detailed design review of the RRS identified that a malfunction of this system can occur due to environmental effects on the excore neutron flux detectors and/or the steam flow transmitters which are located inside the containment. This could result in an unplanned control element assembly (CEA) withdrawal.

For a RCSBOC, the RRS is not impacted, since all of the RRS components are located inside the containment or in areas not subject to this HELB event. For the FWLB, the RRS malfunction effects are bounded by the FSAR Chapter 15 analysis (Subsection 15.2.3.1). This is because, if the RRS should inadvertently withdraw the regulating CEAs during a FWLB, there would be an additional primary to secondary heat removal mismatch resulting in an earlier reactor trip. The earlier reactor trip will provide additional secondary system inventory for post-trip heat removal resulting in a lower peak RCS pressure than that presented in Chapter 15.

Rod motion prior to reactor trip is of no consequence to the large break LOCA analysis, since reactivity is controlled rapidly (within 0.5 second) by moderator voiding. For a small break LOCA if the regulating CEAs are withdrawn, the pre-trip core power will increase resulting in a slower RCS depressurization and a delay in reactor trip and SIAS. The effect of the RRS malfunction on small break LOCA was analyzed and the analysis showed a slight increase in the core uncovery and the peak clad temperature (PCT). The increases, however, were well within the established margins for this event.

For a steam line break, an increase in core power due to the withdrawal of the regulating CEAs could lead to a pre-trip fuel performance degradation more severe than that presented in FSAR Chapter 15. Post-trip consequences are not impacted, since the RRS malfunction cannot affect a successful reactor trip. The Core Protection Calculators (CPCs) provide protection against pre-trip fuel performance degradation during SLBs. However, the CPCs are not qualified to operate in a harsh environment. Partial protection against the spurious rod withdrawal is provided by the high linear power trip (HLPT) function of the plant protection system. The HLPT is qualified to operate for 55 seconds in a harsh environment. Beyond this time, another trip mechanism should be employed to prevent pre-trip fuel performance degradation. For SLBs larger than 0.5 ft^2 in size, a reactor trip occurs within 55 seconds, and consequently these breaks are protected against fuel performance degradation due to spurious rod withdrawal, by the HLPT. For SLBs smaller than or equal to 0.5 ft^2 an analysis was completed to determine whether the high containment pressure trip would provide adequate protection for the event with a RRS malfunction. The analysis showed that no degradation in fuel performance occurs and that adequate protection exists. Consequently, the SLB in combination with a RRS malfunction results in acceptable consequences.

2. Failure of the Pressurizer Level Control System

A detailed design review of the PLCS indicated that the components of this control system are environmentally qualified to withstand the consequences of HELBs. Without undertaking a detailed evaluation of jet impingement effects on this control system, it was assumed that the control system could fail due to jet impingement effects.

For the RCSBOC, the malfunction of the PLCS has no impact since the amount of primary system fluid exiting the break would not be impacted. Note that the limiting break considered in FSAR Section 15.6.3.1 occurs upstream of the letdown control valve, and charging flow is assumed to be at its maximum rate. Similarly, for the SLB, with a PLCS malfunction, the consequences are bounded by the analysis presented in FSAR Section 15.1.3.1. This is because the increasing or decreasing pressurizer level has no significant impact on this cooldown transient.

For a large break LOCA, the PLCS malfunction is not a concern, since a SIAS occurs within a few seconds, and isolates the letdown line. However, for the small break LOCA, the letdown flow remains high until a SIAS due to the postulated PLCS malfunction. Some additional RCS inventory will be lost, which will slightly reduce the time to core uncovery. Calculations were performed for the limiting small break LOCA (0.05 ft^2

break) to quantify the reduction in time to core uncovery and the resulting increase in PCT due to the PLCS malfunction. The calculations assumed that the letdown control valve remained fully open until the time of SIAS. An additional 414 lbms of inventory was lost through the letdown line. This additional loss reduced the time to core uncovery by less than one second, resulting in an increase in PCT of less than 1°F.

The malfunction of the PLCS such that the letdown flow is minimized and charging flow is maximized was evaluated for the FWLB. The results of the evaluation showed that the pressurizer does not fill solid during the initial pressurization period, or prior to operator action at 30 minutes.

3. Failure of the Pressurizer Pressure Control System

A detailed system design review indicated that the PPCS is environmentally qualified. Without resorting to a jet impingement evaluation, it was assumed that the PPCS could fail due to jet impingement on the components of the system.

For the RCSBOC, there is no impact on the control system due to this break, since all the components of the PPCS are located inside the containment or in areas not subject to this HELB event. For the FWLB, the consequences of this malfunction are bounded by the FWLB analysis presented in FSAR Section 15.2.3.1. This is because, (1) if the PPCS malfunctions such that all of the pressurizer heaters turn on, there will only be a very insignificant impact on the peak RCS pressure due to the additional energy input by the heater in comparison to the primary to secondary heat removal imbalance (1.5 MWt vs. 1700 MWt), and (2) if all the pressurizer sprays were actuated initiating flow in excess of what is required to compensate for RCS pressurization with the heaters de-energized, then the effect would be a reduced peak RCS pressure.

For the large break LOCA, from a RCS energy removal consideration the malfunction of the PPCS such that the pressurizer heaters remain on after heater uncovery has no significant impact on the event. This is due to the fact that the heating efficiency is rapidly decreased, subsequent to heater uncovery. The additional energy provided by the heaters is negligible (<0.01%) compared to the RCS energy removal by a large break.

For the small break LOCAs, the pressurizer drains slowly. Consequently, during the period when the pressurizer is draining, if the heaters remain on, it will slow the RCS depressurization, delaying reactor trip and SIAS. A slight increase in PCT may result. Continuous heater operation will also slow the post-LOCA long term cooldown and uncovery. The heaters in the pressurizer are 50 kw Watlow heaters. The loss of submergence of a pressurizer heater while it is energized will result in the deenergization of the unwetted portion of the heater. Time to failure of the uncovered sheath is estimated to be 10 minutes. Typical failures would be: an open circuit caused by melting of the internal resistance wire, or a short circuit caused by arcing between the nickel clad conductors and the heater sheath. Although sheath rupture is an unlikely failure, should it occur a pressure seal internal to each heater and tested to 5000 psi would have to suffer extensive damage to result in any appreciable leakage of primary coolant. In addition, at least a twenty inch length of unheated compressed MgO/BN insulation (similar to concrete) would have to be forced out through the necked down pressure seal area of the sheath to complete a flow path. Therefore, even if the sheath failed there would be no leakage from the pressurizer.

4. Failure of the Main Feedwater Control System

A detailed design review of the MFWCS identified that this control system may fail due to environmental effects of inside containment breaks on the steam flow transmitters, the feedwater flow transmitters, and/or the steam generator downcomer water level transmitters. Consequently, this control system was assumed to fail.

For the RCSBOC, the effect of the MFWCS failure is bounded by the analysis presented in Section 15.6.3.1, since this analysis maximizes the flow out through the letdown line break by assuming no losses through the letdown line and associated valves. Note that the effect of a MFWCS failure such that the primary to secondary heat removal is degraded is a higher RCS pressure during the transient, resulting in a higher break flow. This effect is more than offset by not considering line losses for flow through the letdown line.

For the FWLB, the consequences of the failure in the MFWCS are bounded by the analysis presented in FSAR Section 15.2.3.1. This is because (1) if the MFWCS should increase the feedwater flow, then the steam generator liquid inventory will not be depleted as quickly resulting in a steam generator (SG) behavior similar to that for a smaller break area, and (2) a decrease in feedwater flow rate is bounded by the FSAR FWLB event analysis which assumes that all feedwater flow is diverted to the break.

If a failure in the MFWCS were to erroneously decrease feedwater flow or set the post-trip SG level at a lower-than-programmed level, the RCS heat removal rate and SG level may be decreased during a LOCA. For large break LOCAs this failure is unimportant since SG heat transfer is not a significant factor in RCS heat removal. For small break LOCAs, during the pre-trip portion of the event, this failure may slow RCS depressurization and cause a decrease in SG level. A low steam generator level causes a reactor trip and auxiliary feedwater initiation which terminates the impact of SG heat transfer degradation on the post-trip portion of the transient. A small perturbation in the early part of the transient will not have any measurable impact on the transient results and, therefore, its consequences are bounded by the analysis presented in FSAR Section 15.6.3.3.

If the MFWCS, due to a failure, were to increase feedwater flow or set the post-trip SG level at a higher-than-programmed level, the steam generators may overfill. This will have an impact on the post-LOCA RCS cooldown rate. Since the RCS and SGs are cooled simultaneously, the increased secondary side inventory will slow the cooldown rate for small break LOCAs. This could delay entry into shutdown cooling (SDC) operation. For large break LOCAs, this failure is not a concern, because the break will be able to remove the additional sensible heat from the secondary side. A post-LOCA long term cooling analysis indicated that there is no significant impact on the post-LOCA cooldown due to the increased feedwater flow rate.

For the SLB, the increased post-trip feedwater flow rate due to a failure in the MFWCS could increase the heat removal capacity of the secondary system, with a potential for a post-trip return to power scenario not bounded by the FSAR Chapter 15 analyses. There is no pre-trip concern due to the MFWCS failure induced increased cooldown, since this may merely result in an earlier reactor trip. The impact of an increase in feedwater flow on SLBs was quantitatively analyzed. The method of determining the impact of the potential failure consisted of reviewing the FSAR calculation for conservative assumptions, determining the amount of protection against a return-to-power event gained by removing the conservatisms, and comparing that to the impact of an increase in main feedwater flow. The conservatisms in the FSAR Chapter 15 analyses are the penalty factors applied to the moderator, Doppler, and CEA reactivities. The positive reactivity inserted due to the MFWCS malfunction is less than the margin provided by the analytical conservatisms. Therefore, there is sufficient margin to prevent a return-to-power event due to an increase in feedwater flow during a SLB.

5. Failure of the Boron Control System

The boron control system (BCS) is used to adjust RCS boron concentration during all modes of operation. The BCS is a manually operated system. Therefore, it would require an operator error to alter the boron concentration. Operator errors are not included in the scope of this analysis. Therefore, all HELB/BCS interactions are removed from further consideration.

6. Failure of the Reactor Coolant Pumps

   The reactor coolant pumps maintain a constant coolant flow in the RCS. The impact of losing the flow from all four RCPs due to a loss of offsite power is included in the FSAR Chapter 15 analyses. Other RCP malfunctions impacts on LOCAs, FWLBs, SLBs, and RCSBOC are either bounded by the analyses presented in Chapter 15 of the FSAR or would not be impacted by inside containment breaks.

7. Failure of the Steam Bypass Control System

   A detailed design review of the SBCS indicated that the turbine bypass valves (TBVs) can fail open due to environmental effects on the steam flow and steam header pressure transmitters. Consequently, a failure of this control system was postulated.

   For both the small and large break LOCAs, the consequences of the opening of the TBVs are bounded by the FSAR Chapter 15 analyses (Subsection 15.6.3.3), since opening of the TBVs will assist RCS depressurization and improve the performance of the ECCS.

   For the FWLB, a malfunction of the SBCS resulting in the opening of the TBVs will not adversely impact the event consequences. This is because the opening of the TBVs will increase the secondary system heat removal, resulting in a decrease in the peak RCS pressure.

   For the RCSBOC, the SBCS failure resulting in the opening of the TBVs will increase the secondary system heat removal. This will in turn decrease the RCS pressure resulting in reduced mass releases through the break and reduced radiological releases. Therefore, this scenario is bounded by the RCSBOC analysis presented in FSAR Subsection 15.6.3.1.

   If the SBCS were to malfunction such that the TBVs opened during a SLB, the RCS would respond to a greater steaming rate than that due to the break. If this failure occurs in combination with a large break (>0.5 ft^2), no concern exists because a reactor trip on high containment pressure will occur in the time span in which the high linear power trip is qualified to operate. If the malfunction occurs in combination with a small break (<0.5 ft^2), the containment pressure trip may not occur during the period when the high linear power trip is environmentally qualified to operate. A concern for potential fuel performance degradation exists for small breaks in combination with a failure in the SBCS such that the TBVs open during the event. However, the impact of the SBCS malfunction is less severe than that of the RRS malfunction (see Section A above). In addition, the SBCS and RRS malfunctions cannot simultaneously occur due to the presence of the Automatic Withdrawal Prohibit (AWP). These assessments were reached using system design review results and were verified by means of an evaluation. Thus, the consequences of the SBCS malfunction during a SLB are bounded by those of the RRS malfunction during the SLB.

The potential for post-trip return-to-power during a SLB is dependent on the amount of steam flow for cooling the RCS. If the SBCS were to open the turbine bypass valves, in combination with a stuck open MSIV in the intact SG, the amount of steam flow available to cool the RCS would increase significantly. This impact on SLB event consequences due to a malfunction in the SBCS is not bounded by the FSAR analysis. Therefore, the impact of an increase in steam flow due to the generation of a spurious quick open signal to the SBCS was quantitatively analyzed. The additional steaming would only be of concern for steam line breaks upstream of the MSIV which impact the steam flow transmitters. During such a break the ruptured SG would be blowing down through the break, while the intact generator would be steaming to the condenser. For a break downstream of the MSIV, the steam flow transmitters and the CPCs are not impacted, and hence no return-to-power concerns exist.

A design review of the SBCS determined that the quick open signal would permit the turbine bypass valves to be opened for no more than 20 seconds. The precise opening time depends on the degree of interference caused by the SLB. The amount of RCS cooldown and resultant positive reactivity inserted was calculated for the maximum TBV opening time and compared to the conservatisms in the FSAR Chapter 15 analyses. These conservatisms are the penalty factors applied to the moderator, Doppler, and CEA reactivities. The positive reactivity inserted due to the SBCS malfunction is less than the margin provided by the analytical conservatisms. Therefore, a post-trip return-to-power will be prevented during a SLB upstream of the MSIV due to a malfunction of the SBCS.

8. Failure of the Turbine Generator Control System

The turbine generator control system (TGCS) controls the steam flow rate to the turbine. The system maintains the turbine steam pressure by opening or closing the turbine control valves (TCVs). A malfunction in the TGCS may cause the steam flow rate to increase above or decrease below that presented in the FSAR Chapter 15 analyses. In the HELB evaluations that follow the assumption was made that the TGCS failed either due to environmental effects or jet impingement.

The large break LOCA analysis assumes that the turbine control valves (TCVs) begin to close at the beginning of the transient. If the TCVs were to remain open until turbine trip the excess cooldown would enhance ECCS performance. Therefore, the consequences of TGCS malfunctions for large break LOCA events are bounded by the FSAR Chapter 15 analysis. For the small break LOCA analyses, the small pre-trip steam flow variation is accommodated by the TGCS. If the TGCS is assumed to fail so as to increase the steam flow, RCS depressurization would be hastened, causing an earlier reactor trip and SIAS. If the TGCS is assumed to fail so as to decrease the steam flow, the secondary pressure will increase to the main steam safety valve (MSSV) setpoint earlier than predicted in the FSAR Chapter 15 analysis. This would slightly diminish the RCS heat removal. Once a reactor trip occurs, the FSAR Chapter 15 analysis conservatively assumes that main steam isolation occurs and hence the MSSVs actuate. For the limiting small break (0.05 ft^2) LOCA, the MSSVs actuate at approximately 30 seconds. If the TGCS malfunctioned such that the MSSVs actuated at the beginning of the transient, there would be no measurable impact (less than 5°F) on PCT. This is because, the PCT is achieved very late in the transient (~1900 seconds). Therefore, the consequences of this failure are bounded by the FSAR Chapter 15 analysis.

For the FWLB, if a malfunction of the TGCS would result in an increased turbine control valve (TCV) area, the consequences would be bounded by the FSAR Chapter 15 analysis (Subsection 15.2.3.1). This is because the steaming rate would increase resulting in lower peak RCS pressure. If the TGCS decreased the TCV area during a FWLB, the decreased steaming rate would result in an earlier reactor trip and turbine trip. An earlier reactor/turbine trip than that presented in the FSAR Chapter 15 analysis (Subsection 15.2.3.1) will result in greater post-trip heat removal capability by the steam generators and a lower peak RCS pressure.

A RCSBOC results in a depressurization of the RCS. The RCSBOC occurs outside containment in the auxiliary building and does not impact the TGCS in the turbine building.

If the TGCS were to malfunction such that the TCVs reduced the total steaming rate during a SLB event, the RCS would respond to an excess steam demand smaller than that due to the break. Consequently, plant response would be identical to that of a smaller SLB event. The SLB presented in the FSAR Chapter 15 analysis is the limiting break size event. Therefore, the malfunction of concern will not produce a more severe transient and the consequences are bounded by the FSAR Chapter 15 analysis.

If the TGCS were to malfunction such that the TCVs opened during a SLB, the RCS would respond to a greater steaming rate than that due to the break. If this failure occurs in combination with a large break (>0.5 ft^2), no concern exists because a reactor trip on high containment pressure will occur in the time span in which the high linear power trip is qualified to operate. If the malfunction occurs in combination with a small break (<0.5 ft^2), the containment pressure trip may not occur during the period when the high linear power trip is environmentally qualified to operate. A concern for potential fuel performance degradation exists for small SLBs in combination with a failure of the TGCS which result in the opening of the TCVs. An analysis of the TGCS failure resulting in the opening of the TCVs during a SLB was performed. The failure of the TGCS was modeled as an incremental increase in the break area. The small additional flow area generated when the turbine control valves modulate open caused a small core power (<5% full power) increase. This increase occurred after the time period for which the high linear power trip is qualified to operate. The event was terminated by a reactor trip on high containment pressure. The results of the analysis showed that the total core power increase was not sufficient to cause fuel damage during this event.

9. Failure of the Steam Generator Blowdown System

The steam generator blowdown system (SGBS) removes fluid from the secondary side of the steam generator near the tube sheet and purifies it. The fluid is then discharged into the feedwater system. Misoperation of the system may cause the blowdown flow rate to increase above or decrease below that presented in the FSAR Chapter 15 analysis. In the HELB evaluations that follow the assumption was made that the SGBS malfunctions either due to environmental effects or jet impingement.

In the FSAR Chapter 15 LOCA analysis, the SG blowdown flow is assumed to be zero. A malfunction of the SGBS resulting in an increase in blowdown flow will not impact the consequences. This is because the small loss of secondary inventory due to the increased blowdown flow will not degrade the primary to secondary heat transfer as the main or the auxiliary feedwater flow will maintain the steam generator water level.

A steam line break results in an excess cooldown of the RCS. A malfunction of the SGBS resulting in an increase in blowdown flow during a SLB will result in a less severe RCS cooldown, since less fluid will be available in the steam generator for RCS heat removal. Therefore, the consequences a SLB with a malfunction of the SGBS is bounded by the FSAR Chapter 15 analysis (Subsection 15.1.3.1) which assumes the SGBS to be not removing fluid from the steam generator.

A feedwater line break (FWLB) results in the removal of fluid from the SG. The rate of fluid removal depends on the break size. In the FWLB events presented in the FSAR Chapter 15 analysis the SG blowdown flow is assumed to be zero. If the SGBS were to malfunction, and the blowdown flow were to increase, the secondary fluid removal rate will increase. This would have the same effect as slightly increasing the break size. The FWLB event presented in the FSAR Chapter 15 analysis is only slightly sensitive to the break area. The break area presented in the FSAR (0.2 ft^2) results in maximizing the peak RCS pressure. An increase in blowdown flow rate has the same effect as a slight increase in break area and, therefore, will result in a small decrease in peak RCS pressure. Additionally, the heat removal capability of the intact SG will be preserved, since the SGBS will be isolated on emergency feedwater actuation signal. Therefore, this interaction is bounded by the FSAR Chapter 15 analysis (Subsection 15.2.3.1).

A RCS break outside containment (RCSBOC) results in a depressurization of the RCS. The SG blowdown flow is assumed to be zero during this event. A malfunction in the SGBS such that blowdown flow increases will remove a small amount of secondary fluid from the SGs. The small blowdown flow will not impact primary to secondary heat transfer significantly. There may be an insignificant impact on the primary pressure and enthalpy, and hence, letdown line break flow rate. The consequences of this event scenario are bounded by the FSAR Chapter 15 analysis (Subsection 15.6.3.1).

10. Multiple Failures of Control Systems

(1) Loss of Coolant Accident

Of the potential control system failures investigated only the PLCS, PPCS, and RRS malfunctions adversely influence the peak cladding temperature (PCT) for the small break LOCA. The remaining control systems have no influence on the PCT or their effects are bounded by the FSAR Chapter 15 analysis. Similarly, for small break post-LOCA long term cooling, only a failure of the PPCS may increase the feedwater requirements above the value calculated in the FSAR Chapter 15 analysis. For the large break LOCA, none of the control system failures have a significant impact.

It was assumed that if all of the control systems fail simultaneously, their combined effect will not be worse than the summation of their individual effects. This is because the effects of individual control system failures are separable and small. Combining the increases in PCT for individual malfunctions of the PLCS, PPCS, and RRS during a small break LOCA resulted in an increase in PCT of maximum 158°F as the combined effect of these malfunctions. This small increase was well within the established 688°F PCT margin for this event.

(2) Steam Line Break

The postulated failures which may impact a Chapter 15 SLB event have been demonstrated to be bounded by the FSAR analysis if they occur singularly. Additionally, the impact of multiple control system malfunctions was addressed. The malfunctions considered affect RCS inventory, pre-trip fuel performance, or post-trip return-to-power. Malfunctions which impact different consequences were not combined since no additional impacts on either consequence could be generated. As an example, no impact on pre-trip fuel performance could be generated by opening the TBVs after reactor trip. The malfunctions which had no impact or which were demonstrated to not occur were not combined.

The following malfunction combinations were addressed; malfunction of the TGCS and RRS during small inside containment breaks, and malfunction of the SBCS and FWCS during a SLB. The malfunction of the SBCS is not combined with that of the RRS since the Automatic Withdrawal Prohibit (AWP) will prevent this interaction combination from occurring.

The malfunction of the TGCS and RRS during a small SLB was analyzed. The event analyzed was a small steam line break (<0.5 ft^2) during which the regulating CEAs were assumed to withdraw and the TCVs opened such that at 55 seconds the high linear power trip was not encountered. The trip was then assumed to be inoperable and the event continued until the high containment pressure trip signal was generated and the reactor tripped. The system response to the combined malfunctions of the TGCS and RRS resulted in a power excursion of no more than 125% power. The peak power is not high enough to cause fuel damage. Therefore, this malfunction combination was removed from concern.

The malfunction of the FWCS and SBCS during a SBCS during a SLB was also analyzed. The analysis consisted of adding the positive reactivities inserted due to the separate malfunctions and comparing the sum to the negative reactivity gained by removing the previously discussed analytical conservatisms. The net positive reactivity inserted due to combined malfunctions of the SBCS and MFWCS is less than the negative reactivity available through removing the analytical conservatisms. Thus a post-trip return-to-power will not occur due to these malfunctions. Therefore, this combination was removed from concern.

(3) Other HELBs

There is only one control system failure (PLCS malfunction) that is not bounded by the FSAR Chapter 15 analysis for the FWLB event. Therefore, no multiple failure consideration is relevent for this event. Similarly, for the RCSBOC no multiple failures of the control system need to be evaluated since the individual control system failures are either bounded by the Chapter 15 analyses, or since the control systems are not impacted by the RCSBOC.

11. Single Failure of Safety Related Systems

The FSAR Chapter 15 analyses for the high energy line breaks were performed assuming the worst single failure of safety related systems (e.g., failure of one high pressure safety injection pump for the SLB event). In the HELB evaluations discussed above the individual or multiple control system failures were superimposed on the reference HELB analysis which accounted for a single failure in a safety-related system.

B. <u>Additional Information on Multiple Control System Failures</u>

The Reactor Regulating System (RRS) and the Steam Bypass Control System (SBCS) are briefly described below, followed by an evaluation of the consequences of erroneous pressurizer pressure signals to these systems and the other pertinent control systems.

<u>Reactor Regulating System</u>

The Reactor Regulating System (RRS) provides a means for the automatic control of the average reactor coolant temperature and the capability to follow ramp or step turbine load changes. This objective is accomplished by automatically moving preselected groups of CEAs. The RRS provides contact outputs to the Control Element Drive Mechanism Control System (CEDMCS) to cause CEA insertion or withdrawal and to determine the associated rates when the CEDMCS is in the Automatic Sequential Mode.

The RRS receives a turbine load index signal as a linear indication of load. This power reference is fed to a temperature programmer which establishes the desired reactor coolant average temperature. Using inputs of the hot leg and cold leg temperatures from each primary loop, the RRS calculates the average primary coolant temperature. Additionally, power range neutron flux and pressurizer pressure are compensating inputs to the system. However, in the as-built configuration, the electrical wiring necessary to input pressurizer pressure signal to the RRS is not hooked up to this control system.

The RRS performs dynamic operations on the errors in power (turbine and reactor) and temperature to produce an error signal whose magnitude and sign are used to determine the control actions necessary to maintain reactor coolant temperature to within a band of its programmed value. The error signal has units of pressure and indicates to the operator the error in secondary pressure that would result from a deviation of the average primary coolant temperature from the referenced program. To achieve automatic temperature control, one of the two plant RRS is selected for use by the operator. The RRS supplies status (insert, withdrawal and hold) and rate (high and low) signal to the CEDMCS which responds in accordance with these demands to position the regulating CEAs. A number of RRS status and control signals are provided on the main control board (MCB) to verify system operation.

Steam Bypass Control System

The Steam Bypass Control System (SBCS) provides a means for controlling excess NSSS thermal energy and maximizing unit availability. This objective is achieved by opening the turbine bypass valves (TBVs) to avoid challenges to the secondary safety valves, to the high pressurizer pressure reactor trip, and to the primary safety valves.

The SBCS continuously monitors main steam header pressure and main steam flow rates. Additionally, the pressurizer pressure is a compensating input to the system. When an excessive increase in header pressure is detected, steam from the main steam lines is directly passed to the condenser through the turbine bypass valves which are modulated to control main steam header pressure to a programmed setpoint. To prevent a single component failure from opening more than one valve, the coincidence of two independently generated demand signals is made necessary to open any one valve. For this purpose, two parallel circuits, namely the main and permissive circuits, are employed. Two main steam header pressure setpoints are generated in a similar fashion; one is selected for use in the Main Modulation Control Channel, which generate the valve control signals, and the other in the Permissive Modulation Control Channel, which generates the valve permissive signals.

When a large decrease in load is detected such that it cannot be accommodated by the modulation control of the turbine bypass valves, a "valve quick opening" signal is generated which overrides the modulation control and opens the turbine bypass valves in one second or less. As in the modulation control case, the coincidence of two independently generated demand signals is necessary for the quick opening of any one valve. The "Main Quick Opening Demand" and the "Permissive Quick Opening Demand" signals are employed to accomplish this.

The main and permissive channels identified above receives the pressurizer pressure signals from two independent pressure transmitters. Thus, a single failure in any one of the pressure transmitters will not result in either the modulation or quick opening of the turbine bypass valves.

Consequences of Erroneous Pressurizer Pressure Signals to the RRS and SBCS

The NSSS performance will not be impacted due to an erroneous pressurizer pressure signal to the RRS and SBCS, since:

(1) in the as-built configuration of the RRS the pressurizer pressure signal is not an input to the RRS, and

(2) the failure of a single pressurizer pressure transducer resulting in the generation of an erroneous pressure signal will not cause either the modulation or quick-opening of the turbine bypass valves.

If the RRS were to be wired up such that the pressurizer pressure is an input to this system, Subsections (4) 2.a.1, (4) 2.a.2, and (4) 3.a,b,c, and d in the original response to NRC Question 222.44 would be revised as follows:

(4) 2.a.1 Pressurizer Pressure Signal Fails Low to RRS, SBCS and PPCS

If a malfunction causes a low pressurizer pressure signal to be transmitted, the PPCS effect would be that the pressurizer heaters will be turned on, and the pressurizer sprays will be shutoff. The regulating group CEAs may be withdrawn due to a low pressurizer pressure signal to the RRS. The SBCS will not be impacted since a failure of both pressure transducers is required to open the turbine bypass valves.

Evaluation of Plant Response:

The effect of turning on the pressurizer heaters and withdrawing the CEAs would be an increase in the RCS pressure and core power. The reactor would eventually trip on a low DNBR or a high pressurizer pressure, if the operator did not take any mitigating actions. This scenario is bounded by the loss of condenser vacuum (LOCV) event analysis provided in the FSAR paragraph 15.2.1.3 from a pressure boundary integrity point of view. This analysis assumes a loss of feedwater flow and unavailability of the SBCS. The above scenario would cause a primary to secondary heat removal imbalance and a RCS pressure rise no more limiting than the LOCV. From a fuel performance and radiological consequence perspective, the scenario is bounded by the uncontrolled CEA withdrawal event analysis presented in the FSAR Paragraph 15.4.1.2. The FSAR analysis considers the effect of uncontrolled CEA withdrawal at power which results in a primary to secondary heat removal imbalance impacting fuel performance. No fuel failure was predicted to occur.

(4) 2.a.2 Pressurizer Pressure Signal Fails High to RRS, SBCS and PPCS

If a malfunction causes a high pressurizer pressure signal to be transmitted, the PPCS effect would be that the pressurizer sprays would come on and the pressurizer heaters would be de-energized. The RRS may insert the regulating group CEAs in response to the high pressurizer pressure signal. Additionally, the SBCS will not modulate or quick open the turbine bypass valves since a failure of both pressure transmitters is required to open these valves.

Evaluation of Plant Response:

The reactor would trip on a low pressurizer pressure setpoint and a SIAS may result. This scenario is bounded by the analysis presented in FSAR paragraph 15.6.3.4 for the inadvertent opening of a pressurizer safety valve. RCS depressurization is more rapid for this event. The reactor trips on a low pressurizer pressure trip setpoint, and no fuel pins experience a DNBR less than 1.19 (CE CHF correlation), thus preventing any violation of the fuel thermal limits. Additionally, there are no event related offsite doses since the integrity of the primary and secondary system is maintained.

(4) 3.a Evaluation of Pressure and Level Signals Failing Low Due to Instrument Tap Damage on Plant Response

This event can only be caused by the occurrence of a broken process sensing line coincident with rupture of the level transmitter diaphram. The following analysis of this event is provided.

If the failure causes a low pressure and level signals to be transmitted, the pressurizer heaters would turn on and the pressurizer sprays would decrease flow.

The PLCS would decrease letdown flow and increase charging flow. The RRS may withdraw the regulating group CEAs in response to a low pressurizer pressure signal.

The low pressurizer pressure input to the SBCS would not cause the turbine bypass valves to open, since a low primary pressure would indicate overcooling of the primary. Due to increased core power and actuation of the pressurizer heaters, the reactor may trip on a low DNBR or a high pressurizer pressure. The uncontrolled CEA withdrawal event analysis presented in FSAR Paragraph 15.4.1.2 bounds this scenario.

(4) 3.b <u>Evaluation of Pressure and Level Signals Failing High Due to Instrument Tap Damage on Plant Response</u>

This event is considered unlikely since a process sensing line break causes pressure to fail low and a perfect line crimp would not in itself result in a high pressure. Nevertheless, the following analysis is provided:

If the failure causes a high pressure and level signals to be transmitted then the pressurizer heaters would de-energize and the sprays would increase flow. The RRS may insert the regulating group CEAs in response to the high pressurizer pressure signal. The SBCS will not modulate or quick open the turbine bypass valves since a failure of both pressure transmitters is required to open these valves.

The PLCS would increase letdown flow and decrease charging flow. As a result of these control system actions, a low pressurizer pressure situation may result, leading to a possible reactor trip and SIAS on low pressurizer pressure.

The reactor trip on low pressure would prevent any fuel rods from experiencing a DNBR less than 1.19 (CE-1 CHF correlation) and there is no over-pressurization. Since there is no fuel failure and release of primary fluid to the atmosphere, the letdown line break event of Paragraph 15.6.3.1 clearly bounds the radiological consequences.

(4) 3.c <u>Evaluation of Pressure Signal Failing High and Level Signal Failing Low Due to Instrument Tap Damage on Plant Response</u>

This event is considered unlikely since it requires the process sensing line to be perfectly crimped coincident with a high pressure transient. Nevertheless, the following analysis is provided:

The plant response is similar for the PPCS, RRS, and SBCS as discussed above for the pressure signal failing high. The PLCS, however, would increase charging and decrease letdown. The increases in charging and pressurizer spray flows with no pressurizer heaters will lead initially to a low pressure condition, and to a steadily increasing pressurizer level. Additionally, due to decreasing core power, a low pressurizer pressure reactor trip condition will be reached. The conclusions stated for the pressure and level signals failing high also apply to this scenario.

(4) 3.d Evaluation of Pressurizer Pressure Signal Failing Low and Level Signal Failing High Due to Instrument Tap Damage on Plant Response

As discussed in the evaluation of both signals failing low, the PPCS, RRS, and SBCS response will be similar. The PLCS, however, would increase letdown flow and decrease charging flow. A reactor trip may occur on a low DNBR or on a high pressurizer pressure due to increased core power. The conclusions stated for the pressure and level signals failing low also apply to this scenario.