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U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region V
1450 Maria Lane, Suite 210
Walnut Creek, California 94596-5368

Attention Mr. John B. Martin, Regional Administrator

Dear Sir:

Subject: Docket No. 50-206
Safety Assessment, SONGS 1 Restart Report
San Onofre Nuclear Generating Station
Unit 1

By letter dated March 17, 1989, SCE provided to you a report titled "Safety Assessment, SONGS 1 Restart. This report, hereafter referred to as the restart report, provided SCE's evaluation of recent technical issues in relation to continued operation of San Onofre Unit 1. The purpose of this letter is to provide a supplement to the restart report and to expand on several key issues identified during discussions with Mr. F. R. Huey, the NRC Senior Resident Inspector for San Onofre Units 1, 2 and 3.

Enclosure 1 to this letter provides SCE's evaluation of two additional technical issues that were identified subsequent to submittal of the restart report. The first issue relates to nonconservative safety injection system flow diversion. The second issue relates to a non-conservative setpoint in the Overpressure Mitigation System. Similar to the previous technical issues identified in the restart report, these new issues have been determined to have low safety significance. Enclosure 1 provides a description of the issues including how the issues were identified, root cause determination, corrective actions and the basis for the safety significance determination.

The programmatic changes identified in the restart report as supplemented by Enclosure 1 are considered to bound the root causes of these new issues. Further, the low safety significance of these issues will not invalidate the overall conclusion in the restart report that return-to-service of SONGS 1 is acceptable and does not represent an undue risk to the health and safety of the public.

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During discussions with Mr. F. R.Huey, it was identified that several key aspects of the restart report should be expanded to provide additional clarification in these areas. Based on these discussions, SCE formulated seven specific questions and responses to address these areas. These questions and responses are provided in Enclosure 2 to this letter. This additional information reinforces SCE's position that the corrective actions previously initiated as discussed in SCE's October 3, 1988 letter providing the Independent Assessment of the Engineering and Technical Support to SONGS, and those identified in the restart report remain valid.

If you have any questions regarding the enclosed information, please let me know.

Very truly yours,

M O Medford

cc: F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3
U. S. Nuclear Regulatory Commission, Document Control Desk

ENCLOSURE 1

EVALUATION OF ADDITIONAL

TECHNICAL ISSUES

13. Safety Injection Flow Diversion/Alignment Delay

ISSUE: Operation of the SONGS 1 Safety Injection System (SIS) requires realignment of several pumps and valves within established time constraints. Signals to appropriate components during a large break LOCA or MSLB are initiated by the Safeguard Load Sequencing System (SLSS), and include sequencer-controlled or separate time delay relays. Several functional time delays in the realignment of required valves not accounted for in the accident analyses have been identified. These time delays contribute to diversion of water or delay in delivery of water to the reactor during a LOCA or MSLB. In addition to these delays, operational evolutions may involve alignment of manual valves that would divert water from SI delivery to the reactor. These time delays and potential flow diversion paths are each discussed below.

1. Main Feed Pump Miniflow Valve to Condenser (CV-36/37) Stroke Time - The realignment of the SONGS 1 Main Feedwater pumps for Safety Injection (SI) includes realignment of the pump miniflow protection path from the Condenser to the Refueling Water Storage Tank (RWST). The miniflow valves to the RWST (CV-875A/B) are opened on a signal from the SLSS, and the miniflow valves to the Condenser (CV-36/37) are closed on a limit switch signal from the respective RWST miniflow path valves. The stroke time for CV-36/37 was assumed in the accident analysis to be less than 9 seconds. Actual stroke time was approximately 23 seconds. Also, the Inservice Test program had an allowable stroke time of 45 seconds. Therefore, this diversion path could have allowed up to 36 seconds of flow not accounted for in the LOCA or MSLB analysis.

2. Main Feed Pump Miniflow Valve Wiring Error - In addition to the valve stroke time issue in Item 1, a second issue relating to the interconnection of these two sets of valves was identified. The two sets of valves were assumed to operate concurrently to meet the response time requirement for completing SI realignment assumed

in the transient analyses. However, the CV-36/37 solenoid pilot valves were wired to the wrong set of limit switches on CV-875A/B, so that the two sets of valves would have operated sequentially instead of concurrently. This would have delayed completion of SI realignment by a duration equivalent to the stroke time of CV-875A/B, causing a flow diversion from the operating SI train to the condenser not accounted for in the transient analyses.

3. Main Feed Pump Miniflow Bypass Valves - During certain operational evolutions, the main feed pump miniflow valves to the condenser (CV-36/37) are closed and their manual bypass valves are opened to maintain the miniflow path in operation. For example, to ensure miniflow protection for the feedwater pumps, the bypass valves would be used in the event of CV-36/37 controller oscillations at low main feedwater flow rates. In addition, CV-36/37 have been closed and their manual bypass valves opened if CV-36/37 required maintenance during operation. Opening of the manual bypass valves prevents automatic isolation of the miniflow paths to the condenser for SI actuation. Thus, this represents a potential diversion path of SI flow in the event of a LOCA or MSLB at the same time the manual valves are opened, not accounted for in the transient analyses.

4. Manual Bypass of Main Feedwater Pumps - The SONGS 1 SIS utilizes two SI pumps and the two main feedwater pumps to inject borated water to the reactor. The feedwater pumps are normally aligned to main feedwater system and are realigned to the SI header in the event of a safety injection signal. The main SI header at the discharge of the feedwater pumps is required to be maintained with borated water within Technical Specification concentration limits (currently, greater than 1500 ppm and less than 4300 ppm). During operation, minor leakage from the feedwater system and/or the RCS into the SI header slowly reduces the boron concentration in the SI header over a period of time. Because of this leakage, periodic purging of the SI header during operation is necessary. This purging is accomplished by starting the SI pumps and opening manual bypass valves around the feedwater pumps. This allows injecting borated water from the Refueling Water Storage Tank (RWST)

into the SI header without impacting the operation of the feedwater pumps. A recirculation path back to the RWST is accomplished by opening another manual valve in the SI header. Purging of the SI header by this process degrades the ability of the SIS to perform its safety function in two ways. First, the opening of the manual valve in the SI header which establishes a recirculation path back to the RWST introduces another path for diversion of SI flow from the reactor. Secondly, opening the manual bypass around the feedwater pumps introduces a configuration which reduces SI flow. With a bypass valve open, water would be diverted from the SI discharge of both feedwater pumps back through the bypass line to the section of piping between the SI pump and the suction of the feedwater pump. This diversion would occur due to the differential pressure developed by the feedwater pumps. Thus, this configuration would result in a recirculation path from the discharge of the feedwater pumps back to their own suction. The LOCA and MSLB accident analyses do not account for resultant diversion of pump discharge flow through this recirculation path.

5. Electrical Bus Voltage Dip - An evaluation has determined that concurrent operation of electrical loads during a LOCA transient with a degraded grid voltage condition would result in a voltage dip on the 480V buses. This voltage dip could delay the starting of the valve motors for SI valves MOV-850 A and B, and MOV-20, 21 and 22. This delay in valve operation would have been less than 4 seconds but would have caused a delay in injection of water to the RCS not accounted for in the accident analyses.

6. Delay In Actuating CV-875 A/B Limit Switches - During the Cycle 10 refueling outage modification to increase the closing force to reduce seat leakage on CV-875 A/B, it was identified that the valves do not respond within the time constraints assumed in the safety analyses when an SIS signal is provided to the solenoid pilot

valves. For example, with the initial position of the valve being CLOSED, the CLOSED limit switch which controls CV-36/37 will not change state for approximately 2 seconds after an OPEN signal is provided to the CV-875 A/B valves. Therefore, the signal from CV-875 A/B limit switches to close CV-36/37 will be delayed by this amount. The SLSS load schedule indicates a one second delay between the open signal to CV-875 A/B and the close signal to CV-36/37. Therefore, an additional one second delay in the actuation of CV-36/37 will occur due to this issue. The accident analyses did not account for diversion of flow from the reactor during this period of time.

REFERENCES: LER 1-89-11 dated April 24, 1989
NCR SO1-P-6751 (Rev. 4) dated April 12, 1989
NCR SO1-P-7159 dated April 4, 1989

DISCOVERY: Each of the items associated with this issue was identified at different times during review of modifications to the affected components, or through system reviews for various reasons. Modifications to affected components included increasing the closing force to reduce seat leakage on CV-875A/B, and reducing the stroke time of CV-36/37. System reviews were performed as a result of modifications to electrical bus loads, reanalysis of the LOCA transients as part of the Cycle 10 reload, and as part of the modifications to affected components discussed above. The increased sensitivity to system design engineering and overall accomplishment of safety function contributed to the identification of these issues.

ROOT CAUSE: SCE has completed a study to evaluate general deficiencies in the area of design, engineering and technical work which contributed to this issue. A specific root cause for all of the items associated with this issue cannot be determined at this time and additional evaluation is ongoing. At present, it is believed that the majority of these failures occurred as the result of one or more of the following:

1. There are no programmatic requirements for the development, update, compilation, review or verification of design basis documents.

2. Technical training for engineering personnel is too narrowly defined and fails to properly consider the engineer's function and needs. Technical training generally fails to provide an integrated system knowledge of plant design, analysis and operation. Without this or a detailed design basis, the ability of the individual to produce acceptable results is largely a function of his own capabilities and experience.

3. Opportunities to detect the errors were missed due to an absence on the part of the individuals involved to critically question the assumptions employed, input needed, methodology used or results achieved. Although successful in some organizations, management efforts to develop a questioning attitude have not been fully effective. This may have resulted, in part, from a lack of a formal management statement on this issue.

The above listed possible causes for the occurrence of these issues were previously identified as part of the review of the technical and engineering support for SONGS as documented in SCE's letter to the NRC dated October 3, 1988. Although these deficiencies are considered to bound the root cause of the issues identified, it is the corrective actions initiated as a result of these generic deficiencies that contributed to the identification of these new issues. Performing design changes and reviewing overall system impact with questioning attitude had a significant contribution to identifying these new issues.

<u>CORRECTIVE ACTION</u>: Although generic corrective actions have been implemented for the above generic deficiencies, specific actions have been taken to correct or account for the issues identified here. These actions are as follows:

1. CV-36/37 Stroke Time - A design modification to CV-36/37 has been implemented to reduce the "close" stroke time to less than nine seconds. The IST program has been revised to reflect the maximum closure time in accordance with design basis requirements.

2. Valve Wiring Error - The minimum flow subsystem interlock wiring has been modified such that CV-36/37 receive a close signal from the proper limit switches on CV-875 A/B.

3. Main Feed Pump Miniflow Bypass Valves - Administrative controls have been implemented to prevent the use of these bypass valves during periods of operation when the SI system is required to be operable.

4. Manual Bypass of Main Feed Pumps - Use of these manual bypass valves could result in the inoperability of the feedwater pumps due to the resultant reduction in SI flow. Because the purging of the SI header to maintain boron concentration is required periodically during operation and cannot be accomplished by another means, administrative controls will be implemented to limit the degree of opening for the manual valves such that the total diversion does not exceed safety analyses margins. In conjunction with this, it may be necessary to increase the minimum boron concentration in the SI header to satisfy DNB acceptance criteria. SCE is still evaluating this issue and will implement additional controls as necessary to allow continued use of the bypass valves during operation.

5. Electrical Bus Voltage Dip - The Cycle 10 reload analysis included provisions to account for this voltage dip. The period of time from which a safety injection signal is initiated and water is delivered to the core was revised to account for the potential delay in opening MOV-850 A and B, and closing MOV-20, 21 and 22.

6. Delay in Actuation of CV-875 A/B Limit Switches - No corrective actions are necessary because the voltage dip in Item 5 above occurs concurrently with this delay. Thus, diversion of SI flow due to this issue is inconsequential.

In addition to the above corrective actions, all drawings affected by these issues will be revised as necessary to reflect the proper configuration.

SAFETY SIGNIFICANCE: Each of the items associated with this issue either contributes to the delay or results in unanalyzed diversion of SI flow to the reactor during a LOCA or MSLB. Each of the items was evaluated for safety significance when it was identified. Because not all of the items were identified concurrently, the overall safety significance of these items occurring concurrently was not previously evaluated. Prior to Cycle 10, the LOCA analyses contained sufficient margin to allow a certain degree of SI flow diversion or delay in delivery of SI flow to the reactor. For example, when the electrical bus voltage dip item was identified, the existing transient analyses had sufficient margin to allow the associated delay in the alignment of SI valves. This margin, however, was not sufficient to account for the aggregate impact of all of the time delays and flow diversions occurring concurrently. Transient analyses have been revised to account for the voltage dip issue, but plant modifications and administrative controls were necessary to minimize the impact of the other issues. Prior to Cycle 10, the net impact of all of the delays or diversions occurring concurrently would have delayed reactor vessel refill and resulted in exceeding the acceptance criteria for Peak Cladding Temperature (PCT). Based on the corrective actions taken, however, this possibility no longer exists. Valve and wiring modifications have been completed, adminstrative controls have been implemented and reanalysis of the LOCA and MSLB transients with the most limiting time constraints for valve alignment demonstrates acceptable results including remaining within PCT and DNB limits. Therefore, the safety significance in the context of continued operation of SONGS 1 is low.

14. OVERPRESSURE MITIGATING SYSTEM SETPOINT

ISSUE: The Overpressure Mitigating System (OMS) setpoint specified in T.S. 3.20, "Overpressure Protection System" was found to be non-conservative. Specification 3.20 A(1) requires that two PORVs be operable with a setting $\leq$ 500 psig whenever RCS pressure is less than 400 psig and the pressurizer is greater than 50% level (Mode 4 and 5). The 500 psig setpoint was based on heatup and cooldown curves submitted in May 1984. Heatup and cooldown curves were subsequently revised in March 1986 to add safety margins for the closure flange region. However, the OMS setpoint, which was submitted prior to the heatup and cooldown curves submittal, was not re-evaluated to determine the continued applicability of the setpoint. The 500 psig setpoint also assumed a nominal PORV opening time of 2.0 seconds. Recent tests measured valve opening times of 1.97 seconds (CV-545) and 2.27 seconds (CV-546).

REFERENCES: NCR SO1-P-7160 (Rev. 0) dated April 4, 1989
NCR SO1-P-7161 (Rev. 0) dated April 4, 1989
LER 1-89-013 due May 4, 1989

DISCOVERY: In response to NRC Information Notice 89-32, Surveillance Testing of Low Temperature Overpressure Protection System, an NCR identified that the PORV opening time assumed in the design calculation for the OMS setpoint is not contained in plant procedures or Technical Specifications. Further review of the design calculation indicated that the OMS setpoint was based on heatup and cooldown curves existing prior to those which had been subsequently incorporated into the Technical Specifications.

ROOT CAUSE: The revision to the heatup and cooldown curves (T.S. 3.1.3) occurred in May 1986. However, the Technical Specification for the OMS (T.S. 3.20) was submitted in August 1977 but was not issued until May 1988. By letter dated April 5, 1988, the NRC sent to SCE a preliminary copy of Technical Specification pages affected by the OMS change for review prior to issuance. SCE's review did not identify the discrepancy between the OMS setpoint and the heatup and cooldown curves. T.S. 3.20 included a cautionary note to re-evaluate the OMS setpoint following any revision to the heatup and cooldown curves. Therefore, the revision to the heatup and cooldown curves pre-dated the issuance of the OMS Technical Specification. There was no mechanism to trigger the assessment of the impact of the change in Technical Specification (3.1.3) on another Technical Specification (3.20) which had been proposed, was undergoing extended NRC review, but had not been approved, issued and incorporated in the Technical Specifications.

SCE is currently evaluating possible programmatic deficiencies in the evaluation process of technical information (e.g., revisions to the heatup and cooldown curves), and the dissemination of that information to appropriate organizations for determination of impact on design basis requirements. For example, lack of recognition that the PORV stroke time must be maintained per the requirements of the revised heatup and cooldown curves resulted in this requirement not being incorporated into the IST program so that valve stroke time was not required to be measured.

CORRECTIVE ACTION: As an interim measure, the OMS setpoint will be revised to be consistent with the existing heatup and cooldown curves and conservative with respect to measured PORV opening times. Subsequently, Technical Specification changes will be submitted to revise the heatup and cooldown curves and take advantage of the margins available as discussed below to provide additional operating margin.

The established program for preparation of Amendment Applications will be reviewed and appropriate revisions made to ensure amendment preparers are alerted to this type of problem.

A review of the IST program requirements relative to design basis valve stroke times will be conducted to ensure that all valves requiring protected stroke times, including CV-545 and CV-546, are properly addressed in the IST program.

SAFETY SIGNIFICANCE: Further evaluation indicated that sufficient margins are available in the existing heatup and cooldown curves such that the current OMS setpoint would have provided adequate protection for the limiting low temperature overpressurization event as described below.

The change in heatup and cooldown curves in March 1986 added safety margins to account for the closure flange regions. The change was implemented by SCE in response to NRC review to meet Section IV.A.2. of to 10CFR50, Appendix G. The current margin in the Technical Specifications to account for the closure flange region is based on the inservice hydrostatic test pressure which is much lower than the preservice hydrostatic test pressure of 3750 psig which is the pressure required by the regulation. If the requirement of Section IV.A.2 is based on the actual preservice hydrostatic pressure of 3750 psig, the curves would not require margin to account for the closure flange region. This would provide additional margin for determining an acceptable low temperature PORV setpoint. The heatup and cooldown curves also contain a margin of 60 psig for instrument errors. Actual instrument uncertainty for the PORVs was evaluated to be 30 psig. A comparison of the margins available are summarized below:

<u>Pressure Limit For Low Temperature Overpressure Events</u>

1984 H/C limit at 100°F (Using curve as it would appear if preservice hydrostatic test pressure of 3750 psig were used to determine the need for margin in the closure flange region)	564 psig
Reduction in instrument errors (60 psig indicated on curves versus 30 psig actual)	30 psig
	594 psig

<u>Peak Pressure For Limiting Event</u>

OMS setpoint (current T.S. value)	500 psig
PORV Overshoot for limiting event (as currently assumed in analysis)	60 psig
Increase in Overshoot to account for 2.5 sec valve opening (versus 2 second opening time assumed in overshoot value above)	12 psig
	572 psig

Hence, peak pressure for the limiting overpressure event (572 psig) would have remained below the Appendix G limit (594 psig) with conservatism justifiably reduced for the heatup and cooldown curves and instrument uncertainty.

ENCLOSURE 2

SUPPLEMENTAL INFORMATION ON

KEY ASPECTS OF RESTART REPORT

ADEQUACY OF CORRECTIVE ACTION

1. The Restart Report should address the manner in which we identified the 12 specific issues, i.e., that they were identified by careful engineering reviews and were not chance discoveries. We should take credit for our enhanced engineering reviews.

RESPONSE

The recent reorganization of SCE's design and engineering organization, which consolidated SCE's design-related activities into one department and emphasized performance of work with a questioning attitude, was in large measure responsible for the discovery of a majority of the twelve issues (all except 2, 4, 8, and 12). The emphasis on a questioning attitude was applied during the engineering design process for plant design changes implemented during the Cycle 10 refueling outage, and during review of Technical Specification changes and new safety analyses required as a result of the Cycle 10 outage steam generator tube plugging. Hence, the near term effect of the reorganization and emphasis on quality design engineering and a questioning attitude has been and will continue to identify more reportable issues, since embedded flaws still exist and are being more efficiently and systematically discovered. The increase in the number of reportable issues is not symptomatic of a fundamental problem in plant design and operation. Rather the new initiatives are credited with finding problems before they become event-identified or self revealing.

In Section IV of the Restart Report the root causes common to the twelve issues identified in the report were summarized. It was concluded that the root causes were similar (with one addition) to those responsible for the engineering and technical support deficiencies identified in SCE's letter of October 3, 1988. Where additional aspects need to be addressed, follow-up actions were identified in Section IV and VII. Therefore, the root causes of issues identified in this report are being adequately addressed by the programmatic corrective actions already underway (i.e., reorganization, resources, DBD) and the followup actions.

As the programmatic corrective actions are implemented, one would expect to see a decline in the number of reportable items attributable to these root causes (i.e., should prevent same problems from reoccurring). However, in the near term, embedded existing flaws will continue to be detected.

OVERALL SAFETY SIGNIFICANCE

2A. What does this portend for the potential aggregate significance of undiscovered issues?

RESPONSE

Because the plant reanalysis efforts have been front-end loaded for the susceptible areas (e.g., electrical distribution systems analyses and event-specific single failure response evaluations), it is reasonable to expect that any as yet undiscovered issues will decrease in significance and frequency as the plant reanalysis efforts are completed. This is particularly evident in the single failure analysis area, where increasingly sophisticated analysis yields susceptibilities of decreasing probability.

Therefore, the potential aggregate significance of any undiscovered issues is expected to be bounded by those already identified and that the risk impact will generally follow a decreasing trend in magnitude.

2B. What is the aggregate significance of known deficiencies (i.e., with more than one issue occurring at a time)?

RESPONSE

Of the 12 Technical Issues addressed in the "SONGS Unit 1 Restart Report," and the two additional issues provided in Enclosure 1, 10 were shown to be acceptable as-is based on retesting, qualified alternate instrumentation, additional supporting documentation, reanalysis, or operation within existing analytical bounds or administrative controls. And, as such, the 10 issues posed no additional risk impact.

The remaining 4 issues were as follows:

- Technical Issue 5 - Failure of CCW Isolation Valves to RHR Heat Exchanger to Remain Closed.
- Technical Issue 8 - Failure of Refueling Water Pump G-27S to Auto-start (Jumper Wire)
- Technical Issue 9 - Failure of Containment Sphere Fire Loop Spray Valve CV-92 to Remain Closed
- Technical Issue 12 - Failure of a Diesel Generator to Load on SIS/LOP

Single Failure Evaluation

Of these items, Items 5, 8 and 12 are significant only with the single active failure of one electrical or pumping train. Item 9 (containment spray flow diversion), which involves the single active failure of a control circuit for a specific valve, does not cause refueling pump run-out and is otherwise independent of the other 3 items. As such, only Items 5, 8 and 12 can be credibly considered in combination within the single failure design basis of the plant.

Failure of one electrical train during a design basis primary or secondary pipe break inside containment could have caused loss of CCW flow (Item 5) and of containment spray with a concurrent loss of off-site power (Item 8), affecting recirculation mode and injection mode containment heat removal, respectively. Failure of the lead diesel generator during a design basis pipe break event with a loss of offsite power (Item 12) could conceivably have caused the necessary failure for Items 5 and 8.

Probabilistic Evaluation

The probable risk associated with Technical Issue 8, failure of RWP G-27S to automatically start, was not dependent on any of the other failures because operator action to ensure that both RWPs are running is a specific procedural step in the operative Emergency Operating Instruction. Furthermore, there is substantial conservatism in the already very small calculated risk.

The probable risk associated with Technical Issue 12, failure of a Diesel Generator to load on SIS/LOP, was not dependent on issues 5 and/or 9 because the operator's diagnosis of the load failure would occur within the first 5 minutes, long before the operators would be aware of issues 5 and/or 9. Subsequent recovery actions required by issue 12 are simple and do not interfere with operator diagnosis and actions to recover issues 5 and 9.

The risk associated with Technical Issues 5 and 9 was not dependent on issue 8 because resolution of issue 8 is achieved early in the scenario (less than 20 minutes), while operator response to issues 5 and 9 would not be expected in the first hour.

Technical Issues 5 and 9 could conceivably have occurred concurrently, given a LOCA requiring recirculation and no Loss of Offsite Power. However, the increased risk associated with these events occurring simultaneously is considered minimal for the following reasons: 1) the probability of simultaneous events is the product of two small probabilities, 2) the amount of time available for operator diagnosis and recovery is substantial and ample, and 3) the numbers used in the current analysis for operator response are very conservative.

Consequently, the aggregate of the known deficiencies compared to the same deficiencies evaluated separately does not significantly increase core melt risk.

SCHEDULE FOR RE-DOING THE NUS SINGLE FAILURE ANALYSIS

3. Is the nine month schedule for re-doing the NUS single failure analysis sufficiently aggressive? If it is, what can be done to prioritize the work such that high payback areas are addressed early in the process?

RESPONSE

While SCE will endeavor to improve on this schedule, it is expected that preparation and review of the ECCS SFA cannot be completed in significantly less than 9 months following restart, due to the following:

- Work Must Be Done In-House

  o In accordance with our previously initiated programmatic corrective actions, any effort of this type needs to be done by in-house personnel familiar with the unique aspects of SONGS 1, including extensive inter-disciplinary and Station review.

- Limited In-House Resources

  o There are presently only a limited number of personnel sufficiently familiar with SONGS 1 single failure analysis requirements to lead such an effort. Several of the personnel involved with the recent RPS SFA and ESF SFA efforts are no longer available.

  o The available personnel are presently required to support other essential tasks including update of the existing single failure analysis data bases to reflect Cycle 10 outage modifications.

  o Effective training of additional personnel for the SFA effort cannot begin until these other tasks have been completed.

- Major Scope Effort

  o The ECCS SFA will encompass several major fluid systems, their automatic actuation systems, and the electrical distribution and standby power system.

  o An ECCS SFA data base must be created to maintain the SFA as a living document and to support automated sort for the common power supply and interface device single failures.

The potential for prioritizing work is limited because:

- Similar to the RPS SFA effort, it will be necessary to perform the ECCS SFA in sequential, overlapping segments in order to adequately address single failure of common power supplies and interface devices.

ADEQUACY OF AUXILIARY ELECTRICAL POWER SYSTEMS

4. Is enough being done prior to restart to address electrical load issues? Should SCE do a sampling analysis of electrical load calculations?

RESPONSE

As a result of analyzing the impact of the Cycle X plant design changes and recent discovery of potential equipment overloads, the SONGS 1 auxiliary power system has been undergoing a thorough reevaluation for adequacy of steady-state and motor starting voltages, bus loading capabilities, and adequacy of equipment and cable ratings. Presently, the auxiliary power system voltage response and bus loading calculations, the diesel generator loading calculation and the 125 VDC battery calculations have been extensively revised/updated to verify that the electrical power system is capable of performing its post-accident functions under worst postulated load conditions. Additionally, calculations and evaluations to analyze short circuit conditions, cable derating, and bus loading and voltage regulation during normal plant operation are being updated and performed. The electrical calculations and evaluations which have been performed to date have revealed that the existing electrical distribution system is capable of supporting post-accident requirements. Since these calculations have evaluated the adequacy of the major system components and cable feeders under worst case conditions, any deficiencies revealed by the remaining electrical calculations and evaluations will be of low significance.

OVERALL EQ PROGRAM

5. What does the charging pump motor rewind issue imply about the overall EQ program? The report should better establish why this is an isolated occurrence.

RESPONSE

The charging pump rewind issue is considered to be an isolated anomaly found within an EQ program which is otherwise considered to be both comprehensive and accurate.

When Unit 1 was licensed, there were no specific criteria which required significant consideration be given EQ type issues, nor any programmatic needs relative to establishing and maintaining equipment in an EQ status. As a result, plant equipment repairs which met the then-existing criteria, did not have the level of documentation developed which meets current criteria needs. In this case a rewind was performed on a pump motor which was not reflected in its maintenance history nor otherwise physically identified on the equipment itself. Therefore, the rewind effort was not obvious when in January 1980 the NRC issued IE Bulletin 79-01B which specified EQ requirements for operating plants for the first time. In order to respond to the IEB, SCE contracted both Wyle Labs and Bechtel to develop EQ equipment lists and initiate determination of EQ status for each equipment item listed. Information was gathered and plant validation walkdowns performed by both groups and forwarded to the NRC in June, 1980. The charging pump motors were included within the scope of equipment requiring EQ. The NRC hired its own contractor, Franklin Research Labs, to evaluate the data provided by SCE. Franklin personnel visited the site in July 1980 to audit the documentation previously provided and discuss the EQ program. Additional submittals were prepared in 1981 to cover EQ equipment added in response to TMI, and to respond to questions raised by the NRC and Franklin Labs. SCE met with the NRC on several occasions and NRC Staff reviewed EQ documents to support issuance of an SER relative to EQ. In October of 1984 the NRC audited the SONGS 1 EQ program. A second EQ audit was held in September of 1985. Throughout these developments SCE reviewed available documentation to establish the EQ status for EQ equipment, and conducted several walkdowns to verify that the documentation matched equipment found in the field. None of the reviews, however, either by SCE and its contractors or by the NRC and its contractor uncovered the Charging Pump motor rewind.

Following discovery of this anomaly, SCE initiated a review of Operating logs and Non-Conformance Reports back through initial plant operation and confirmed that similar removal of equipment from service and the reasons for such removal from service had been explicitly documented, but that the charging pump motor rewind was the only such activity not previously accounted for in the EQ program. Pump and fan motors were considered to be the primary equipment of concern as it would have been more economical to replace other EQ equipment items (such as cable, transmitters or limit switches) rather than to repair them. The only other potential repair candidate, actuator motors, have already been scrutinized in response to previous NRC IEB and the MOVATS program.

The methodology for developing and controlling the EQ configuration of equipment as described below provides additional assurance that the overall EQ status of plant equipment is properly maintained.

Each piece of equipment which has environmental qualification (EQ) requirements must have an associated Environmental Qualification Data Package (EQDP). Among the items identified in the EQDP are any special installation requirements (e.g., the need to have an environmental seal assembly installed to protect the device), any relevant normal maintenance requirements (e.g., the type and periodicity or lubricant changeouts), and the replacement criteria identified. This information is summarized on an EQ Maintenance Information Sheet (EQMIS) which is finally transmitted to the SONGS Maintenance Group.

The maintenance engineer at San Onofre evaluates each MIS for its impact. If a procedural change is needed, the information is passed to the procedures group. If the MIS calls for repetitive maintenance activities, the information is passed to the planners for the preparation of an RMO. If the MIS requirements need clarification, the maintenance engineer prepares a memo to nuclear engineering requesting resolution of the issue. Nuclear engineering responds to the memo with a written clarification and/or a change to the MIS form, as required.

Records generated during EQ maintenance activities are retained in CDM along with the MIS for and the maintenance evaluation of the MIS.

These activities are covered by station procedure SO123-I-1.31.

If discrepancies are identified in the EQ maintenance activities, non-conformance reports are written. Nuclear engineering reviews or prepares the disposition to assure proper engineering cognizance.

OVERALL EQ PROGRAM (Continued)

In all cases the specific maintenance requirements associated with an EQDP are clearly addressed by maintenance and their policies assure continued compliance with EQ needs.

There are two areas where further programmatic EQ action will be required. When Reg Guide 1.97 commitments are finalized, that equipment will also require EQ. Additionally, EQ has been established predicated on Mode 3 as the Safe Shutdown condition. Any change in the assumption (i.e., Mode 4 or 5) will require an EQ review of equipment required to achieve and maintain that mode.

RESIDUAL HEAT REMOVAL PIPE WALL THICKNESS

6. Why is the ISI data base adequate to resolve the pipe schedule issue? If we can say that conclusions of seismic work would be unchanged if an analysis assumed schedule 120 in lieu of schedule 160 across the board, we should include such a statement in the report.

RESPONSE

NCR S01-P-6896 identified a discrepancy between various SCE design documents and the as-built condition regarding pipe schedule for the RHR suction line (line RCS-5002-8"-2501). The design documents indicated that this line was constructed of schedule 160 pipe while ultrasonic testing had revealed that the wall thickness corresponded to schedule 120 pipe.

An analysis was performed to determine the acceptability of schedule 120 pipe in this application. The analysis demonstrated that schedule 120 pipe satisfied all ASME Section III stress limits under all design loading conditions. Consequently, the design documents were revised to reflect the actual pipe schedule.

In addition to the RHR line, the stress analyses of all the other large bore lines (in piping material class 2501) connected to the primary loop, were evaluated to determine the acceptability of these lines should they be furnished as schedule 120 pipe. Again, the evaluation demonstrated that all ASME Code stress requirements would have been satisfied even if these lines had been fabricated out of schedule 120 pipe.

In addition to the above, all the Inservice Inspection records for ASME Code Class 1 and 2 large bore piping were reviewed to extract wall thickness data. This review retrieved wall thickness data for 9 out of 21 lines. With the exception of the RHR line, no discrepancies were found between the as-built condition and the design documents. This data suggests a high level of confidence that the RHR line documentation discrepancy was an isolated case.

Based on the above, it is concluded that the RHR line wall thickness discrepancy was an isolated case with no safety significance since the line satisfies all ASME Section III stress requirements under all design load conditions.

It should be noted that on March 29, 1989, this matter was discussed in a conference call between Messrs. D. F. Kirsch and C. Clark of the NRC and Messrs. M. A. Wharton, J. A. Mundis, and A. D. Sistos of SCE. As a result of the information presented by SCE, Mr. Kirsch indicated that he considered this matter closed.

POST-MODIFICATION AND POST-MAINTENANCE TESTING

7. Two of the items should have been caught by testing (the electrical jumper and sequencer defect). Should SCE include in the DBD Program an assessment of what testing has been done and whether or not additional testing is required.

RESPONSE

SCE's Design Bases Documentation (DBD) Program Plan (described in SCE's letter to the NRC dated January 9, 1989) does not address this issue directly. However, it is the intent of the DBD program to review applicable startup and surveillance testing to establish that system functions and key design parameters have been adequately demonstrated via testing. The process we intend to use to achieve this review is currently under development. As this process is finalized it will take into account the specifics of the items identified in the restart report.

It is noted that post-maintenance testing at SONGS 1 is formally controlled by Procedure SO1-XV-1.0, Retest Manual. This procedure is a comprehensive manual for post-maintenance testing, providing specific testing requirements and procedural cross-references for all normally encountered maintenance activities. SO1-XV-1.0 also provides the procedure to use to establish post-maintenance testing requirements for any maintenance activities not already covered by the manual. The testing requirements are established by Station Technical in conjunction with Maintenance and Operations. Operations must also approve the work authorization for maintenance activities, and would identify any additional testing required to establish operability of the affected systems and equipment.

In practice, the post-maintenance testing (including any additional operability testing) is equivalent to startup acceptance and preoperational testing for the affected functions.