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January 17, 1980

Director, Office Of Nuclear Reactor Regulation
Attention: Mr. D. G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Subject: Docket No. 50-206
Additional Information in Support of Responses to NRC TMI Requirements
San Onofre Nuclear Generating Station
Unit 1

By letters dated October 17, 1979, November 21, 1979 and December 14, 1979, we provided information concerning our plans to implement each of the NRC requirements established following the Three Mile Island accident. These letters were submitted in accordance with the NRC letters dated September 13, 1979 and October 30, 1979 and in response to telephone discussions with representatives of the NRC staff.

The purpose of this letter is to (1) advise you of our progress to date with respect to implementing each of the NRC requirements, (2) provide additional information in support of our previous commitments as required by your October 30, 1979 letter, (3) provide additional information in response to telephone discussions with representatives of the NRC staff and (4) provide our response to the NRC letter dated October 17, 1979 regarding the North Anna and related incidents. Accordingly, enclosed are forty copies of the report entitled "Additional Information in Support of Responses to NRC Requirements Established Following the Three Mile Island Accident, San Onofre Nuclear Generating Station, Unit 1, January, 1980."

As discussed in the enclosed report, the design details associated with those items which require prior NRC approval will be submitted in a timely manner so that approval and our implementation can be completed by the required implementation date. For those items which do not require prior

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NRC review and approval, our method of implementation will be documented and maintained on file available for NRC inspection as discussed in our November 21, 1979 letter.

If the information contained in the enclosed report is inadequate to allow a complete NRC review of our implementation methods, or you have any other questions concerning the enclosed report, please contact me.

Very truly yours,

KPBaskin/JH

Enclosures

ADDITIONAL INFORMATION IN SUPPORT
OF RESPONSES TO NRC REQUIREMENTS
ESTABLISHED FOLLOWING THE
THREE MILE ISLAND ACCIDENT

SAN ONOFRE NUCLEAR GENERATING STATION
UNIT 1

JANUARY, 1980

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INTRODUCTION

The information contained herein supports our responses to each of the actions required by NUREG-0578 as modified and/or supplemented by NRC letters dated September 13, 1979 and October 30, 1979. Our responses were provided by letters dated October 17, 1979, November 21, 1979 and December 14, 1979. Also contained herein are responses to NRC staff questions discussed during numerous telephone conversations and to the NRC letter dated October 17, 1979 regarding North Anna and related incidents.

The organization of this report corresponds to the enumeration of recommendations in our report entitled "Responses to NRC Requirements Established to Date Following The Three Mile Island Accident, San Onofre Nuclear Generating Station, Unit 1, October, 1979", submitted to the NRC by letter dated October 17, 1979.

SECTION 2.1.1

NUREG-0578 requires that motive and control components associated with the Pressurizer Relief (PORV) block valves be capable of being supplied from either offsite or onsite power. Our October 17, 1979 letter indicated that the motive and control components associated with the PORV block valves can be powered from either offsite power or the emergency diesel generators. However, we committed to provide a backup nitrogen pneumatic supply to the PORV block valves to ensure the functioning of the valves following loss of station instrument air. By letter dated November 21, 1979, we provided additional information concerning our plans to install the backup nitrogen pneumatic supply. As discussed with members of the NRC staff, the words "controls grade" were inadvertently omitted from that letter to fully describe this modification. Notwithstanding this omission, information was provided concerning our plans to implement a qualification program to qualify the equipment, if necessary, by January 1, 1981. As requested by members of the NRC staff during telephone conversations, we will submit a description of our qualification program, including the extent of the qualification required and a completion schedule. Since the information required to finalize the details of the qualification program will not be available until the modifications have been completed, we will submit the description of the qualification program in a timely manner as soon thereafter as is practicable.

In response to requests for additional information from the NRC staff during their review of our letters dated October 17, 1979 and November 21, 1979, the existing station instrument air system to the PORV block valves does not meet the single failure criteria and there are no instrument air accumulators dedicated to the PORV block valves. However, following completion of the modifications associated with the backup pneumatic nitrogen supply to the PORV block valves as discussed above, the PORV block valves will have a redundant source of motive power in the event of the loss of the station instrument air system. As discussed in Section 2.1.4 of this report, additional design modifications which are required to allow operation of the PORV block valves include:

1. Removing the containment isolation valve inside containment on the nitrogen supply line to the Pressurizer Relief Tank (source for the backup nitrogen pneumatic supply) and replacing it with a check valve and an associated normally open block valve to permit containment penetration leak rate detection and testing,

2. Removing the automatic containment isolation signal from the isolation valve outside containment while retaining remote manual isolation capability, and

3. Providing a nitrogen pneumatic supply to the isolation valve outside containment as a redundant source of motive power to prevent valve closure in the event of loss of the station instrument air system.

These design modifications will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980.

In addition to the above requirements, NUREG-0578 requires that procedures and training be established to make the operator aware of when and how to connect the required pressurizer heaters to the emergency buses in order to maintain natural circulation at hot standby conditions.

Our October 17, 1979 letter identified the pressurizer heater capacity required to maintain natural circulation and the time requirement for connecting them to the emergency diesel generators. Station procedure S-3.5.5, "Loss of Coolant", has been revised to incorporate these requirements.

SECTION 2.1.2

NUREG-0578 requires that testing be coordinated to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force submitted the report entitled "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," Revision 2, November 21, 1979. SCE endorces the EPRI program and considers the program to be responsive to the NUREG-0578 requirements. In accordance with the established NRC schedule, the test program will be completed by July, 1981.

By letter dated May 3, 1979, we provided our response to IE Bulletin 79-06A and Revision 1 thereto. Our response to Item 6 of that bulletin indicated that we would investigate the capability of the PORV block valves to close under high flow conditions. Our investigation included a record search to retrieve any information indicating that analysis and/or testing was done to verify that the isolation valves would close with the PORV's wide open at design temperature and pressure. We have not been able to locate any such information. However, as described in the aforementioned EPRI report, the valve test program is expected to include system tests which will determine the ability to isolate a stuck-open relief valve using its associated block valve.

SECTION 2.1.3.a

NUREG-0578 requires that pressurizer relief valves and safety valves be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe. Our letter dated October 17, 1979 indicated that a positive valve position indication (either using stem mounted limit switches or an acoustic device) would be provided in the Control Room for the Power Operated Relief Valves (PORV), their associated block valves and the safety valves. Our letter dated November 21, 1979 provided additional information concerning installation of the valve position indication. However, as discussed with members of the NRC Staff during several telephone conversations, the words "controls grade" were inadvertently omitted from that letter to fully describe our plans for these modifications. Notwithstanding this omission, information was provided concerning our plans to implement a qualification program to qualify the equipment, if necessary, by January 1, 1981. As requested by members of the NRC staff during telephone conversations, we will submit a description of our qualification program, including the extent of the qualification required and a completion schedule. Since the information required to finalize the details of the qualification program will not be available until the modifications have been completed, we will submit the description of the qualification program in a timely manner as soon thereafter as is practicable.

As discussed with members of the NRC staff during telephone conversations, stem mounted limit switches powered from vital buses will be provided on the PORV's, their associated block valves and the safety valves with a position indication for each valve in the control room rather than providing an acoustic device. In addition, single alarms (two) will be provided in the control room to indicate if either of the PORV's (one alarm) or safety valves (one alarm) are open (no alarm will be provided for the PORV block valves since they are normally open).

The design modifications discussed above will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980.

In addition to the positive valve position indication to be provided as discussed above, other methods of determining valve positions are available and are discussed in the emergency procedures as an aid to operator diagnosis and action. These methods and procedures are discussed in our May 3, 1979 response to Item 6 of IE Bulletin 79-06A and Revision 1 thereto.

SECTION 2.1.3.b

NUREG-0578 included requirements for instrumentation, procedures, and training necessary to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling. These requirements were to be addressed in two stages. The first was to be based on detection of reduced coolant level or the existence of core voiding with the existing station instrumentation. The second stage was to study and develop system modifications that would not require major structural changes to the station and that could be implemented in a relatively rapid manner to provide more direct indication than that available with existing instrumentation. This study was specifically required to address vessel level detectors.

Existing Instrumentation

The Westinghouse Owners' Group, of which SCE is a member, has performed analyses as required by Section 2.1.9 of NUREG-0578 to study the effects of inadequate core cooling. These analyses were provided to the NRC for review by letter dated October 31, 1979. In addition, procedural guidelines to recognize inadequate core cooling entitled "Instruction to Restore Core Cooling During a Small LOCA" accompanied the October 31, 1979 letter. The guidelines provide the basis for procedural changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. Based on these guidelines, Operating Instruction S-3-4.2, "Inadequate Core Cooling", has been implemented for San Onofre Unit 1.

The existing instrumentation relied upon at San Onofre Unit 1 includes:

1. Wide range core exit thermocouples
2. Excore neutron detectors
3. Steam generator pressure and level
4. Reactor coolant pump motor currents, vibration, and flow
5. Reactor coolant system temperatures and pressure

New Instrumentation

Installation of a primary coolant saturation recorder has been completed. Operating Instruction S-3-4.2 includes instructions for its use and for utilizing a backup curve based on the steam tables. The alarm setpoint of $50^{o}F$ subcooling, as discussed in our letters dated October 17, 1979 and November 21, 1979, has been revised to $40^{o}F$ to conform with the Westinghouse Owner's Group recommendations and to prevent alarm annunciation during normal full load operation (full load margin to saturation is about $44^{o}F$).

After examining many different methods and principles for determining the water level in the reactor vessel, the Westinghouse Owners' Group indicates that a basic delta pressure measurement system from the bottom to the top of the vessel appears to provide the most meaningful and reliable information to the operator. One of the reasons for choosing this system is that the sources of potential errors are better known for this system than for any other new or untested system.

The system would utilize a connection at the top of the reactor vessel (vent line or spare rod control cluster mechanism penetration) and a connection at the bottom of the vessel (incore flux monitoring thimble). However, the system is not feasible for stations such as San Onofre Unit 1, which have top-mounted incore flux instrumentation systems.

No other new instrumentation to provide more direct indication of inadequate core cooling has been identified by the Westinghouse Owner's Group which could be utilized at San Onofre Unit 1. Therefore, existing instrumentation, including the primary saturation recorder, as discussed in our station procedures will be utilized to fulfill the NUREG-0578 requirement.

SECTION 2.1.4

NUREG-0578 establishes requirements for containment isolation on diverse signals, review of isolation provisions for non-essential systems, and design review and modification, as necessary to eliminate the potential for inadvertent reopening upon reset of the containment isolation signal. Our letters dated October 17, 1979, November 21, 1979 and December 14, 1979 provided our commitments to implement the requirements for the diverse containment isolation signal and the prevention of automatic reopening of containment isolation valves. However, as discussed with members of the NRC staff during telephone conversations, the words "controls grade" were inadvertently omitted from our November 21, 1979 letter which further described our commitments with respect to the containment isolation provisions. Notwithstanding this omission, our November 21, 1979 letter described our plans to implement a qualification program to qualify the equipment, if necessary, by January 1, 1981. In accordance with a telephone request by members of the NRC staff, we will submit a description of our qualification program, including the extent of the qualification required and a completion schedule. Since the information required to finalize the details of the qualification program will not be available until the modifications have been completed, we will submit the description of the program in a timely manner as soon thereafter as is practicable.

Our December 14, 1979 letter clarified and revised our previous commitments contained in our October 17, 1979 and November 21, 1979 letters with respect to eliminating reliance on administrative controls to prevent automatic reset of containment isolation valves. The method of eliminating reliance on administrative controls will be as follows:

1. Upon receipt of a safety injection signal or a containment high pressure signal, all automatic containment isolation valves will close.

2. The ability to reopen any or all of these isolation valves will be dependent on the following prerequisites:

   a. Containment pressure as seen by both redundant pressure switches must be below the actuation set point of two psig.

   b. Both Safety Injection System load sequencers must be reset, if they were actuated.

   c. The containment isolation valve control switches, for all valves which could automatically reopen following containment isolation signal reset, must be placed in the closed position.

   d. Both containment isolation reset switches must be actuated for system reset. If any of the above three prerequisites are not met, actuation of the reset switches will have no effect.

The design modifications discussed above will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980. The completion of these design modifications will represent compliance with all technical requirements established by the NRC for containment isolation provisions, except that associated with the opening of automatically actuated containment isolation valves individually following reset and/or override of the containment isolation signal. As discussed in our December 14, 1979 letter, this capability may not be completed prior to the Fall, 1981 refueling outage based on an estimated procurement lead time of eighteen months for containment electrical penetrations. As subsequently discussed with the NRC Staff, we believe that lead time may be reduced to twelve months. In any event, we are proceeding with engineering of modifications to permit individual valve reopening following overriding of the containment isolation signal, and will advise the NRC Staff of any possible improvement in the implementation schedule of this additional modification. Notwithstanding the fact that we are proceeding with engineering of modifications to permit the reopening of all automatically actuated containment isolation valves independent of containment pressure, we believe that the individual reopening capability need only be provided for selected systems, as discussed below:

We have reevaluated our containment isolation provisions giving careful reconsideration to the definition of essential and non-essential systems as required by NUREG-0578. The reevaluation included a review of those lines penetrating containment which are not automatically isolated and those that are automatically isolated by a containment isolation signal. The results of the reevaluation are provided below:

1. Lines Not Automatically Isolated

   The initial review with respect to lines not automatically isolated by a containment isolation signal (i.e., essential systems) was provided by letter dated June 25, 1979 as supplemental information in response to IE Bulletin 79-06A and Revision 1 thereto. The supplemental information included a list of those systems which are not automatically isolated and the bases for not providing automatic isolation. The reevaluation of this information has substantiated our previous assessment of these systems.

   The following additional information is provided regarding "lines associated with engineered safety features" and "lines which are normally closed during operating modes requiring containment integrity and remain closed following an accident." This information was requested by NRC letter dated November 17, 1979 and referred to those categories of lines identified in our June 25, 1979 submittal as lines not automatically isolated.

   "Lines associated with engineered safety features"

   - Safety Injection Recirculation from containment
   - Reactor coolant charging/hot leg injection
   - Safety injection to loop A
   - Safety injection to loop B
   - Safety injection to loop C
   - Sphere spray supply line

- Alternate hot leg safety injection line

- Containment pressure sensing lines for containment isolation and containment spray initiation (four sensing lines)

"Lines which are normally closed during operating modes requiring containment integrity and remain closed following an accident"

- Containment integrated leak test pressure sensing line

- Containment integrated leak test line from reference chambers

- Instrument air supply line for containment integrated leak test pump back verification system

- Nitrogen supply line to penetration leak test header inside containment

- Safety injection purge line to refueling water storage tank

- Main steam line bellows penetration leak test/pressure balance system

- Fuel transfer tube

2. Lines Automatically Isolated

As discussed above, our June 25, 1979 letter identified those systems which are not automatically isolated and the basis therefore. The systems penetrating containment which are automatically isolated are controlled by individual switches or gang switches as follows:

a. Systems controlled by individual switches include:

   - Reactor Coolant Loop B & C and Residual Heat Removal Samples

   - Pressurizer Liquid and Steam Space Samples

   - Pressurizer Relief Tank Gas Space Sample

   - Nitrogen Supply to the Pressurizer Relief Tank (automatic isolation signal will be removed as discussed below)

   - Primary Water Makeup to the Pressurizer Relief Tank

   - Nitrogen Supply to the Reactor Coolant System Drain Tank

   - Turbine Plant Cooling Water Supply to Containment Sphere (automatic isolation signal will be removed as discussed below)

- Turbine Plant Cooling Water Return from Containment Sphere (automatic isolation capability will be removed as discussed below)

- Containment Sphere Equalizing Line

b. Systems controlled by gang switches include (switch identification shown in parentheses):

- Containment Sphere Purge Air Supply (CS-1)

- Containment Sphere Purge Air Outlet (CS-1)

- Containment Sphere Sump Pump Discharge (CS-3)

- Reactor Coolant System Drain Tank Discharge (CS-3)

- Reactor Coolant System Drain Tank Vent (CS-3)

- Containment Sphere Atmosphere Sample to Operational Radiation Monitoring System Channels 1211 and 1212 (CS-3)

- Steam Generator A Steam Sample (CS-3)

- Steam Generator B Steam Sample (CS-3)

- Steam Generator C Steam Sample (CS-3)

- Steam Generator A Blowdown Sample (CS-3)

- Steam Generator B Blowdown Sample (CS-3)

- Steam Generator C Blowdown Sample (CS-3)

- Containment Sphere Service Air (CS-3)

- Containment Sphere Service Water (CS-3)

- Containment Sphere Instrument Air Vent (CS-3)

- Containment Sphere Vent (CS-3)

Based on our reevaluation of these systems, the turbine plant cooling water supply and return lines and the nitrogen supply line to the Pressurizer Relief Tank have been identified as essential systems. The turbine plant cooling water supply and return lines may be required to support extended operation of the reactor coolant pumps since they supply cooling water to the reactor coolant pump enclosure air conditioning units. The nitrogen supply line to the Pressurizer Relief Tank currently provides a redundant pneumatic motive power source to the Power Operated Relief Valves and will provide a similar function to their associated block valves as discussed in Section 2.1.1 of this report. Accordingly, these lines will be modified as follows:

1. Turbine Plant Cooling Water Supply and Return Lines

   The automatic containment isolation signal will be removed from the isolation valves. However, remote manual isolation capability will be retained for these valves. These design modifications will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980.

2. Nitrogen Supply Line to the Pressurizer Relief Tank

   As discussed in Section 2.1.1 of this report, the isolation valve inside containment will be removed and replaced with a check valve and an associated block valve to permit containment penetration leak rate detection and testing. In addition, the automatic containment isolation signal will be removed from the isolation valve outside containment with remote manual isolation capability being retained for this valve. A nitrogen pneumatic supply will also be provided for this valve as a redundant source of motive power in the event of a loss of the station instrument air system.

Subject to valve availability for the nitrogen supply line, these design modifications will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980.

Of the remaining systems which are automatically isolated on a containment isolation signal, the capability to override the CIS pressure signal and reopen the associated isolation valves individually should be provided for the following systems:

System	Basis
Containment Sphere Equalizing Line and Containment Sphere Vent	Possible long term recovery aid to containment atmosphere clean up.
Steam Generator Steam and Blowdown Samples	Possible diagnostic aid for steam generator primary-secondary integrity and determination of offsite release calculations.

Those systems which should not be provided with the capability to override the CIS pressure signal and reopen the associated isolation valves individually are as follows:

System	Basis
Containment Sphere Sump Pump Discharge	Not required for possible long term recovery or diagnostic aid.
Containment Sphere Service Air	Not required for possible long term recovery or diagnostic aid.

Reactor Coolant System Drain Tank Discharge	Not required for possible long term recovery or diagnostic aid.
Primary Water Makeup to the Pressurizer Relief Tank	Not required for possible long term recovery or diagnostic aid.
Containment Sphere Atmosphere Sample to Operational Radiation Monitoring System Channels 1211 and 1212	Could result in high personnel exposures during sample collection (as discussed in Section 2.1.8.a, a new sample station will be constructed to obtain post-accident Containment Atmosphere Samples)
Reactor Coolant Loop B & C and Residual Heat Removal Samples, Pressurizer Liquid and Steam Space Samples and Pressurizer Relief Tank Gas Space Sample	Could result in high control building radiation levels (as discussed in Section 2.1.8.a, a new sample station will be constructed to obtain Reactor Coolant System samples)
Nitrogen Supply to the Reactor Coolant System Drain Tank	Not required for possible long term recovery or diagnostic aid
Containment Sphere Purge Air Supply and Outlet	Not required for possible long term recovery or diagnostic aid
Reactor Coolant System Drain Tank Discharge and Vent	Not required for possible long term recovery or diagnostic aid
Containment Sphere Service Water	Not required for possible long term recovery or diagnostic aid
Containment Sphere Instrument Air Vent	Not required for posible long term recovery or diagnostic aid

As discussed above, we are proceeding with engineering of modifications to permit individual valve reopening for all automatically actuated containment isolation valves as now required by the NRC Staff. However, as also discussed above, we believe that capability is not required for all valves. Prompt NRC Staff review of alternate plans to provide the reopening capability for only those systems identified above as having some possible post-accident function is respectfully requested.

We are also proceeding with engineering of modifications in conjunction with the individual valve reopening effort described above to provide electrical separation for isolation valves inside and outside containment (i.e., inside isolation valves would be electrically independent from outside isolation valves).

In conjunction with our reevaluation discussed above, we have reviewed the Westinghouse Owner's Group's report entitled, "Classification of Lines Penetrating Containment and a Review of Containment Isolation Logic and Philosophy." The report presents the results of a generic study to define essential/non-essential systems, including those "desireable" for long term recovery or diagnostic aid, relative to containment isolation. Based on our review of the generic study, our definition of (1) essential versus non-essential systems, and (2) those systems to be provided with the capability to open automatically actuated containment isolation valves individually is consistent with the containment isolation logic and philosophy contained therein applied on a station specific basis.

SECTION 2.1.5.a

NUREG-0578 requires that the existing purge system used for post accident combustible gas control of the containment atmosphere have a containment isolation system that is dedicated to that service only, that meets redundancy and single failure requirements and that is sized to satisfy the flow requirements. Our October 17, 1979 letter indicated we were considering (1) upgrading the existing purge system, and (2) installing hydrogen recombiners inside or outside containment as the method for meeting the NUREG-0578 requirements.

As discussed in our October 17, 1979 letter, Westinghouse has completed the study to determine post accident hydrogen generation inside containment. The results of the study indicate that control of hydrogen could be required in about two days assuming five percent clad interaction (as specified by 10CFR50.44 for zircaloy clad cores). Based on the results of the Westinghouse study and our evaluation of the two methods discussed above, redundant safety grade hydrogen recombiners will be installed inside containment to meet the NUREG-0578 requirements. However, if detailed design efforts indicate that installation of the recombiners is not feasible (i.e., accessibility problems), then another method to meet the NUREG-0578 requirements will be implemented.

Combustible gases inside the Reactor Coolant System will be controlled by use of the Reactor Coolant System High Point Vents discussed in Section 3.2 of this report.

As discussed in our October 17, 1979 and November 21, 1979 letters, the installation of hydrogen recombiners inside containment will be deferred pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

SECTION 2.1.6.a

NUREG-0578 requires an immediate and continuing program for leak reduction and measurement for systems outside containment which could contain highly radioactive fluids during post accident conditions. The October 30, 1979 NRC letter provided clarification of the NUREG-0578 requirements and included a new requirement to consider leakage from potential release paths due to design and operator deficiencies as discussed in the October 17, 1979 NRC letter to all operating plants regarding North Anna and related incidents. In addition, a list of those systems which are excluded from the program was requested.

We have completed leak reduction measures and leak testing of those systems outside containment which may be required to function in the course of achieving cold, stable conditions following an accident and could contain highly radioactive fluid. This program follows the guidelines and recommendations prepared by the Westinghouse Owner's Group, as they apply to San Onofre Unit 1.

The Westinghouse Owner's Group recommends that priority be given to portions of five separate systems that may be required to function in the mitigation or short-term recovery phases following an accident. These five systems are:

1. Safety Injection System
2. Containment Spray System
3. Chemical and Volume Control System
4. Residual Heat Removal System
5. Boron Recycle System

Systems 4 and 5 above are not applicable to San Onofre Unit 1. The Residual Heat Removal System is located totally inside containment and there is no Boron Recycle System. Thus, our testing program included applicable portions of Systems 1, 2 and 3 above. In addition, we included portions of the reactor coolant sample and containment atmosphere sample systems.

The extent and results of our leak testing are provided below:

Liquid Systems (Systems 1, 2 and 3 above and Reactor Coolant Sampling System)

All Safety Injection Recirculation and Containment Spray System piping and components outside containment which could contain post accident radioactive fluids are currently limited to less than 625cc/hour equivalent leakage by existing Specification A(4) of station Technical Specification 3.3.1., "Leakage". This limit ensures that the combined 0-2 hour Exclusion Area Boundary thyroid dose due to recirculation loop leakage and containment leakage will not exceed the limits of 10CFR100. As required by Specification II.C of station Technical Specification 4.2, "Safety Injection and Containment Spray System Periodic Testing", these systems are visually inspected for leakage at intervals not to exceed the normal station refueling outage. In addition, the Specification requires that pumps and valves of these systems which are used during normal operations be visually inspected for leakage at intervals not to exceed once every six months. If leakage can be detected, measurements of such leakage are made. A refueling interval leak test of these systems was last performed on October 29, 1978. The equivalent leakage measured at that time has 175.4 cc/hour. A six month interval test of those pumps and valves which are used during normal operation was last performed on September 26, 1979. The Equivalent leakage measured at that time was 180.5 cc/hour.

Portions of the Chemical and Volume Control System and Reactor Coolant Sampling Systems which might be utilized during post accident conditions were visually inspected/leak tested during December, 1979. After minor adjustments and repairs, no measureable leakage was detected. The inspected/tested portions of the Safety Injection Recirculation, Containment Spray and Chemical Volume and Control Systems are shown on the attached Piping and Instrumentation diagrams. (See Appendix 1.)

Therefore, the total leakage from liquid systems outside containment which could carry radioactive fluids following an accident has been reduced to as-low-as practical levels. The total leakage from liquid systems outside containment is below that required to ensure that offsite thyroid doses will not exceed the limits of 10CFR100.

Gaseous Systems (System 3 above and Containment Atmosphere Sampling System)

Those portions of the Gaseous Radwaste System which would be utilized to direct gases vented from the Volume Control Tank to the Waste Gas Decay Tanks and the Containment Atmosphere Sampling System were also visually inspected/ leak tested during December, 1979. The boundaries for this inspection/testing are shown on the attached diagrams. The inspection/testing was accomplished via a two-stage process with repairs and/or adjustments made as practical to reduce leakage to as-low-as practical values. The first phase consisted of testing system piping connections, valves, joints, and fittings using a liquid soap solution. A total of seven sources of leakage were identified in this manner and repaired or adjusted until no leakage was observable. The second phase consisted of introducing Helium through various connections and "sniffing" system piping connections, valves, joints and fittings. Six additional leaks were identified in this manner and subsequently repaired.

Station Operation Instruction S-3-3.26, "Leakage Test of Radioactive Systems Outside of Containment", has been implemented to perform a visual inspection/leak test of the liquid and gaseous systems discussed above at refueling intervals. The procedure utilizes a checkoff list and data sheet for potential leakage points on the affected systems. Maintenance Procedure S-I-1.71, "Maintenance of Auxiliary System Outside the Containment", has been implemented to keep these systems as leak tight as practical.

Station Operation Instruction S-3-2.40, "Post Accident Operation of Radwaste System", and an associated checkoff form have been implemented to limit the liquid and gaseous system boundaries as shown on the diagrams prior to placing the Letdown System in service following an accident. Systems and partial systems which were excluded from our program are listed below:

1. Liquid Radioactive Waste System, including Containment Sphere Sump and Reactor Coolant System Drain Tank Discharges. (Basis: Lines entering Liquid Radwaste System from containment are automatically isolated on a containment isolation signal. Systems will not be used during post-accident conditions.)

2. Pressurizer Relief Tank Gas Space, Volume Control Tank Gas Space, and Charging Line Samples. (Basis: Pressurizer Relief Tank Gas Space line is automatically isolated on a containment isolation signal. None of these sample systems will be used during post-accident conditions.)

3. Chemical and Volume Control System Demineralizers, and Reactor Coolant Filter. (Basis: Systems will be valved out locally prior to initiating Letdown System flow during post accident conditions).

4. Portions of the Gaseous Radioactive Waste System, including the Cryogenic Waste Gas Treatment System, and piping connections with several parts of the Liquid Radioactive Waste System. (Basis: Systems will be valved out locally prior to initiating Letdown System flow during post accident conditions).

Using only the systems marked on the attached diagrams, it would be possible to recirculate fluids and hold miscellaneous gases released from the Reactor Coolant System as discussed above. However, accessibility to use the systems discussed above would be limited by high radiation levels. As discussed in Section 2.1.6.b of this report, improved shielding will be provided to limit the radiation levels and allow personnel access.

In addition, the following design modifications are being evaluated in conjunction with the shielding review to limit the extent of the systems potentially containing high level radioactive fluids:

1. Provide a direct path from the Volume Control Tank Vent to the Radwaste Gas Surge Tank. This would eliminate the need to pressurize the flash tank.

2. Provide additional isolation valves to reduce the portions of Gaseous Radwaste System pressurized by radioactive fluids subsequent to an accident.

3. Provide modifications to pump the radwaste gas compressor discharge back to the containment.

4. Provide high point venting of the Reactor Coolant System inside containment as discussed in Section 3.2 of this report.

The implementation of any of these modifications will be deferred, together with any shielding requirements as discussed in Section 2.1.6.b of this report, pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

The NRC October 17, 1979 letter discussed an event at North Anna Unit 1 where, due to operator error, the Volume Control Tank was overpressurized. The Volume Control Tank relieved to the High Level Waste Drain Tank. The High Level Waste Drain Tank in turn relieved to the plant stack via an orifice. However, due to a construction error, the High Level Waste Drain Tank was never connected to the stack or the orifice installed and the radioactive gas vented directly to the Auxiliary Building. Additionally, the High Level Waste Drain Tank vented to the Low Level Waste Drain Tanks which were in turn vented to the Auxiliary Building. It is likely that, even if the High Level Waste Drain Tank vent had been properly connected, excessive venting into the Auxiliary Building would have occurred.

The importance of the venting pathway of the Volume Control Tank vent is greatly increased during post accident conditions due to high activity levels in the systems. We have completed a study to determine the venting pathways of all relief valves within the boundaries of systems outside containment which might contain highly radioactive fluids during post accident conditions.

The following is a list of relief valves within the boundaries of the systems as shown on the attached diagrams and the consequence of their functioning:

System	Valve No. & Description	Results
Chemical Volume and Control System	RV 226, Volume Control Tank Vent	Vents to Waste Gas Surge Tank via flash tank. See RV 75, below.
Chemical Volume and Control System	RV 289, Seal Water Return	Vents to Volume Control Tank. See RV 226, above.
Chemical Volume and Control System	RV 210, Letdown Header Relief	Vents to Volume Control Tank. See RV 226, above.
Chemical Volume and Control System	RV 259, Test Pump Relief	Vents to pump suction. No release of contents outside system.
Safety Injection System	RV 882, Safety Injection Recirculation	Relieves to Volume Control Tank. See RV 226, above.
Radioactive Waste System	RV 62, 73 and 74, Decay Tank Relief	Relieves to Waste Gas Surge Tank. See RV 75, below.
Radioactive Waste System	RV 101 and 80, Waste Gas Compressor Relief	Relieves to pump suction. No release of contents outside system.

System	Valve No. & Description	Results
Radioactive Waste System	RV 75, Waste Gas Surge Tank Relief	Vents to Hold Up Tanks, then to atmosphere outside Auxiliary Building. This is a potential release path.

The information above indicates that, except for RV 75, no relief path within the system boundaries shown on the attached diagrams could cause an unmonitored or unexpected release of radioactivity. Operation of RV 75 could result in such a release. The venting of the Volume Control Tank to the Waste Gas Surge Tank and subsequent failure of the relief valve to the Hold-Up Tank could result in a direct release of high radioactive fluids to the atmosphere via the Hold-Up Tank loop vent since it vents directly to the atmosphere through the roof of the Auxiliary Building.

A design modification is currently being considered which would reroute the Waste Gas Surge Tank relief valve vent to the station stack via the filters. Implementation of any design modifications will be deferred together with similar modifications discussed above, pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

SECTION 2.1.6.b

NUREG-0578 requires that a radiation and shielding design review be performed to determine if personnel occupancy may be unduly limited in areas which require access or if safety-related equipment may be unduly degraded by the radiation field during post-accident operations. Our October 17, 1979 letter indicated that such a review was being performed in accordance with the NUREG-0578 requirements. In addition, the design review was performed consistent with the clarification provided by the October 30, 1979 NRC letter.

A. Station Access

As part of the design review, certain station areas were identified where post accident diagnostic and long-term recovery activities may be required outside the control room. The activities which may be required include:

1. obtaining reactor coolant, station stack particulate and iodine cartridges and containment atmosphere samples,

2. operating the Letdown System to the Volume Control Tank (operable from the control room; however, certain equipment will be valved out locally prior to initiating letdown flow as discussed in Section 2.1.6.a and manual control of the Waste Gas System in the Auxiliary Building is required),

3. utilizing the Radiochemical and Chemical Analysis Laboratory, and

4. obtaining drawings and documents from the station Engineering Drawing Management Center.

As discussed below, the results of the design review indicate that additional shielding is required to perform some of these activities.

Performance of Activities 1 and 2

Access to perform these activities is required west of the containment sphere in the areas of the station stack, the roof of the Auxiliary Building and the Auxiliary Building. Access to these areas will be from the Administration and Control Building around the north side of the containment sphere. Modifications are required to shield the containment emergency air lock and piping penetration doghouse to reduce potential operator exposure from these sources. Additional shielding around the recirculation heat exchanger area, the recirculation piping tunnel, and the roof over the Waste Gas Decay Tanks is also required to reduce potential exposure. Areas inside the Auxiliary Building which require shielding to reduce potential exposure during operation of the Waste Gas System include the entrance to the Demineralizer alleyway and the valving area for the Waste Gas Decay Tanks. As an alternative to shielding modifications inside the Auxiliary Building and to the roof over the Waste Gas Decay Tanks, alternative design modifications as discussed in Section 2.1.6.a of this report are being considered to eliminate operation of the Letdown System and the necessity for operation of the Waste Gas System.

Performance of Activity 3

Access to the Radiochemical and Chemical Analysis Laboratory which is adjacent to the Control Room is possible without leaving the Administration and Control Building. Existing shielding for this laboratory is adequate to allow frequent access and to permit onsite analysis of radioactive samples without potentially interfering background levels of radioactivity.

Performance of Activity 4

Access to the Engineering Drawing Management Center where station records are stored will be from the Administration and Control Building through the north vital area access gate, then south along the east bluff to the Administration, Warehouse and Shop Building. Access to the Center will be minimized with the establishment of the Onsite Technical Support Center as discussed in Section 2.2.2.b of this report. However, post accident radiation levels would allow infrequent access without additional shielding.

For completeness, other areas of the station outside the Control Room have also been evaluated. The results of this evaluation are as follows:

1. Although no post accident access for operational or diagnostic purposes is anticipated, the current station shielding design would allow infrequent access to the 4 kV Switchgear Room, the Diesel Generator Building and the DC Battery Rooms.

2. Access would be restricted south of the containment sphere in the Turbine Building and in outside areas south of the Safety Injection Recirculation System piping. The restricted areas include the 480 Volt Room, the Motor Control Center 3 area, the Auxiliary Control Panel area, the Auxiliary Feedwater Pump area, the Safety Injection/Feedwater Pump areas, and the UPS Battery area. No post accident access for operational or diagnostic purposes is anticipated in any of these station areas.

With respect to the performance of activities in the Control Room and Technical Support Center, post accident radiation levels meet the dose exposure limits established in General Design Criteria 19. However, additional shielding is required on the north and west walls of the Control Room to maintain the dose level equal to or less than 15 mr/hour immediately following an accident as required by the October 30, 1979 NRC letter.

In order to facilitate NRC staff review of our radiation and shielding design review, the attached figures (see Appendix 2) show radiation zones within the station at various times with and without the additional shielding discussed above. Figures 1 through 8 show the radiation zones for the station as currently constructed. Figures 9 through 16 show the radiation zones for the station after the additional shielding as discussed above is installed. The time periods chosen were 0, 30 minutes, 1 day and 30 days after an accident. For the purposes of this review, actuation of Safety Injection Recirculation was assumed to occur between 0 and 30 minutes.

As discussed in our October 17, 1979 and November 21, 1979 letters, implementation of any design shielding modifications will be deferred pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program. The attached figures schematically show the extent of the affected areas and the required shielding.

B. Equipment Qualification

The qualification of safety-related equipment to operate in a post-accident radiation environment at San Onofre Unit 1 has been addressed by SCE in Amendment 30 to the Final Safety Analysis Report and again in letters dated February 24, 1978 and February 13, 1979. The letter submittals were provided in connection with the NRC's Systematic Evaluation Program. The completeness and acceptability of this information is scheduled to be reviewed by the NRC within the next several months. Implementation of any equipment modifications resulting from this NRC review will be incorporated into the integrated assessment of the Systematic Evaluation Program.

SECTION 2.1.7.a

NUREG-0578 requires automatic initiation of the Auxiliary Feedwater System. Our October 17, 1979 letter indicated our plans to implement the NUREG-0578 requirements and our November 21, 1979 letter provided the design details associated with providing automatic initiation of the Auxiliary Feedwater System.

Enclosure 1 to the NRC letter dated November 15, 1979 set forth (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant specific requirements applicable only to San Onofre Unit 1. These requirements, in part, duplicate the requirements of Section 2.1.7.a of NUREG-0578 to provide automatic initiation of the Auxiliary Feedwater System. The NRC November 15, 1979 letter requested that we evaluate each of the requirements contained in Enclosure 1 thereto with respect to the existing design and procedures for the San Onofre Unit 1 Auxiliary Feedwater System and advise the NRC of the results of the evaluation including associated schedules and commitments necessary to comply with the requirements. Our January 2, 1980 letter indicated that a response to Enclosure 1 of the NRC November 15, 1979 letter has been delayed and will be submitted by January 10, 1980.

All future correspondence related to the requirements of Section 2.1.7.a of NUREG-0578 will be associated with the NRC review of our responses to Enclosure 1 to the NRC November 15, 1979 letter.

SECTION 2.1.7.b

NUREG-0578 requires that indication of auxiliary feedwater flow to each steam generator be provided in the control room. Our letter dated October 17, 1979 indicated our plans to implement the NUREG-0578 requirements and our November 21, 1979 letter provided the design details associated with providing auxiliary feedwater flow indication in the control room.

As discussed in Section 2.1.7.a of this report, the NRC November 15, 1979 letter also duplicates the requirements of Section 2.1.7.b of NUREG-0578. Accordingly, future correspondence related to the requirements of Section 2.1.7.b of NUREG-0578 will be associated with the NRC review of our responses to Enclosure 1 to the NRC November 15, 1979 letter.

SECTION 2.1.8.a

NUREG-0578 establishes requirements to obtain and analyze (onsite) reactor coolant and containment atmosphere samples. Our October 17, 1979, November 21, 1979 and December 14, 1979 letters indicated our plans to implement the NUREG-0578 requirements.

In accordance with our commitments, a review of the Reactor Coolant and Containment Atmosphere Sample Systems has been completed. The results of the review and a description of required modifications are provided below.

Reactor Coolant Sample System

The existing reactor coolant sample station is located adjacent to the Control Room. If this station is utilized to obtain a pressurized sample, control room radiation levels could be as high as 10 R/hour. A preliminary evaluation indicates that the current size of the sample station will not accommodate the installation of shielding or remote handling devices to obtain and analyze pressurized reactor coolant samples in a timely manner.

Accordingly, a new sample station will be constructed such that a pressurized reactor coolant sample can be obtained and analyzed in a timely manner without jeopardizing control room habitability or exposing individuals to high levels of radiation. As discussed in our October 17, 1979 and November 21, 1979 letters, the construction of a new sample station has been deferred pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

Until the new sample station is completed, station procedures will be implemented to manually obtain an unpressurized reactor coolant sample from the Letdown System. As discussed in our December 14, 1979 letter, station procedures will also be implemented to obtain a pressurized reactor coolant sample from the existing sample station without adherance to NRC dose and time criteria. The station procedures will contain suitable precautions, based on specific station and containment radiation levels, to prevent obtaining a pressurized sample if excessive control room radiation levels or personnel exposure would result. Since the capability may not currently exist for onsite analysis of the samples, station procedures will also be implemented to prepare these samples for shipment offsite for analysis (including analysis of dissolved gases). The station procedures will be implemented prior to resumption of power operation following the outage currently scheduled in January, 1980.

Containment Atmosphere Sample System

Containment atmosphere samples are currently obtained from a manually operated station on the line supplying Operational Radiation Monitoring System Channels 1211 and 1212. The station is located on the north wall of the ventilation building where radiation levels could be as high as 10^4 R/hour. As discussed in Section 2.1.6.b of this report, additional shielding is required to gain access to the sample station without exposing individuals to high levels of radiation.

Accordingly, the new sample station being constructed to obtain a pressurized reactor coolant sample to permit dissolved gas analysis as discussed above will also include provisions to obtain and analyze a containment atmosphere sample under positive and negative pressures in a timely manner without exposing individuals to high levels of radiation. As discussed above, the construction of this new station has been deferred pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

Until the shielding has been completed, station procedures will be implemented to manually obtain a containment atmosphere sample from the existing sample station. As discussed in our December 14, 1979 letter, the station procedures will not adhere to the NRC dose and time criteria. However, the station procedures will contain suitable precautions, based on specific station and containment radiation levels, to prevent obtaining a sample if excessive radiation levels exist or excessive exposure would result. The station procedures will be implemented prior to resumption of power operation following the outage currently scheduled in January, 1980.

SECTION 2.1.8.b

NUREG-0578 requires an interim method for quantifying high level noble gas, radioiodine and particulate effluent releases from all potential release points if the existing instrumentation goes off scale until such time as extended range monitors and associated equipment can be installed. Based on a review of San Onofre Unit 1, two potential release points have been identified requiring interim methods for quantifying high level releases. The release points are: (1) the station stack (which includes containment, fuel storage and auxiliary building ventilation and condenser air ejector exhaust), and (2) the steam safety and atmospheric steam dump valves. A description of the interim methods is provided below.

Noble Gases

Station procedures will be implemented for noble gas effluents from the stack and steam dump valves to take in-situ radiation readings by an individual in communication with the Control Room every 15 minutes. Radiation readings will be taken using portable Teletectors or Xetex detectors shielded and directed at the stack sample line and/or steam dump valves headers if the existing effluent instrumentation goes offscale. Station procedures will also be implemented to convert these radiation readings to noble gas release rates.

In addition, a Technical Associates remote readout GM detector will be provided to improve the capability to obtain the in-situ readings at the stack sample line. Delivery of this instrument is expected by mid-1980. Installation of the detector with remote readout in the Control and Administration Building will be accomplished and station procedures will be developed for its use within sixty days following receipt.

Descriptions of the Xetex, Teletector, and Technical Associates instrumentation, including range, energy dependence, calibration frequency and technique are listed below:

	Xetex	Teletector	Technical Associates
Range:	.01 to 999 R/hr.	0 to 1,000 R/hr.	.01 to 100 R/hr
Radiation Detected:	Gamma	Beta & Gamma	Gamma
Energy Dependence:	$\pm$15% from 70 kev to 1.3 Mev	$\pm$15% from 100 kev to 3 Mev	$\pm$20% from 80 kev to 3 Mev
Calibration Frequency:	Once per quarter	Once per quarter	Once per quarter
Calibration Technique:	Check at 1, 9, 70, 250,800 and 1,000 R/hr. Instrument must be within $\pm$15%.	Minimum two points of radiation dose on each scale. Instrument must be within $\pm$15%.	Minimum two points of radiation dose on each scale. Instru. must be within $\pm$20%.

The Xetex and Teletector instruments rely on DC batteries which may not be capable of providing continuous readout for seven consecutive days as specified in the NRC October 30, 1979 letter. However, several instruments are available and sufficient spare batteries will be maintained onsite to meet this requirement.

The Technical Associates instrument relies upon AC power which will be obtained from normal distribution panels which may be supplied from the onsite diesel generators in the event of loss of offsite power.

Radioiodine and Particulate Effluents

Station procedures will be implemented for installation, removal and analysis of particulate and iodine samples from the existing stack sampling equipment adjacent to the station stack. Silver zeolite cartridges will be used to accurately determine iodine concentrations in the presence of large amounts of noble gas. The cartridges will be analyzed onsite using a Canberra Ge-Li analyzer. In addition, station procedures will be implemented to estimate the release rate of particulates and iodines from the steam dump valves. Main steam samples will be collected and analyzed onsite using a Ge-Li analyzer to determine iodine and particulate concentrations. Station procedures will be implemented to convert these concentrations into estimated release rates using precalculated conversion factors and steam discharge flow rate.

A Ge-Li analyzer is currently installed in the Radiochemical and Chemical Analysis Laboratory adjacent to the Control Room. This location provides adequate shielding to prevent potentially interfering background levels of radioactivity.

The stack sampling equipment and Ge-Li analyzer utilize AC power from normal distribution panels. The emergency diesel generators may be used to provide an alternate supply of AC power in the event of loss of offsite power.

The station procedures for noble gas, radioiodine and particulate sampling will cover all aspects of the measurement analysis, including minimizing radiation exposure, calculational methods for determining release rates, methods for dissemination of information, and calibration frequency/ technique. The station procedures will be implemented prior to resumption of power operation following the outage currently scheduled in January, 1980.

SECTION 2.1.8.c

NUREG-0578 and the clarification provided by the NRC letter dated October 30, 1979 require the capability to monitor iodine concentration in station areas where personnel may be present during post-accident conditions. Our October 17, 1979 letter indicated that equipment currently exists to determine the airborne iodine concentrations.

Existing portable AC powered air samplers with silver zeolite cartridges will be utilized for collection of airborne samples. The silver zeolite cartridges will be removed from the air samplers and analyzed on the existing station Ge-Li analyzer. The analyzer is located in the Radiochemical and Chemical Analysis Laboratory which is on a ventilation system without emergency air treatment capability. Station modifications to provide clean air to this laboratory will be deferred pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

Station procedures will be developed for utilizing the existing equipment to accurately determine airborne iodine concentrations during post accident conditions. The station procedures will be implemented prior to resumption of power operation following the outage currently scheduled in January, 1980.

Portable battery powered air samplers will be utilized to provide additional flexibility for collecting silver zeolite cartridge samples in those areas where AC power may not be conveniently accessible. Delivery of these new samplers is expected in February, 1980. Station procedures for use of these samplers will be developed and implemented within thirty days following receipt of the equipment.

AC powered air samplers with built-in single channel analyzers calibrated to detect I-131 will be provided. The capability will also exist to remove the sample cartridges from these samplers for laboratory analysis. Delivery of the new air cart mounted samplers/analyzers is expected in mid-1980. Station procedures for use of these air samplers/analyzers will be developed and implemented within thirty days following receipt of the equipment.

SECTION 2.1.9

NUREG-0578 requires analyses, procedures and training addressing small break loss-of-coolant accidents, inadequate core cooling and other transients and accidents and the submittal of the LOFT pretest calculations. Our October 17, 1979 letter indicated our plans to implement these requirements.

The information provided below describes the status of our efforts to date, via participation with the Westinghouse Owner's Group, with respect to meeting the NUREG-0578 requirements.

1. The small break loss-of-coolant analyses have been completed and were generally reported in WCAP-9600 which was submitted to the NRC by letter dated June 29, 1979. Incorporated in that report were guidelines that were developed as a result of generic small break loss-of-coolant accident analyses. These guidelines have been reviewed and approved by the NRC and were presented to the Westinghouse Owner's Group utility representatives in a seminar held on October 16 through 19, 1979. Revised station procedures based on these guidelines will be implemented and operator training will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980. A station specific small break loss-of-coolant accident analysis is being performed for San Onofre Unit 1 to confirm the applicability of the generic analyses and guidelines and is scheduled for completion by March 31, 1980. Any required revisions to station procedures will be implemented sixty days following completion of the station specific analysis.

2. Analysis related to the definition of inadequate core cooling and guidelines for recognizing the symptoms of inadequate core cooling based on existing station instrumentation and for restoring core cooling following a small break loss-of-coolant accident were submitted to the NRC by the Westinghouse Owner's Group by letter dated October 31, 1979. This analysis is a less detailed analysis than was originally proposed, and will be followed up with a more extensive and detailed analysis which will be available during the first quarter of 1980, as required. Station procedures and training based on the analysis submitted by the October 31, 1979 letter will be implemented prior to resumption of power operation following the outage currently scheduled in January, 1980.

3. With respect to other transients and accidents, the Westinghouse Owner's Group is performing an evaluation of the actions which occur by constructing sequence of event trees for each transient and accident. From these event trees, a list of decision points for operator action will be prepared, along with a list of information available to the operator at each decision point. Following this, criteria will be set for credible misoperation and time available for operator decisions will be qualitatively assessed. The information developed will then be used to test Abnormal and Emergency Operating Procedures against the event sequences and determine if inadequacies exist in these procedures. The results of this evaluation will be provided to the NRC by March 31, 1980, as required.

4. The Westinghouse Owner's Group has provided the results of the LOFT Pretest Calculations to the NRC. The results were provided by letter dated December 15, 1979.

SECTION 2.2.1.a

NUREG-0578 requires that (1) a management directive be issued emphasizing the primary management responsibilities of the shift supervisor for safe operation, (2) procedures be issued defining the duties, responsibilities and authority of the shift supervisor and control room operators, (3) procedures be issued for training the shift supervisor, and (4) procedures be issued outlining and limiting the administrative duties of the shift supervisor. Our October 17, 1979 letter indicated our plans to implement each of these requirements. Further information concerning the implementation of each of these requirements is provided below.

Management Directive

The Vice President of Nuclear Engineering and Operations has issued a management directive to meet the NUREG-0578 requirements by letter dated January 2, 1980. This directive will be reissued annually.

Operator and Shift Supervisor Duties

Station Order S-0-100, "Station Operations", and Station Operating Instruction S-0-4, "Watch Engineers Authority, Responsibilities and Duties," have been implemented to meet the NUREG-0578 requirements as they apply to the shift supervisor.

Station Order S-0-100, Station Operating Instructions S-0-3, "Control Operators Inspections and Duties", S-0-2, "Assistant Control Operators Inspections and Duties", and S-0-1, "Plant Equipment Operators Inspections and Duties" have been implemented to meet the NUREG-0578 requirements for control room operators.

Shift Supervisor Training Programs

Shift Supervisor training to meet the NUREG-0578 requirements will be completed prior to resumption of power operation following the outage currently scheduled in January, 1980.

Corporate Management Review of Shift Supervisor Administrative Duties

The Vice President of Nuclear Engineering and Operations has reviewed the administrative duties of the Watch Engineer, as specified in Station Operating Instruction S-0-4. As a result of this review, administrative functions not important to safe station operation have been identified and will be delegated to other personnel to meet the NUREG-0578 requirements. Revisions to Station Order S-0-100 and Operating Instruction S-0-4, as necessary to reflect this delegation, will be implemented prior to resumption of power operation following the currently scheduled outage in January, 1980.

SECTION 2.2.1.b

NUREG-0578 requires a Shift Technical Advisor to the Shift Supervisor. Our October 17, 1979 and November 21, 1979 letters indicated our plans to provide a Shift Technical Advisor.

In accordance with our commitments, a Shift Technical Advisor has been placed on duty. Station Operating Instruction S-0-6, "Shift Technical Advisor Duties" has been implemented to define the duties, responsibilities and authority of the Shift Technical Advisors.

SECTION 2.2.1.c

NUREG-0578 requires shift and relief turnover procedures to provide checklists to include critical station parameters, essential systems alignment and status, and maintenance or test status. A system to evaluate the effectiveness of the shift and relief turnover procedures is also required. Our October 17, 1979 letter indicated our plans to implement these requirements.

In accordance with our commitments, Station Operating Instruction S-0-5, "Operations Shift Relief, Logs and Responsibilities", has been implemented to meet the NUREG-0578 requirements. In addition, Station Order S-0-100, "Station Operations", has been implemented which includes requirements for a quarterly evaluation of the effectiveness of the shift relief procedures.

SECTION 2.2.2.a

NUREG-0578 requires procedures establishing the authority of the person in command to limit control room access and establishing a line of succession for the person in command. Our October 17, 1979 letter indicated our plans to provide procedures to implement this requirement.

In accordance with our commitment, Station Order S-0-100, "Station Operations", and Station Order S-A-103, "Control Room Access", have been implemented to meet the NUREG-0578 requirements.

SECTION 2.2.2.b

NUREG-0578 requires the establishment of an Onsite Technical Support Center. Our October 17, 1979 and November 21, 1979 letters indicated that such a Center has been established in the Visitor's Viewing Area and described the current capabilities of this Center. Station Emergency Procedure S-VIII-1.21, "Technical and Operational Support Center," has been implemented to describe the Center and provide instructions for its activation and use and for its maintenance in a "ready" status.

As requested by NRC staff members during telephone conversations, additional information further describing the capabilities of the Onsite Technical Support Center is provided below:

Communication

Communication links in the Onsite Technical Support Center include:

1) Direct line to NRC (Washington)

2) PAX phone (Company)

3) Bell telephone (outside separate line)

4) Direct line to San Clemente City Hall (Emergency Operations Center)

Records

The Onsite Technical Support Center currently contains a viewer and a complete set of station drawings on microfilm. Hard copies of piping and instrumentation diagrams, electrical elementaries and electrical one line diagrams are also readily available in the Control Room.

Data Analysis

Our planned methods for obtaining station data in the Onsite Technical Support Center is discussed in our November 21, 1979 letter. Distraction to and burden on the control room operators will be minimized by administrative control of the number of personnel admitted to the Control Room for removal of data. In addition, incore thermocouple and flux mapping data and some recorded information can be obtained from the area behind the control boards. A summary of control room recorded data which will be available for removal to the Onsite Technical Support Center is provided below:

1. NIS power, intermediate and source range recorders

2. NIS wide range power recorder

3. Events recorder

4. Volume control tank level

5. Pressurizer pressure and level

6. All three cold leg RTD's

7. All three vessel Tavg and Average Tavg.

8. RHR Heat Exchanger outlet temperature

9. Main steam and feedwater pressure

10. All three steam generator levels, feedwater flows and steam flows.

11. Control rod position indications

12. Letdown and charging flow

13. All three loop delta-T (behind panels)

14. All three RCP seal water leak off flow

15. ORMS & ARMS

16. Meteorological wind direction, speed, delta-T, and sigma

17. Tsat recorder (Thot, Pressure, Tsat)

18. ERMS

19. Pressurizer steam space temperature (behind panels)

20. All three RCP vibration (behind panels)

21. Boric acid tank level

22. RCS boric acid concentration (ppm) (behind panels)

23. % Axial offset (behind panels)

24. Feedwater temperature (behind panels)

25. Condensate flow (behind panels)

26. Condenser back pressure (behind panels)

27. Miscellaneous equipment bearing temperatures, generator gas, etc.

28. Turbine supervisory

The method discussed above will be used until the Technical Data Display and Transmit System can be installed in the Onsite Technical Support Center as discussed in our October 17, 1979 letter. We are currently evaluating the Westinghouse Owner's Group's recommendations for data acquisition and display parameters.

SECTION 2.2.2.c

NUREG-0578 requires establishment of an Onsite Operational Support Center. Our October 17, 1979 letter indicated that such a Center has been established on the first floor of the Control and Administration Building. Emergency Procedure S-VIII-1.21, "Technical and Operational Support Center" has been implemented to describe the Center and provide instructions for its activation and use and for its maintenance in a ready status.

SECTION 3.1.1 and 3.1.2

Our previous responses to these NRC requirements as contained in our letter dated October 17, 1979 do not require modification or clarification at this time.

SECTION 3.1.3

The NRC letter dated September 13, 1979 requires containment water level indication capability up to the elevation equivalent to a 500,000 gallon capacity. By letter dated October 30, 1979, the NRC revised this requirement to 600,000 gallons with provisions for deviations for older plants with smaller water capacities.

Our October 17, 1979 letter indicated our plans to provide containment water level indication up to the elevation equivalent to a 500,000 gallon capacity. Based on the NRC October 30, 1979 letter, the design of our wide range containment level indications will include the capability to monitor up to the elevation equivalent to a 600,000 gallon capacity. The installation of this indicator capability will be completed as discussed in our October 17, 1979 and November 21, 1979 letters.

SECTION 3.2

The NRC letter dated September 13, 1979 requires that high point vents be provided for the reactor coolant system. Additional clarification of this requirement was provided by the NRC letter dated October 30, 1979. Our October 17, 1979 and November 21, 1979 letters indicated our plans to implement the requirements for high point vents. In accordance with our commitments, the information provided below further describes the functional design of the venting system to vent non-condensible gases from the reactor coolant system during post-accident conditions at San Onofre Unit 1.

The venting system will include vents located on the reactor vessel head and the pressurizer capable of venting to the containment and/or the Pressurizer Relief Tank. The system will be safety grade and qualified commensurate with the requirements for the Reactor Coolant System pressure boundary.

Venting to the containment will terminate at a location where good mixing with the containment air and cooling of the vent gases is provided and will be accomplished through solenoid valves supplied with power from different vital buses. In addition, remote manual operation of the venting will be provided from the control room with valve position indication provided in the control room.

The system will be sized such that leakage from a postulated line break will be less than the capacity of one charging pump. Leak detection requirements are discussed in station Technical Specification 3.1.4, "Leakage". To minimize the probability of inadvertent actuation of the system, the design of the system will also include provisions such as removing power from the power-operated valves during normal operation.

The design details of the venting systems are being prepared. As discussed in our November 21, 1979 letter, the design details of the venting systems will be provided in a timely manner when they become available.

As discussed in our October 17, 1979 letter, implementation of the design modifications discussed above and the associated procedures for use will be deferred pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program.

APPENDIX 2

RADIATION ZONE FIGURES