RESPONSES TO NRC REQUIREMENTS

ESTABLISHED TO DATE

FOLLOWING THE THREE MILE ISLAND ACCIDENT

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 1

OCTOBER, 1979

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1. INTRODUCTION AND SUMMARY

Purpose

This report provides Southern California Edison Company's response for San Onofre Unit 1 to the September 13, 1979 letter from D. G. Eisenhut of the Nuclear Regulatory Commission to all operating nuclear power plants.

Scope

Section 2 of this report responds to all TMI-2 Lessons Learned Task Force Short Term Recommendations as contained in NUREG 0578 as modified by the September 13, 1979 letter identified above. The organization of this section corresponds to the enumeration of recommendations in NUREG 0578.

Section 3 of this report responds to additional recommendations for containment instrumentation, reactor coolant system vents, and emergency plan improvements as contained in Enclosures 3, 4 and 7 to the September 13, 1979 letter identified above. The organization of this section is consistent with the position requirements of Enclosures 3, 4 and 7.

Appendices 1 and 2 provide summaries of the design criteria and implementation schedules, respectively, for station modifications which are being implemented as discussed in Sections 2 and 3.

Southern California Edison Company was represented at the September 26, 1979 Las Vegas, Nevada Regional Meeting concerning TMI Short-Term Implementation Action. In addition, the Company was represented at a series of Topical Meetings concerning further clarification of the Onsite Technical Support Center, Shift Technical Advisor, Relief and Safety Valve Performance Testing, and Reactor Coolant System Venting held in Bethesda, Maryland October 10-12, 1979. The information contained in this report reflect consideration of the NRC guidance provided in those meetings.

2. RESPONSE TO NUREG-0578 RECOMMENDATIONS

Section 2.1.1 - Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWRs

A. Positions on Pressurizer Heater Power Supply

Position 1:

The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. All pressurizer heaters and associated controls can be supplied from either offsite power or the emergency diesel generators. In order to determine the minimum heater capacity and the time frame when the heaters must be available to maintain natural circulation, a study was performed by Westinghouse as authorized by the Westinghouse Owner's Group. The results of this study conservatively indicate that for the 1300 ft^3 pressurizer at San Onofre Unit 1 a heater capacity of 125 KW applied within four hours is adequate to maintain reactor coolant system pressure, thus keeping the primary coolant subcooled and providing core cooling via natural circulation.

At San Onofre Unit 1, each emergency diesel generator can separately and independently energize a normal pressurizer heater group having a capacity of 117 KW and a backup pressurizer heater group having a capacity of 482 KW. Accordingly, the power supply configuration at San Onofre Unit 1 assures that a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation can be connected in a manner that provides redundant power supply capabilty.

Position 2:

Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters. (Schedule: Complete implementation by January 1, 1980.)

Response:

Based on the results of the study discussed in response to Position 1 above, operating procedures and training will be implemented which makes the operator aware of when and how to energize the pressurizer heaters. A review of emergency power loads indicates that sufficient diesel generator capacity exists such that load shedding is not required.

The operating procedures and training will be implemented by January 1, 1980.

Position 3:

The time required to accomplish the connection of the preselected pressurizer heaters to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. All pressurizer heaters may be energized from the control room by manually operating the pressurizer heater control switches. As discussed in the response to Position 1 above, the required pressurizer heaters must be energized within four hours to prevent loss of subcooling. To provide additional conservatism for maintenance of natural circulation conditions, the procedure revisions discussed in response to Position 2 above will include a requirement to be able to energize the required pressurizer heaters within one hour. One hour provides sufficient time to perform the manual operation of energizing the pressurizer heaters.

Position 4:

Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. The design, construction and operational characteristics of the pressurizer heater interfaces are the same as those utilized for safety-related interfaces with the electrical buses.

B. Positions on Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

Position 1:

Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. The Pressurizer power operated relief valves (PORV's) (2) are spring loaded, normally closed type and are normally supplied with station instrument air to open. The station instrument air system can be powered from either offsite power or the emergency diesel generators. In addition, the PORV's have a backup pneumatic supply from the station nitrogen system to ensure their functioning on loss of station instrument air. The backup pneumatic nitrogen supply is from a pressurized source which does not rely on components depending on electrical power.

For each PORV, the pneumatic supply is controlled by a solenoid valve which is energized-to-open. The control circuits for these valves are supplied from independent vital buses. The vital buses can be supplied from either offsite power or the emergency diesel generators.

Position 2:

Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. The PORV block valves are spring loaded, normally open and require station instrument air to close. The station instrument air system can be powered from either offsite power or the emergency diesel generators. Although it is not required to meet the stated position requirement, a backup pneumatic supply from the station nitrogen system will be installed for each PORV block valve similar to that discussed in response to Position 1 above for the PORV's. This backup pneumatic supply will ensure the functioning of the valves following loss of station instrument air.

For each PORV block valve, the pneumatic supply is controlled by a solenoid valve which is energized-to-open. The control circuits for these valves are supplied from independent vital buses. The vital buses can be supplied from either offsite power or the emergency diesel generators.

The design criteria to be utilized for the modifications associated with the back up pneumatic supply are included in Appendix 1. The implementation schedule for the modifications is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by January 1, 1980. In addition, completion of the construction efforts is expected to require approximately one month of which the last two weeks require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence on January 1, 1980; construction will be completed during the first outage of sufficient duration, or during the next refueling outage which is now scheduled for March-April, 1980.

Position 3:

Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. The design, construction and operational characteristics of the PORV's and their associated block valves interfaces are the same as those utilized for safety-related interfaces with the electrical buses.

Control power to the PORV's and PORV block valves is provided such that one PORV and its associated block valve is fed from the same vital bus. The other PORV and its associated block valve is fed from a separate vital bus. Each of these vital buses can be supplied from offsite or emergency onsite power.

This "train" alignment configuration, in conjunction with the valve fail-safe positions (i.e., PORV's fail closed and block valves fail open), provides for both single failure protection and redundancy.

Position 4:

The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. All pressurizer level indication instrument channels are currently powered from the vital buses. These buses have the capability of being energized from either offsite power or the emergency diesel generators.

Section 2.1.2 - Performance Testing for BWR and PWR Relief and Safety Valves

Position:

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping and supports as well as the valves themselves. (Schedule: Submit program description and schedule by January 1, 1980 and complete test program by July, 1981.)

Response:

A full scale prototype qualification testing program of relief and safety valves under expected operating conditions for design basis transients and accidents will be undertaken on an industry-wide basis rather than by individual licensees. The Electric Power Research Institute (EPRI) is currently developing such a program. Southern California Edison Company will assist EPRI by providing financial support and technical assistance as requested through the Westinghouse Owner's Group. A program description and schedule for completion of the testing will be submitted prior to January 1, 1980.

Section 2.1.3.a - Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs

Position:

Reactor system relief and safety valves shall 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. (Schedule: Complete implementation by January 1, 1980.)

Response:

Both PORV's and their associated block valves have stem mounted limit switches which provide positive indications of the valve positions in the control room. These switches are not currently safety related and do not have alarm functions. The pressurizer safety valves do not presently have direct position indications. Currently, two alternative methods for providing safety related positive control room indications and alarms are under evaluation.

One method would utilize nonredundant safety related stem mounted limit switches on each of the valves with position indications in the control room. In addition, an alarm will be provided in the control room which would indicate if any of the PORV's or safety valves are not fully closed. No alarm function would be provided for the PORV block valves since they are normally open.

The other method would utilize a nonredundant safety-related acoustic device on the piping downstream of the PORV's and the safety valves with indication and alarm functions in the control room.

Based on the results of the evaluation, one of the above methods will be utilized. The design criteria to be utilized for the modifications are included in Appendix 1. The implementation schedule for the modifications is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by January 1, 1980. In addition, completion of the construction efforts is expected to require approximately one month of which the last two weeks require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence on January 1, 1980; construction will be completed during the first outage of sufficient duration, or during the next refueling outage which is now scheduled for March-April, 1980.

Sedtion 2.1.3.b - Instrumentation for Detection of Inadequate Core Cooling In PWRs and BWRs

Position 1:

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation conditions. Operator instruction as to use of this meter shall include consideration that it is not to be used exclusive of other related plant parameters. (Schedule: Develop procedures and describe existing instrumentation and install a subcooling meter by January 1, 1980.)

Response:

The Westinghouse Owners Group is currently in the process of developing generic procedure guidelines to enable the operator to recognize inadequate core cooling with existing instrumentation. All guidelines so developed will be appropriately incorporated into San Onofre Unit 1 procedures. The San Onofre Unit 1 procedure revisions are scheduled for completion by January 1, 1980, based on the current Westinghouse Owner's Group schedule for completion of the generic guidelines by October 31, 1979. As described in our June 25, 1979 letter in Docket No. 50-206, station operators have already been instructed regarding instrumentation available to detect core voiding and verify natural circulation.

A controls grade primary coolant saturation recorder is currently scheduled for installation prior to January 1, 1980.

This recorder will receive input from a pressurizer pressure transmitter and a hot leg RTD from any reactor coolant loop. The recorder will have switching capability to choose any one of the three hot leg loop temperature signals. All pressure and temperature input signals are from safety related instrumentation which is not used for any other control or indication functions. The recorder will be powered from a vital bus which has the capability of being energized from either offsite power or the emergency diesel generators.

The recorder will display hot leg temperature, the saturation temperature corresponding to measured pressurizer pressure and the margin to saturation, all in ^{o}F. The recorder will also provide an alarm signal for margin to saturation of less than $50^{o}F$.

Additionally, procedures have been implemented requiring operator use of a saturation temperature/pressure curve, including instructions for its use and significance, indicating the saturation pressure which would correspond to the hot leg temperature. Operator use of the saturation temperature/ pressure curve includes use of safety related instrumentation independent from that which will be used for the saturation recorder. This curve and its instructions will be utilized as a back-up to the primary coolant saturation recorder.

Operating procedures will be developed or revised, as appropriate, to include instructions as to use of the saturation recorder and its associated back-up curve. The procedures will specify that the saturation recorder is not to be used exclusive of other related station parameters. These procedures will be implemented coincident with placing the recorder in service.

Position 2:

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided. (Schedule: Submit new level instrument design by January 1, 1980, and complete installation by January 1, 1981.)

Response:

The Westinghouse Owner's Group is currently performing analytical work in the area of inadequate core cooling to determine if additional instrumentation or controls are necessary. The analytical work is scheduled to be completed by October 31, 1979. Any instrumentation or controls determined to be necessary will be appropriately incorporated into the San Onofre Unit 1 design.

The functional design of any such additional instrumentation or controls will be submitted by January 1, 1980, along with a description of the procedures to be used with the equipment, and the analysis used in developing these procedures. However, the installation of any instrumentation 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 (SEP). SEP Review Topics which are expected to have a direct bearing on implementation of the stated position requirement include: Topics III-4.C, III-5.A and III-6. For example, Topic III-5.A, Effects of Pipe Breaks on Structures, Systems and Components Inside Containment, will evaluate the effect of pipe breaks inside containment on the ability to safety shutdown and to mitigate the consequences of the pipe break. Based on the evaluation, pipe rerouting, additional supports and relocation may be required. Since installation of instrumentation is dependent on the layout inside containment and the potential requirement for pipe whip or jet impingement protection, this review topic must be completed prior to initiating engineering work. Similarly, completion of Topic III-4.C, Internally Generated Missiles; and III-6, Seismic Design Considerations will also likely result in new requirements and/or design criteria which affect the installation of instrumentation.

The assumptions and requirements to be utilized in the analytical work are the subject of continuing discussions between the Owner's Group and the NRC. Accordingly, the completion dates discussed above are also subject to change based on the outcome of the discussions between the Owner's Group and the NRC.

Section 2.1.4 - Containment Isolation Provisions for PWRs and BWRs

Position 1:

All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation. (Schedule: Complete implementation by January 1, 1980.)

Response:

The Containment Isolation Signal (CIS) at San Onofre Unit 1 will be modified to include diversity in the parameters sensed for initiation as recommended by SRP 6.2.4. CIS currently takes place on containment pressure above 2 psig. The Safety Injection Actuation Signal (SIAS) will be incorporated as the diverse initiation parameter.

The design criteria to be utilized for the modifications are included in Appendix 1. The implementation schedule for the modifications is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by February 15, 1980. In addition, completion of the construction efforts is expected to require approximately three months of which the last seven weeks require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence February 15, 1980; construction will be completed during the first outage of sufficient duration, or during the next refueling outage which is now scheduled for March-April, 1980.

Position 2:

All plants shall give careful reconsideration to the definition of essential and nonessential systems, shall identify each system determined to be essential, shall identify each system determied to be nonessential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the reevaluation to the NRC. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. A thorough review of the San Onofre Unit 1 containment isolation design was completed as part of the Sphere Enclosure Project and assessment of compliance with 10CFR50 Appendix J. The results of this evaluation were provided in Attachment 2 of the enclosure to our April 21, 1976 letter in Docket No. 50-206 to the NRC which identified (1) containment penetrations by number, (2) isolation valves provided for such penetrations (including valve designation and location, and (3) whether the isolation valves are subject to type C leakage rate testing. In addition, tabular information was provided comparing each line/pentration with 10 CFR 50, Appendix J requirements.

An additional review of the containment design was completed in response to I.E. Bulletin No. 79-06A and Revision 1 thereto. Our letter dated June 25, 1979 in Docket No. 50-206 to the NRC provided additional information to supplement our responses to I.E. Bulletin No. 79-06A and Revision 1 thereto. This information identified the containment penetrations which are not isolated by a CIS and provided the basis for not isolating these penetrations.

Position 3:

All nonessential systems shall be automatically isolated by the containment isolation signal. (Schedule: Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. All containment penetrations which are required to be isolated are isolated by a CIS. Where isolation is not warranted, the basis for nonisolation is as described in the referenced letters discussed in the response to Position 2, above.

Position 4:

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. (Schedule: Complete implementation by January 1, 1980.)

Response:

The design of control systems for automatic containment isolation valves will be modified to prevent automatic reopening of these valves upon reset of the CIS.

The design criteria to be utilized for the modifications are included in Appendix 1. The implementation schedule for the modifications is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by February 15, 1980. In addition, completion of the construction efforts is expected to require approximately three months of which the last seven weeks require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence February 15, 1980; construction will be completed during the first outage of sufficient duration, or during the next refueling outage which is now scheduled for March-April, 1980.

Section 2.1.5.a - Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems

Position:

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system. (Schedule: Provide a system description and implementation schedule by January 1, 1980 and complete installation by January 1, 1981.)

Response:

San Onofre Unit 1 uses a purge system for post-accident combustible gas control of the containment atmosphere. Based on the concerns surrounding the generation of hydrogen identified in I.E. Bulletin 79-06A and Revision 1 thereto, Westinghouse was requested to determine the amount of combustible gas which may be generated based on station specific design (e.g., use of stainless steel fuel cladding and hydrazine containment spray additive). This evaluation is scheduled to be completed by November 1, 1979.

Based on the results of this evaluation, a system will be provided to adequately control post-accident combustible gas which may be generated inside containment. Two alternative methods which are being evaluated include (1) upgrading the existing purge system, and 2) installing hydrogen recombiners inside or outside containment.

A description of the method to control post-accident combustible gas will be provided by January 1, 1980. However, implementation of the 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 (SEP). SEP Review Topics which are expected to have a direct bearing on implementation of the stated position requirement include: Topics III-2, III-4.A, III-4.C, III-5.A, III-5.B, III-6, VI-5, VI-2.D and VI-3. For example, Topic III-6, Seismic Design Considerations, will include the specification of a new seismic spectra for the purpose of seismic reevaluation. Since modifications will require seismic analyses which utilize the new response spectra, this review topic must be completed prior to initiating engineering work. Similarly, completion of Topic III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-4.C, Internally Generated Missiles; III-5.A, Effects of Pipe Breakdown Structures, Systems and Components Inside Containment; III-5.B, Pipe Break Outside Containment; VI-5, Combustible Gas Control; VI-2.D, Mass and Energy Release for Postulated Pipe Breaks Inside Containment; and VI-3, Containment Pressure and Heat Removal Capability will also likely result in new requirements and/or design criteria which affect the post-accident combustible gas control system.

Section 2.1.5.b - Inerting BWR Containments

San Onofre Unit 1 is a Westinghouse PWR; accordingly this item is not applicable.

Section 2.1.5.c - Capability to Install Hydrogen Recombiner at each Light Water Nuclear Power Plant

No action is required on this position requirement at this time as discussed in the NRC letter dated September 13, 1979 to all operating nuclear power plants.

Section 2.1.6.a – Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxilary Systems) for PWRs and BWRs

Position:

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

(Schedule: Complete implementation by January 1, 1980.)

Response:

The Recirculation System (RS); portions of the Containment Spray System (CSS); portions of the Chemical and Volume Control System (CVCS) including Letdown and Makeup; the Primary Coolant and Containment Atmosphere Sampling Systems; and the Gaseous Radioactive Waste Systems have been identified as systems which process primary coolant, and could contain high level radioactive materials. Programs for these systems will be implemented as discussed below. The Residual Heat Removal System is located entirely within the containment and is not included among the systems covered by these programs.

Procedures outlining the leak reduction measures will be implemented by January 1, 1980. In addition, procedures will be implemented by January 1, 1980 describing preventive maintenance to reduce leakage to as-low-as-practical levels, including periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

All practical leak reduction measures will be implemented for the above systems. Procedures describing the leak rate testing will be prepared and implemented. However, the leak rate testing of the above systems can only be performed during a shutdown. Therefore, the initial measurements of leakage rates for the above systems will be performed during the first outage of sufficient duration, or during the next refueling outage which is now scheduled for March-April, 1980. The actual leak rate measured will be reported to the NRC within 30 days following completion of the leak rate testing.

Section 2.1.6.b - Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations

Position:

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas activity contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation field during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility. (Schedule: Complete the design review by January 1, 1980. Implement plant modifications by January 1, 1981.)

Response:

A radiation and shielding design review of the spaces around systems that could contain highly radioactive materials is being performed assuming that the equivalent of 50% of the core radioiodine and 100% of the core noble gas is contained in the primary coolant. The design review also assumes radiation levels are limited to less than 15 mr/hr for areas requiring continuous occupancy, less than 100 mr/hr for areas requiring possible frequent access, and less than 10 CFR Part 20 for other areas. Based on the results of this review, installation of additional permanent- or temporary shielding, and/or post-accident procedure revisions and/or station modifications will be accomplished to provide adequate operational access to vital areas and protection of appropriate safety equipment.

The design review will be completed and any station modifications will be identified to the NRC by January 1, 1980. However implementation of any 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 (SEP). SEP Review Topics which are expected to have a direct bearing on implementation of the stated position requirement include: Topic III-2, III-4.A, III-4.C, III-5.B and III-6. For example, Topic III-2, Wind and Tornado Loadings, will evaluate the design basis tornado winds and pressure drop on structures or equipment in accordance with Regulatory Guides 1.76 and 1.117. It is likely that some tornado criteria will be established (e.g., Regulatory Guide 1.76) since such criteria was not a design basis at San Onofre Unit 1. Since the erection of shielding will require consideration of tornado loadings, this review topic must be completed prior to initiating engineering work. In addition, Topic III-6, Seismic Design Considerations, will include the specification of a new seismic response spectra for the purpose of seismic reevaluation. Since the erection of shielding will require seismic analyses which utilize the new response spectra, this review topic must be completed prior to initiating engineering work. Similarly, completion of Topic III-4.A, Tornado Missiles; III-4.C, Internally Generated Missiles; and III-5.B, Pipe Break Outside Containment will also likely result in new requirements and/or design criteria which affect the erection of shielding.

Section 2.1.7.a - Automatic Initiation of the Auxiliary Feedwater System for PWRs

Position:

Consistent with satisfying the requirement of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements. (Schedule: Complete implementation of control grade by January 1, 1980. Complete implementation of safety grade by January 1, 1981.)

Response

Conceptual engineering efforts have been initiated to provide automatic actuation of the auxiliary feedwater system consistent with the above position requirements. However, final engineering, procurement and construction efforts required to fully implement the above position requirements will be deferred until (1) the NRC Bulletins and Orders Group completes their review of the auxiliary feedwater system, and (2) the completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the Systematic Evaluation Program (SEP). SEP Review Topics which are expected to have a direct bearing on implementation of the stated position requirement include: Topics III-2, III-4.A, III-4.C, III-5.B, III-6, VII-3 and X.

For example, Topic III-6, Seismic Design Considerations, will include the specification of a new seismic response spectra for the purpose of seismic reevaluation. Since modifications to automate the auxiliary feedwater system will require seismic analyses which utilize the new response spectra, this review topic must be completed prior to initiating final engineering work. In addition, Topic II-5.B, Pipe Break Outside Containment, will reevaluate the previous pipe break outside containment analysis in accordance with current criteria. This will include the evaluation of other high energy lines and reconsideration of the acceptability of augmented inservice inspection. This latter item particularly affects the containment penetration area where the feedwater pipes and auxiliary feedwater pipes interconnect. Pipe relocation and/or pipe break restraints may be necessary. Since automation of the auxiliary feedwater system will involve modifications in the containment penetration area, this review topic must be completed prior to initiating final engineering work. Similarly, completion of Topic III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-4.C, Internally Generated Missiles; VII-3, Systems Required for Safe Shutdown; and Topic X, Auxiliary Feedwater System will also likely result in new requirements and/or design criteria which affect the auxiliary feedwater system.

Section 2.1.7.b - Auxiliary Feedwater Flow Indication to Steam Generators for PWRs

Position:

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

(Schedule: Complete implementation of control grade by January 1, 1980. Complete implementation of safety grade by January 1, 1981.)

Response:

The automation of the auxiliary feedwater system described in response to Section 2.1.7.a of NUREG 0578 of this report will include addition of control room flow indication for each steam generator. The auxiliary feedwater flow indication will be consistent with the above position requirements, except the stated single failure criterion. The single failure criterion will be met by having separate and independent flow indication to each steam generator since only one of three steam generators is required to remove decay heat from the reactor coolant system. In addition, each steam generator has water level indication displayed and alarmed in the control room.

Section 2.1.8.a - Improved Post-Accident Sampling Capability

Position:

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift. (Schedule: Complete the design review, prepare revised procedures and describe proposed modifications by January 1, 1980. Implement plant modifications by January 1, 1981.)

Response:

A design and operational review of the reactor coolant and containment atmosphere sampling systems will be performed to determine the capability of personnel to obtain samples in a timely manner during accident conditions without exposing any individual in excess of 3 and 18-3/4 Rems to the whole body or extremities, respectively. Any procedure revisions or station modifications determined to be necessary as a result of this review will be provided by January 1, 1980. However, the implementation of any necessary station modifications (i.e., permanent or temporary shielding) 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 basis for this deferral is as discussed in the response to Section 2.1.6.b of NUREG-0578 in this report.

With respect to radiological spectrum and chemical analyses, the Westinghouse Owner's Group is currently studying methods to perform these analyses. The study will include development of guidelines for sample preparation, evaluation and recommendations regarding application of automatic or in-line analyses, review of alternative manual analysis procedures (including specification of equipment and shielding requirements), and specification of minimum capability for gamma spectroscopy equipment. The tentative Westinghouse Owner's Group schedule for completing the study is March, 1980.

Following completion of the study, the results will be reviewed for application at San Onofre Unit 1 and methods will be developed to perform the radiological spectrum and chemical analyses. These methods may involve procedure revisions and/or station modifications. Any procedure revisions and/or station modifications determined necessary to perform the analyses will be provided within 90 days from receipt of the Westinghouse Owner's Group's study. However, implementation of any necessary modifications (i.e., permanent or temporary shielding) will be deferred as discussed above.

Section 2.1.8.b - Increased Range of Radiation Monitors

Position 1:

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

a. Noble gas effluent monitors with an upper range capacity of 10^5 microcuries/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 microcuries (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten. (Schedule: Complete procedures by January 1, 1980; complete installation by January 1, 1981.)

Response:

A noble gas effluent monitor will be provided such that capability to monitor the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 microcuries/cc (Xe-133). The design criteria to be utilized for the monitor are included in Appendix 1. The implementation schedule for the monitor is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by December 1, 1980. In addition, completion of the construction efforts is expected to require approximately four months of which the last two months require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence December 1, 1980; construction will be completed during the first outage of sufficient duration, or during the refueling outage which is now scheduled for September-October, 1981.

In addition, procedures will be revised or developed as necessary for estimating the noble gas release rates if the existing effluent instrumentation goes off scale. All release sources, such as the main steam safety valves, will be considered. The procedures will be implemented by January 1, 1980.

Position 2:

Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis. (Schedule: Complete procedures by January 1, 1980; complete onsite laboratory analysis capability by January 1, 1981.)

Response:

Stack effluent radioiodine samples are currently obtained by adsorption on charcoal cartridges. In addition, particulate cartridge samples are also obtained. These samples are currently analyzed onsite. The capability for onsite laboratory analysis of these cartridges during accident conditions is currently being evaluated, and the results of this evaluation, including a proposed method for meeting the intent of the stated position requirement, will be provided by January 1, 1980. However, the implementation of any necessary station modifications (i.e., permanent or temporary shielding) 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 basis for this deferral is as discussed in the response to Section 2.1.6.b of NUREG-0578 in this report.

In addition, procedures will be revised or developed as necessary by January 1, 1980 for estimating the radioiodine release rates if the existing effluent instrumentation goes off scale. All release sources, such as the main steam safety valves, will be considered.

Position 3:

In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment. (Schedule: Complete installation by January 1, 1981.)

Response:

Two radiation level monitors with a maximum range of 10^8 rad/hr will be installed in containment. The design criteria to be utilized for the monitors are included in Appendix 1. The implementation for the monitors is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by December 1, 1980. In addition, completion of the construction efforts is expected to require approximately four months of which the last two months require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence December 1, 1980; construction will be completed during the first outage of sufficient duration, or during the refueling outage which is now scheduled for September-October, 1981.

Section 2.1.8.c - Improved In-Plant Iodine Instrumentation

Position:

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. (Schedule: Complete implementation by January 1, 1980.)

Response:

Equipment for gamma energy spectrum analysis to determine the airborne iodine concentrations currently exists at San Onofre Unit 1. Procedures for utilizing this equipment to determine airborne iodine concentrations will be reviewed and revised, as required, and the associated training completed by January 1, 1980.

Section 2.1.9 - Analysis of Design and Off-Normal Transients and Accidents

Position:

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling, and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability.

Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will consititute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures. (Schedule: Analyses, procedural changes and operator training shall be provided following the schedule in Table B-2 of NUREG-0578.)

Response:

Southern California Edison Company is participating as a member of the Westinghouse Owner's Group in the review of areas described in the above stated position. The Owner's Group is performing generic analyses, and developing procedure guidelines to address these areas as discussed below:

1. The small break LOCA generic analyses and prepration of emergency procedure guidelines have been completed and a report (WCAP-9600) was submitted to the NRC on June 29, 1979. Additional information regarding WCAP-9600 was submitted to the NRC by Owner's Group's letters dated September 11, 1979 and September 28, 1979. Station specific small break LOCA analysis is being performed for San Onofre Unite 1 to conform the applicability of the generic results and is scheduled for completion by October 31, 1979. Station procedure revisions and operator training based on the applicable generic procedure guidelines are scheduled for completion by January 1, 1980.

2. Inadequate core cooling analyses are being performed and procedure guidelines developed by the Owner's Group. The results are scheduled to be submitted by October 31, 1979. Following completion of the generic analyses, an evaluation will be performed to determine the need for any station specific analysis. Station procedure revisions will be completed by January 1, 1980 based on the applicable generic guidelines.

3. Analysis of transients and accidents will be performed and procedure guidelines developed by the Owner's Group. A short term generic program has been developed with results scheduled to be submitted by January 1, 1980. Short term station specific program results will be completed by April 1, 1980. Revised station procedures and operator training will be completed by July 1, 1980.

4. LOFT pretest calculations are being performed by the Owner's Group and will be submitted by November 15, 1979.

The completion dates for the inadequate core cooling analysis and transients and accidents analysis discussed in Items 2 and 3 above are subject to continuing discussions between the Owner's Group and the NRC in order to develop specific requirements. Accordingly, the completion dates for the related station specific review, procedure guidelines and operator training discussed in Items 2 and 3 above are also subject to change based on the outcome of the discussions between the Owner's Group and the NRC.

Section 2.2.1.a - Shift Supervisor's Responsibilities

Position 1:

The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties. (Schedule: Complete implementation by January 1, 1980.)

Response:

The Vice President of Power Supply will issue a management directive by January 1, 1980 and annually thereafter to meet the above stated position requirement.

Position 2:

Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

(Schedule: Complete implementation by January 1, 1980.)

Response:

Administrative procedures will be reviewed and revised, as required, to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to meet the stated position requirements. These procedures will be implemented by January 1, 1980.

Position 3:

Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety. (Schedule: Complete implementation by January 1, 1980.)

Response:

Training programs will be developed for the shift supervisors to meet the stated position requirement by January 1, 1980.

Position 4:

The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room. (Schedule: Complete implementation by January 1, 1980.)

Response:

The Vice President of Power Supply will review the administrative duties of the shift supervisor by January 1, 1980. Any administrative functions that are determined to detract from or are subordinate to management responsibilities for assuring safe plant operation will be delegated to other operations personnel not on duty in the control room.

Section 2.2.1.b - Shift Technical Advisor

Position:

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Based on a reassessment of the stated position requirement by the NRC, entitled, "Alternatives to Shift Technical Advisors" (Enclosure 2, to the NRC letter dated September 13, 1979 to all operating plants) was also provided as additional guidance to meeting the intent of the stated position requirement. (Schedule: Shift technical advisor on duty by January 1, 1980 and completely trained by January 1, 1981.)

Response:

An on-shift technical advisor will be provided to meet the intent of the stated position requirement as clarified by the "Alternatives to Shift Technical Advisors" contained in the NRC letter dated September 13, 1979. The advisor will be placed on shift by January 1, 1980 and completely trained by January 1, 1981.

Section 2.2.1.c - Shift and Relief Turnover Procedures

Position:

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

    a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

    c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments). (Schedule: Complete implementation by January 1, 1980.)

Response:

Procedures governing shift and relief turnover will be implemented to meet the stated position requirements by January 1, 1980.

In addition, a system will be established to evaluate the effectiveness of the turnover procedures by January 1, 1980.

Section 2.2.2.a - Control Room Access

Position:

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

(Schedule: Complete implementation by January 1, 1980.)

Response:

Administrative procedures will be developed and implemented to meet the provisions of the stated position requirements by January 1, 1980.

Section 2.2.2.b - Onsite Technical Support Center

Position:

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay. (Schedule: Establish center by January 1, 1980 and upgrade to meet all requirements by January 1, 1981.)

Response:

An onsite technical support center (OTSC) is currently established in the Visitor's Viewing Area adjacent to the Control Room and is large enough to hold at least 25 individuals. The OTSC has communication links with the control room, operational support center (as described in response to Section 2.2.2.c of NUREG-0578 of this report), emergency operations center (as described in response to Requirement 3 of the Near Term Requirements for Improving Emergency Preparedness of this report) and the NRC (as described in the May 3, 1979 letter in Docket No. 50-206 submitted in response to I.E. Bulletin No. 79-06A and Revision 1 thereto). The OTSC is habitable to the same degree as the Control Room for postulated accident conditions.

Station records, including but not limited to, systems descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, and electrical drawings are currently stored onsite (in the Engineering Drawing Management (EDM) Center) in another building. These documents and records are readily accessible to the OTSC under emergency conditions.

Currently, the OTSC does not have the capability to display vital station technical data; however, a message control box (pass through type) is available which would allow the OTSC to gain access (without entering the

Control Room) to strip chart recordings of vital station technical data which might be utilized by engineering and management personnel in support of reactor operations in the event of an accident. To improve the capability to support reactor operations in the event of an accident, the OTSC will be upgraded by addition of a Technical Data Display and Transmit System. Alternatives being evaluated include (1) use of high resolution, remote scan and zoom, color, closed circuit television cameras with display consoles having video-tape playback provisions, (2) hard wired instrumentation and recorders monitoring vital station technical data, or (3) a combination of 1 and 2.

The design criteria to be utilized for the Technical Data Display and Transmit System are included in Appendix 1. The implementation schedule for the Technical Data Display and Transmit System is included in Appendix 2. Based on the implementation schedule, engineering and procurment efforts will be completed by December 19, 1980. In addition, completion of the construction efforts is expected to require approximately four months of which the last two months require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence December 19, 1980; construction will be completed during the first outage of sufficient duration, or during the refueling outage which is now scheduled for September-October, 1981. In addition, the San Onofre Emergency Plan will be revised to describe the existence and function of the OTSC by mid-1980 consistent with the near term requirements for improving emergency preparedness contained in the September 13, 1979 NRC letter to all operating power plants.

Section 2.2.2.c - Onsite Operational Support Center

Position:

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management. (Complete implementation by January 1, 1980.)

Response:

The current station design complies with the stated position requirement as discussed below. An onsite operational support center is currently available on the first floor of the Administration and Control Building. Communication with the control room via in-plant telephones is currently available. In addition, the San Onofre Emergency Plan will be revised to reflect existence and function of the center by mid-1980 consistent with the near term requirements for improving emergency preparedness contained in the September 13, 1979 NRC letter to all operating power plants.

Section 2.2.3 - Revised Limiting Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability

No action is required on this position requirement at this time as discussed in the NRC letter dated September 13, 1979 to all operating nuclear power plants.

3. RESPONSE TO ADDITIONAL FOLLOWUP RECOMMENDATIONS IDENTIFIED IN THE SEPTEMBER 13, 1979 LETTER FROM D. G. EISENHUT TO ALL OPERATING NUCLEAR POWER PLANTS

Section 3.1 - Instrumentation to Monitor Containment Conditions During the Course of an Accident

3.1.1 - Containment Pressure Indication

Position

A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments. The design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability, shall be met. (Schedule: Complete implementation by January 1, 1981.)

Response

Containment pressure indication instrumentation will be installed to meet the stated position requirements.

The design criteria to be utilized for the instrumentation are included in Appendix 1. The implementation schedule for the instrumentation is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by December 19, 1980. In addition, completion of the construction efforts is expected to require approximately four months of which the last two months require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence December 19, 1980; construction will be completed during the first outage of sufficient duration, or during the refueling outage which is now scheduled for September-October, 1981.

3.1.2 - Containment Hydrogen Monitor

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure. The design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability shall be met. (Schedule: Complete implementation by January 1, 1981.)

Response

Containment hydrogen concentration indication instrumentation will be installed to meet the stated position requirements.

The design criteria to be utilized for the instrumentation are included in Appendix 1. The implementation schedule for the instrumentation is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by December 19, 1980. In addition, completion of the construction efforts is expected to require approximately four months of which the last two months require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence December 19, 1980; construction will be completed during the first outage of sufficient duration, or during the refueling outage which is now scheduled for September-October, 1981.

3.1.3 - Containment Water Level Indication

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. Also for PWRs, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool. The wide range instrumentation shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability. The narrow range instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested. (Schedule: Complete implementation by January 1, 1981.)

Response

Containment wide range and narrow range water level instrumentation will be installed to meet the stated position requirements.

The design criteria to be utilized for the instrumentation are included in Appendix 1. The implementation schedule for the instrumentation is included in Appendix 2. Based on the implementation schedule, engineering and procurement efforts will be completed by December 19, 1980. In addition, completion of the construction efforts is expected to require approximately four months of which the last two months require a station outage. Accordingly, construction which does not require a station outage is scheduled to commence December 19, 1980; construction will be completed during the first outage of sufficient duration, or during the refueling outage which is now scheduled for September-October, 1981.

Section 3.2 - Reactor Coolant System High Point Vents

Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentrations limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

(Schedule: Submit design description by January 1,1980; complete installation by January 1, 1981.)

Response

A design for reactor coolant system high point venting to meet the stated position requirements is currently being evaluated. A description of the design will be provided by January 1, 1980. However, implementation of the design and the associated procedures for use will be deferred pending completion of the integrated assessment of potential modifications identified by review of design and operation in connection with the Systematic Evaluation Program (SEP). SEP Review Topics which are expected to have a direct bearing on implementation of the stated position requirement include: Topics III-4.C, III-5.A and III-6. For example, Topic III-4.C, Internally Generated Missiles, will evaluate the effects of postulated internally generated missiles on equipment and structures. Consideration of such missiles were not evaluated as part of the design basis for San Onofre Unit 1. Since the design of the venting systems will require consideration of internally generated missiles, this review topic must be completed prior to initiating engineering work. In addition, Topic III-6, Seismic Design Considerations, will include the specification of a new seismic spectra for the purpose of seismic reevaluation. Since the design of a venting system will require seismic analyses which utilize the new response spectra, this review topic must be completed prior to initiating engineering work. Similarly, completion of Topic III-5.A, Effects of Pipe Breaks on Structures, Systems and Components Inside Containment will also result in new requirements and/or design criteria which affect the design of a venting system.

Section 3.3 - Emergency Preparedness Improvements

3.3.1 - Emergency Plan Conformance to Regulatory Guide 1.101

Position:

Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters. (Schedule: Complete Implementation by mid-1980.)

Response:

The San Onofre Emergency Plan complies with the requirements contained in Regulatory Guide 1.101. The plan will be revised by mid-1980 to make use of the upgraded uniform action level criteria based on plant parameters. These action level criteria are currently under development and will meet the intent of the requirements issued by the NRC.

3.3.2 - Incorporation of Plant Instrumentation into Emergency Plan Action Level Criteria

Position:

Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

Response:

Information provided by plant instrumentation which is used to follow the course of an accident will be factored into the San Onofre Emergency Plan action levels by mid-1980. Instrumentation for post-accident sampling, high range radioactivity monitoring, inplant radiodine monitoring, and detection of inadequate core cooling will be considered for inclusion in the action level criteria.

3.3.3 - Emergency Operations Center Requirements

Position:

Determine that an emergency operations center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendations for an in-plant technical support center is underway. (Schedule: (a) Designate location and alternate location and provide communications to plant by mid-1980, (b) Upgrade Emergency Operations Center in conjunction with in-plant technical support center by January 1, 1981.)

Response:

(a) An emergency operations center for Federal, State and local personnel has been established at the San Clemente City Hall. The emergency operations center has been provided with a two-way automatic ring down circuit to the San Onofre Unit 1 technical support center. SCE, in concert with local authorities, is also conducting a search for an appropriate alternate emergency operations center. This effort is scheduled for completion by mid-1980.

(b) The emergency operations center will be further upgraded if additional requirements become known. For the present, the established emergency operations center meets all defined in-facility requirements.

3.3.4 - Improved Offsite Monitoring Capabilities

Position:

Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites. (Schedule: Complete implementation prior to mid-1980.)

Response:

Additional TLD monitoring locations will be established by mid-1980 to meet this requirement.

3.3.5 - Compatibility of Federal, State, Local, and Utility Emergency Plans

Position:

Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement. (Schedule: (a) Against current criteria by mid-1980, (b) Against upgraded criteria by January 1, 1981.)

Response:

(a) The NRC has assessed the State of California/Local Emergency Response Plans and concurred on August 15, 1978. In addition, the NRC has recently assessed the relationship of these plans to the San Onofre Emergency Plan and Federal plans. Southern California Edison is now awaiting the results of that review and will cooperate with Federal, State and local agencies to provide the capability to take appropriate emergency actions. Any revisions determined to be required for the San Onofre Emergency Plan based on this review will be completed by mid-1980.

(b) SCE has contacted State, County and local authorities and requested that a joint revision of all emergency response plans be conducted to meet upgraded criteria by January 1, 1981. This will establish new emergency planning zones in accordance with forthcoming State and Federal requirements. Southern California Edison will cooperate with the Office of State Programs and the Office of Inspection and Enforcement to ensure compliance with this requirement.

3.3.6 - Test Exercises of Approved Emergency Plans

Position:

Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrent reviews of the Office of State Programs. As a preliminary planning bases, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking, (Schedule: (a) Conduct Test of licensees emergency plans by mid-1980, (b) Conduct Test of State emergency plans by mid-1980, (c) Conduct Joint Test exercise of emergency plans (Federal, State, local, licensee) for all operating plans within 5 years.)

Response:

(a) A test of the San Onofre Emergency Plan will be scheduled for completion by mid-1980.

(b) Southern California Edison will request that a test of the California Emergency Response Plan be conducted concurrently with the San Onofre Emergency Plan test.

(c) Southern California Edison will cooperate with the NRC and Federal Agencies to schedule and conduct a joint test exercise of emergency plans within 5 years.

APPENDIX 1: SUMMARY OF DESIGN CRITERIA

APPENDIX I

DESIGN CRITERIA FOR SHORT-TERM MODIFICATIONS

The following design criteria is applicable to the modifications described in the attached table as identified against the respective modifications.

1. Category A Structures, Systems and Components

   Seismic Category A structures, systems, and components shall be designed for no loss of function when subjected to the design basis earthquake (DBE). These structures, systems, and components shall also be designed to remain within the allowable stress limits when subjected to the operating basis earthquake (OBE). The maximum free-field ground-motion acceleration for the DBE and OBE shall be at least 0.5g and 0.25g, respectively, based on Section 9.2 of the San Onofre Unit 1 FSAR.

   Analysis of the dynamic loads of seismic category A piping is accomplished using response spectrum or time-history approaches, which utilizes the natural period, mode shapes and appropriate damping factors of the particular system. The adequacy of design shall be such that there is no loss of function during and after the prescribed seismic disturbance, i.e., OBE, and DBE. Damping shall be taken at 2%.

2. Single Failure Design

   Items requiring single failure design shall be capable of withstanding a single active failure without loss of function.

3. Electrical Classification to IE

   This classification is applied to electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment. Class IE systems and equipment criteria is described in IEEE Standard 308-1974.

4. Containment Building

   The highest containment flood to result from/an accident is the 10' elevation. All safety related systems and equipment, to be located inside the containment below this elevation, shall be designed to operate in a submerged condition.

5. Requirement to Operate from Emergency Power

   All electrically operated equipment to be installed shall be operable from an emergency power source.

6. Environmental Conditions

All equipment to be installed shall be designed to withstand the environmental conditions of normal operation.

Normal Operation

A. In Containment at Reactor Coolant Loop

1.	Pressure	14.7 psia
2.	Temperature	70-120^{o}F
3.	Relative Humidity	50-100%
4.	Integrated Dose	6 X 10^{6} rads

B. In Containment Separated from Reactor Coolant Loop

1.	Pressure	14.7 psia
2.	Temperature	70-120^{o}F
3.	Relative Humidity	50-100%
4.	Integrated Dose	4 X 10^{3} rads

C. In Reactor Auxiliary Building and Spent Fuel Handling Building

1.	Pressure	14.7 psia
2.	Temperature	36-104^{o}F (36-140^{o}F in Turbine Lube Oil Area)
3.	Relative Humidity	0-90%
4.	Integrated Dose	3 X 10^{3} rads

D. All Other Areas

1.	Pressure	14.7 psia
2.	Temperature	36-104^{o}F
3.	Relative Humidity	0-90%
4.	Integrated Dose	10^{3} rads

7. Post LOCA Operation

Equipment inside and outside the containment, which is required to be operable during and subsequent to a LOCA, shall be capable of operation in the following conditions.

A. Inside Containment

1. Pressure	50 psig
2. Temperature	70–291^{o}F
3. Relative Humidity	50–100%
4. Integrated Dose	1 X 10^{9} rads

B. Outside Containment

1. Pressure	14.7 psig
2. Temperature	36–104^{o}F
3. Relative Humidity	0–90%
4. Integrated Dose	3 X 10^{7} rads

SYSTEM/COMPONENT	QUALITY CLASS	1. SEISMIC CATEGORY	2. SINGLE FAILURE DESIGN	3. ELECTRICAL CLASSIFICATION	4. CONTAINMENT FLOODING	5. OPERATION FROM EMERGENCY POWER	6. ENVIRONMENTAL CONDITIONS	7. POST LOCA OPERATING CONDITIONS
2.1.1 <u>Capability to Operate PORV Block Valves During Loss of Offsite Power</u>								
Nitrogen backup supply	SR	A					X	X
Support hardware	SR	A					X	X
Associated electrical & controls	SR	A		IE	X	X	X	X
2.1.3.a <u>Position Indication of PORV, Block & Safety Valves</u>								
Flow indication sensors	SR	A					X	X
Piping & fittings	SR	A					X	X
Associated electrical & controls	SR	A		IE	X	X	X	X
2.1.3.b Primary Coolant Saturation Meter	NSR	A(C_2)*				X	X	
2.1.4 Additional CIAS on SI Actuation								
Modifications to electrical	SR	A	X	IE	X	X	X	X

* Items classified as seismic A (C_2) are to be designed to withstand a DBE without causing a failure which could impact a SR component, system or structure (reference NRC Reg. Guide 1.29, paragraph C_2).

		QUALITY CLASS	1. SEISMIC CATEGORY	2. SINGLE FAILURE DESIGN	3. ELECTRICAL CLASSIFICATION	4. CONTAINMENT FLOODING	5. OPERATION FROM EMERGENCY POWER	6. ENVIRONMENTAL CONDITIONS	7. POST LOCA OPERATING CONDITIONS
2.1.8.b	Stack Noble Gas Monitor	SR	A		IE		X	X	X
2.1.8.b	Radiation Monitors Inside Containment	SR	A	X	IE	X	X	X	X
2.2.2.b	Onsite Technical Support Center Data Display and Transmit Capability	NSR	$A(C_2)$*				X	X	
3.1.1, 3.1.2 & 3.1.3	Containment Monitors for Pressure, Hydrogen Concentration, and Water Level	SR	A	X	IE	X	X	X	X

* Items classified as seismic A (C_2) are to be designed to withstand a DBE without causing a failure which could impact a SR component, system or structure (reference NRC Reg. Guide 1.29, paragraph C_2).

APPENDIX 2: SUMMARY OF PLANT MODIFICATIONS SCHEDULE

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SONGS-1 TMI SUMMARY SCHEDULE

1979 AUG. SEP. OCT. NOV. 1979 DEC. 1980 JAN. FEB. MAR. APR. MAY JUN. JUL. AUG. SEP. OCT. NOV. 1980 DEC. 1981 JAN.

REFUELING OUTAGE 3-19 TO 4-30

NUREG-0578 ITEM		Start	Finish
2.1.1	PORV BLOCK VALVE EMERGENCY POWER SUPPLY REQUIREMENT		
	ENGINEERING	9-10	12-28
	PROCUREMENT	10-19	12-28
	CONSTRUCTION		*
2.1.3.A	DIRECT INDICATION OF VALVE POSITION		
	ENGINEERING	8-17	12-28
	PROCUREMENT	10-19	12-14
	CONSTRUCTION		*
2.1.4	DIVERSE CONTAINMENT ISOLATION		
	ENGINEERING	8-31	12-28
	PROCUREMENT	11-30	2-15
	CONSTRUCTION		*

NOTES:

*COMPLETION OF CONSTRUCTION REQUIRES 2 WEEK PLANT OUTAGE, (EXCEPT 2.1.4 WHICH REQUIRES 7 WEEK PLANT OUTAGE), TO OCCUR AT FIRST SHUTDOWN OF SUFFICIENT DURATION.

**COMPLETION OF CONSTRUCTION REQUIRES 2 MONTH PLANT OUTAGE, TO OCCUR AT FIRST SHUTDOWN OF SUFFICIENT DURATION.

SONGS-1 TMI SUMMARY SCHEDULE

1979 AUG. SEP. OCT. NOV. 1979 DEC. 1980 JAN. FEB. MAR. APR. MAY JUN. JUL. AUG. SEP. OCT. NOV. 1980 DEC. 1981 JAN.

REFUELING OUTAGE 3-19 TO 4-30

NUREG-0578 ITEM		Start	Finish
2.1.8.B	HIGH RANGE RADIATION MONITORS		
	ENGINEERING	9-7	11-1
	PROCUREMENT	1-11	12-1
	CONSTRUCTION		**
2.2.2.B	ONSITE TECHNICAL SUPPORT CENTER		
	ESTABLISH	10-17 △	
	ENGINEERING	5-1	10-1
	PROCUREMENT	6-1	12-1
	CONSTRUCTION		**
	HABITABILITY FOR ONSITE TECHNICAL CENTER		
	ENGINEERING	5-1	10-1
	PROCUREMENT	6-1	12-1
	CONSTRUCTION		**

SONGS-1 TMI SUMMARY SCHEDULE

1979 AUG. SEP. OCT. NOV. DEC. 1979 1980 JAN. FEB. MAR. APR. MAY JUN. JUL. AUG. SEP. OCT. NOV. 1980 1981 DEC. JAN.

REFUELING OUTAGE 3-19 TO 4-30

NUREG-0578 ITEM		Start	Finish
2.1.8.B	HIGH RANGE RADIATION MONITORS		
	ENGINEERING	9-7	11-1
	PROCUREMENT	1-11	12-1
	CONSTRUCTION		
2.2.2.B	ONSITE TECHNICAL SUPPORT CENTER		**
	ESTABLISH	10-17	
	ENGINEERING	5-1	10-1
	PROCUREMENT	6-1	12-1
	CONSTRUCTION		**
	HABITABILITY FOR ONSITE TECHNICAL CENTER		
	ENGINEERING	5-1	10-1
	PROCUREMENT	6-1	12-1
	CONSTRUCTION		**