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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 1, 1980

Docket No. 50-312

Mr. J. J. Mattimoe
Assistant General Manager and
Chief Engineer
Sacramento Municipal Utility District
6201 S Street
Sacramento, California 95813

Dear Mr. Mattimoe:

Enclosed for your information is the staff's evaluation for the Rancho Seco Nuclear Generating Station of the actions you have taken to satisfy the Category "A" items of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." This evaluation is based on your submitted documentation and the discussions between our staff at a site visit on March 17 and 18, 1980.

Based on our evaluation, we conclude that with the exception of Items 2.1.3a and 2.1.7a, you have satisfactorily met all Category "A" requirements. In our letter of February 1, 1980, we concluded that equipment availability would preclude implementation of Item 2.1.3a by January 1980, but stated that it should be implemented no later than June 1, 1980. Item 2.1.7a has been the subject of a separate review and was discussed in our letter of February 26, 1980. Our Office of Inspection and Enforcement will verify the adequacy of implemented procedures. These items are discussed in our evaluation.

It should also be noted that Item 2.1.9, which was classified neither Category "A" or "B", has also been the subject of separate reviews. The tasks required to be completed under this item are listed in Table B-2 of NUREG-0578. In this regard, you have made submittals relating to Tasks 1 through 4 and 7 of Table B-2. Our evaluation of your completion of Tasks 1 and 2 was documented in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants." Your submittals related to Tasks 3, 4, and 7 are still under staff review. Your preliminary proposals with respect to Tasks 5 and 6, which have been the subjects of meetings between Babcock & Wilcox operating reactor plant licensees and the NRC, are also still under staff review. Evaluations of the incomplete Tasks will be the subject of additional staff reports.

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These evaluations do not address the Technical Specifications necessary to ensure the limiting conditions for operation and the long-term operability surveillance requirements for the systems modified during the Category "A" review. You should be considering the proposal of such Technical Specifications. We will be in communication with you on this item in the near future.

Should you have any questions regarding our evaluation, please contact us.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure: Evaluation
for Rancho Seco

cc w/enclosure:
See next page

Sacramento Municipal Utility
District

cc w/enclosure(s):

David S. Kaplan, Secretary and
General Counsel
6201 S Street
P. O. Box 15830
Sacramento, California 95813

Sacramento County
Board of Supervisors
827 7th Street, Room 424
Sacramento, California 95814

Business and Municipal Department
Sacramento City-County Library
828 I Street
Sacramento, California 95814

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region IX Office
ATTN: EIS COORDINATOR
215 Fremont Street
San Francisco, California 94111

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

Thomas Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, D. C. 20036

Herbert H. Brown, Esq.
Lawrence Coe Lanpher, Esq.
Hill, Christopher and Phillips, P. C.
1900 M St., NW
Washington, D. C. 20036

Helen Hubbard
P. O. Box 63
Sunol, California 94586

Christopher Ellison, Esq.
Dian Grueuich, Esq.
California Energy Commission
1111 Howe Avenue
Sacramento, California 95825

Ms. Eleanor Schwartz
California State Office
600 Pennsylvania Avenue, S.E., Rm. 201
Washington, D.C. 20003

Docketing and Service Section
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Resident Inspector
P. O. Box 48
Fair Oaks, California 95628

Dr. Richard F. Cole
Atomic Safety and Licensing Board
Panel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Frederick J. Shon
Atomic Safety and Licensing Board
Panel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Elizabeth S. Bowers, Esq.
Chairman, Atomic Safety and
Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Michael R. Eaton
Energy Issues Coordinator
Sierra Club Legislative Office
1107 9th St., Room 1020
Sacramento, CA 95814

Sacramento Municipal Utility
District

cc w/enclosure(s)

Atomic Safety and Licensing Board
Panel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal
Board Panel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

California Department of Health
ATTN: Chief, Environmental
Radiation Control Unit
Radiological Health Section
714 P Street, Room 498
Sacramento, California

EVALUATION OF LICENSEE'S COMPLIANCE WITH
CATEGORY "A" ITEMS OF NRC RECOMMENDATIONS
RESULTING FROM TMI-2 LESSONS LEARNED

SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

Date: May 1, 1980

I. INTRODUCTION

By letter to Sacramento Municipal Utility District (licensee) dated September 13, 1979[1], the NRC transmitted the short term requirements related to the lessons learned from the TMI-2 accident that must be met for the Rancho Seco Nuclear Generating Station. This letter clarified, augmented, corrected and invoked the staff positions presented in the NRC Report NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations." In a later letter to the licensee dated October 30, 1979 [2], the NRC provided additional guidance and clarification concerning the staff positions and requirements that were transmitted by the September 13, 1979 letter. The short term requirements were divided into two categories, A and B. In accordance with the schedule given in the October 30, 1979 letter, Category A requirements were to be implemented by January, 1980 and Category B requirements were to be implemented by January 1981.

By letters dated October 18[3], November 19[4] and 26[5], 1979 and January 7[6] and 16[7], March 5 [8], 1980 and April 11 [9] and 17 [10], 1980 the licensee submitted commitments and documentation of actions taken at the Rancho Seco Nuclear Generating Station to implement our requirements. To expedite our review of the licensee's actions, members of the staff visited the licensee's facility on March 17 and 18, 1980. This report is an evaluation of the licensee's efforts to implement each Category A item which was to have been completed by January 1980 but no later than the startup from the January 1980 refueling outage.

The implementation of our requirements will be completed prior to the startup from the January 1980 refueling outage. One item, New Instrumentation for Inadequate Core Cooling, has been addressed for the Category A requirement but the staff has stated that we require the licensee to continue a design review and development program to arrive at an acceptable design for new instrumentation. This is considered a Category B item.

II. EVALUATION

Each of the Category A requirements applicable to PWR's is identified below. The numbered designation of each item is consistent with the identification used in NUREG-0578.

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS (Pressurizer Heaters)

The Sacramento Municipal Utility District has determined based on B&W calculations and startup testing experience that a minimum of 126 kilowatts of pressurizer heaters, which according to the licensee corresponds to a single bank of pressurizer heaters, should be available from an assured power source within two hours after loss of offsite power to establish and maintain natural circulation at hot standby conditions. Startup testing demonstrated that the total heat loss was 106 kilowatts. We have reviewed this information and note this calculated and measured heat loss is similar to heat loss estimates that have been accepted for other pressurized water reactors. We conclude that sufficient heater capacity has been provided to maintain pressure control in the pressurizer during normal hot standby conditions.

The licensee has modified his design to provide manual loading of 126 kilowatts of pressurizer heater capacity to each emergency diesel generator. Each redundant group of heaters are supplied from non-safety related motor control centers during all modes of operation. These non-safety related motor control centers are powered from emergency buses. Upon loss of offsite power or safety actuation signal, these heater breakers will be tripped and will be manually loaded on the safety buses. All actions to energize these heaters from the diesel generators can be accomplished from the control room. The licensee has stated that procedures have been modified covering the manual loading of these pressurizer heaters on the safety buses.

Based on our review of the above design, we conclude that the licensee has met the emergency power supply requirements for this item. Verification of the adequacy of the licensee's procedures and pressurizer heater power supply modification will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

EMERGENCY POWER SUPPLY REQUIREMENTS (Pressurizer Level and Power Operated Relief Valve (PORV) and Block Valves)

The current design has its PORV powered from a 125 volt DC bus supplied by a battery which is not safety grade. However, the charger for the battery is capable of being powered by either a diesel generator or by offsite power. The PORV block valve is an AC motor operator valve and its power supply has been modified to be powered from an essential motor control center. The power supplies for the PORV and its associated block valve are therefore independent and diverse.

The current design has four channels of safety grade pressurizer level instrumentation. These channels are also powered from safety buses. Thus the power supplies for PORV, PORV block valve and pressurizer level instrumentation are capable of being powered from both offsite and onsite emergency sources.

We conclude that the licensee has met the emergency power supply requirements for PORV, PORV block valve and pressurizer level instrumentation for this item. Our Office of Inspection and Enforcement will verify the adequacy of the power supply modification to the PORV block valve and will be documented in an appropriate inspection report.

2.1.2 PERFORMANCE TESTING FOR RELIEF AND SAFETY VALVES

The licensee has stated in its response to this item that it will participate in the Electric Power Research Institute (EPRI) program to conduct performance testing of PWR relief and safety valves. A description of the test program was provided by EPRI in December 1979. At present this program is under review to ensure that the NUREG-0578 requirements are met.

We will review the test program to confirm the applicability to the Rancho Seco plant. Completion of the test program is on a schedule different from Category "A" items. Therefore, we conclude that the licensee has satisfied the Category "A" requirements of this item.

2.1.3.a DIRECT INDICATION OF POWER OPERATED RELIEF VALVE AND SAFETY VALVE POSITION

The licensee has installed an acoustical monitoring system to monitor the position of PORV and safety valves. The acoustical monitoring system is provided by Technology for Energy Corporation (TEC) and is similar to those found acceptable by the staff for this purpose for other pressurized water reactors. Each valve will be monitored by two accelerometers. A charge amplifier amplifies each accelerometer output. The signal is then sent to a signal processing unit mounted in the cabinet located in the computer room just outside of the control room. Valve position indication and annunciation for each monitored valve has been provided in the control room. All portions of the system within the reactor building have been installed. The signal conditioner is not available at this time, however, the whole system will be operable by June 1, 1980. The licensee has stated that the valve position indication components will be seismically and environmentally qualified as appropriate for conditions applicable to their location by August 1, 1980.

Backup valve position indication is provided from temperature elements located in the PORV and safety valve tailpipes. Tailpipe temperature information is obtainable from the plant computer in the control room. A high temperature alarm is not available at present, however, the licensee has committed to add this alarm in the control room. An additional backup to the valve indications are pressurizer relief tank level and pressure which are indicated and alarmed in the control room.

Based on our review of the licensee's design, we conclude that the licensee will meet the requirements for direct indication of PORV and safety valve position for this item upon installation of the signal conditioner in May 1980. Our Office of Inspection and Enforcement will verify (1) the adequacy of installation of the above design, (2) the adequacy of the qualification and documentation of the valve position indication components and, (3) that the procedures for backup valve position indication are included in the plant emergency procedures. This will be documented in an appropriate inspection report.

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING FOR PWR'S AND BWR'S

The licensee has provided a description of the existing instrumentation for detection of inadequate core cooling. Further, plant emergency procedures and associated operator training with regard to existing instrumentation to be used by the operator to detect inadequate core cooling conditions, have been updated based on guidelines provided by B&W. Our generic review of this item is not yet complete and our evaluation thereof for this item will be reported separately.

ADDITIONAL INSTRUMENTATION

The licensee has stated that it has reviewed several conceptual designs for reactor vessel water level indication. By letter of March 5, 1980[8] the licensee informed the staff that it does not consider any of these designs that it has considered to date to be acceptable. We require that the licensee continue its efforts to provide an appropriate design. We conclude that the licensee has satisfied our short term requirement. However, the need to supplement existing instrumentation and to provide unambiguous indications of inadequate core cooling are still under review. We will complete this item during the review of Category "B" items.

SUBCOOLING METER

The licensee has installed two primary coolant saturation meters. These saturation meters will continuously display the margin between actual primary coolant temperature and the saturation temperature (the temperature at which boiling occurs). Each saturation meter will have one pressure input with a range of 0-2500 psig and two temperature inputs with a range of 120-650^{o}F from each hot leg. The temperature inputs to the saturation meters are not safety grade, however, they should be upgraded to safety grade requirements by January 1981 to include increasing the input temperature range to greater than the saturation temperature for the maximum system pressure (2500 psig). The power to each saturation meter will be provided from vital sources.

In addition to the saturation meters, backup capability already exists to detect inadequate core cooling conditions. This includes primary coolant temperature and pressure which are directly available to the operator by means of existing console indicators. Steam tables are also provided in the control room for use by the operator to determine saturation margin manually.

Based on our review of the licensee's design, we conclude that the design of the subcooling meters meets our requirements for this item. Verification of the adequacy of the installation of this design and inclusion of the use of subcooling meter in the operating procedures will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.4 CONTAINMENT ISOLATION

The NRC lessons learned requirements concerning containment isolation direct the licensee to: a) determine whether systems penetrating containment are considered essential or non-essential to safety; b) modify containment isolation circuitry to automatically isolate all non-essential systems by diverse parameters; and c) modify containment isolation circuitry to assure that clearing of the containment isolation signals does not cause the inadvertent opening of containment isolation valves. In addition, the isolation system was reviewed to assure that certain systems which are isolated but might be desirable to use following an accident or transient, can be reopened; and to assure that operator controls of containment isolation are not ganged to reopen multiple systems with a single operator action.

The licensee has identified the essential systems as a) those systems required immediately after a Safety Features Actuation Signal (SFAS) and b) those systems whose continued operation will not cause accident recovery problems and whose continued operation may aid in accident recovery. Non-essential systems are those not required immediately after an SFAS signal.

Systems included in category b) above are the RCP seal supply lines, the component cooling water (CCW) inlet and outlet lines and the control rod drive (CRD) cooling water lines. The RCP seal supply and the CCW provide cooling for RCP seals to prevent seal damage that could result in a small LOCA. The seal return is isolated and check valves prevent back flow from the seal injection line. Thus, primary coolant would not be released via this route. The CCW and CRD cooling water systems are closed systems not in contact with primary coolant with capability for manual isolation if required.

The Rancho Seco FSAR description of the isolation provisions of the CCW, CRD supply and return, and RCP seal injection include automatic isolation on SFAS. However, after issuance of the operating license, the licensee eliminated the automatic isolation portion of these systems. The licensee determined that this change did not constitute an unreviewed safety question and made the change under the provisions of 10 CFR 50.59, without prior review by the NRC. The NRC is presently considering if this change did constitute an unreviewed safety question and must be reviewed. The results of this review will determine whether or not the licensee will be required to reestablish automatic isolation of these systems. Because of the special requirements for use of these systems following certain upset conditions, isolation based on minimum of a single parameter may be acceptable.

The SFAS signal which isolates all other non-essential systems is generated by diverse parameters: a) RCS pressure less than 1600 psig or b) containment pressure greater than 4 psig.

Penetrations controlled by remotely operated valves receive containment isolation signals, whether they are open or closed during normal operation. Penetrations controlled by local manual valves which are closed during normal operation are locked closed. The containment isolation valves do not reopen automatically if the containment isolation signal clears. Manual action is required.

The automatic containment isolation valve controls utilize a manual/automatic mode select switch and an open/close select switch mounted together for each valve. Following containment isolation, the operator can reopen any valve by first selecting manual mode and then pushing the open button. This is possible whether or not the containment isolation signal has cleared. Selection of manual mode does not in itself open the valve.

We conclude that the licensee has satisfied the requirements of this item. Review of the CCW, CRD supply and return, and RCP seal injection isolation provisions is continuing. Verification of the adequacy of the procedures will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.5.a DEDICATED H_2 CONTROL PENETRATIONS

Rancho Seco was licensed to use a hydrogen purge system for post-accident combustible gas control of containment atmosphere. The hydrogen purge system meets the original licensing requirements for containment isolation. The licensee has submitted a design to have the system meet the single failure criteria for both containment isolation and operation of the system. Plant modifications are Category "B" requirements which should be completed by January 1981.

Based on the above, we conclude for Rancho Seco that the licensee has met the Category "A" requirements of this item.

2.1.5.b INERTING BWR CONTAINMENTS

This item does not apply to Rancho Seco.

2.1.5.c H_2 PURGE PROCEDURES

The licensee has reviewed the present procedures and shielding for operating the hydrogen purge system. This system provides post-accident combustible gas control of the containment atmosphere. It is operated near the equipment, at the mezzazine level of the Auxiliary Building outside the control room, because valves in the system must be manually operated. The equipment is located next to containment in an area which may not be accessible for some time after an accident. Existing procedures do cover the operation of the system provided the radiation level in the area is below that permitted by GDC 19.

The licensee has proposed to modify the hydrogen purge system to provide operation of the system from the control room or an accessible location. These modifications include replacing manually operated valves with remote operated valves and providing for remote operation of the hydrogen purge blowers. Plant modifications are Category "B" requirements which should be completed by January 1981.

We conclude that the licensee has met the Category "A" requirements for this item. Verification of the licensee's procedures will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.6.a SYSTEM INTEGRITY

The licensee has listed the plant systems outside containment which would or could contain highly radioactive fluids during a serious transient or accident. These systems are the makeup and purification system, decay heat removal system, high pressure injection system, reactor building spray system, waste gas system, reactor coolant sampling system, hydrogen purge system and appropriate parts of the miscellaneous radwaste system and coolant radwaste system. The licensee has implemented an immediate leak reduction program for these systems to reduce their present leakage. The licensee has measured and reported the "as-corrected" leakage for these systems except for the makeup and purification system, high pressure injection system and the reactor coolant sampling system. The licensee will measure the leakage from these three systems before startup from the present refueling outage and will report the measured leakage within two weeks of startup.

The licensee has established a permanent leak reduction program to keep future leakage from the above systems to as low-as-practical levels. This program includes integrated leak rate tests once per refueling cycle; identification of leakage by means of visual surveillance by plant personnel and responses of area and effluent radiation monitors; and the plant preventive maintenance program.

The licensee has reviewed the plant design for potential leakage release paths from the above systems due to design and operator deficiencies as discussed in the NRR letter to the licensee regarding North Anna and Related Incidents dated October 17, 1979. The licensee will make two changes to the plant. The relief valves for the make-up filter and the reactor coolant pump seal return will be routed to more suitable tanks or sumps instead of to open floor drains and the grade level of the Auxiliary Building will be changed in a manner to prevent contaminated water from a spill from leaving the building. These changes should be completed by January 1981.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements for this item. There are no Category "B" requirements for this item. Verification of the procedures which implement the licensee's permanent leak reduction program and the plant modifications discussed above will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.6.b PLANT SHIELDING REVIEW

The licensee has performed a radiation and shielding study of the spaces near the plant systems outside containment which would or could contain highly radioactive fluids during a serious transient or accident. These systems are those discussed in the evaluation of item 2.1.6.a. The radioactive source terms assumed in the study above are consistent with the source terms given in the NRC letter to the licensee dated October 30, 1979.

The licensee is continuing its review of the results of the above study to determine that during the postulated post-accident conditions all vital plant areas are accessible and all safety equipment outside containment would not be unduly degraded by radiation. The licensee has identified the areas of the plant requiring continuous occupancy and intermittant occupancy. The licensee has described additional shielding that is needed to make these areas accessible during post accident conditions. These areas are: 1) the containment personnel hatch at 40 feet elevation and the containment emergency hatch at grade level must be shielded and 2) the east wall of the Diesel Generator Room may require shielding. The licensee states that post-accident radiation levels in the control room, the interim Technical Support Center and the onsite Operational Support Center are acceptable for continuous occupancy.

The licensee is continuing its review of the radiation qualification of safety equipment outside containment which may be unduly degraded by radiation during post-accident conditions. The licensee has identified the safety equipment, both electrical and mechanical, located outside containment and has listed the integrated dose to the equipment over 200 days during an accident. This exposure must be compared to the dose the equipment is qualified to or can be qualified to. The results of this review for both electrical and mechanical equipment will be reported in the licensee's response to IE Bulletin 79-01B. The licensee will also report to NRC the plant modifications, if any, which are needed to protect the safety equipment; eg. additional shielding, moving equipment or replacing equipment.

We conclude that the licensee has met the Category "A" requirements for this item. Plant modifications are Category "B" requirements which should be completed by January 1981. An evaluation of the licensee's design review and any corrective actions which are proposed by the licensee in his reports to NRC will be performed as part of the review of the Category "B" requirements for this item.

2.1.7.b AUXILIARY FEEDWATER FLOW

The licensee currently has installed control grade auxiliary feedwater flow indicators to each steam generator. The instrument is a sonic device and indication is provided in the control room. The system is testable to within 10 percent at high flow conditions. Power to the system is diesel backed. The single failure criterion is satisfied by one control grade flow indication and control grade level indication on each steam generator.

The licensee plans to upgrade the system with safety grade flow indication based on a system of orifice plates and pressure sensors. Redundancy will be provided by either dual pressure sensors or upgrading of the sonic system to Class IE. The NRC position requires that the upgrading should be completed by Janaury 1981.

We conclude that the licensee has satisfied the Category A requirements of this item. Verification of the adequacy of the installation and procedures for use of the instrumentation will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.8.a POST-ACCIDENT SAMPLING

The licensee has performed a design and operational review of reactor coolant and reactor building atmosphere sampling. The reactor coolant sample station and reactor building atmosphere sample station may not be accessible for accidents which result in significant fractions of the core inventory radioactivity being released from the core. The licensee has completed its review of procedures to use the two sample stations during post-accident operations when the sample stations are accessible. The licensee is designing and will install, when equipment is available, an interim sampling system for both the reactor coolant and reactor building atmosphere which will allow personnel to obtain a post-accident sample and to minimize personnel radiation exposure.

The above system will also be the licensee's long term shielded sampling facility to obtain a post-accident sample within NRC exposure criteria. The licensee will address the design and capabilities of the long term post-accident sampling facility in a report prior to June 1, 1980. This is a Category "B" requirement.

The present sample taps to collect a reactor coolant sample are the letdown line outside the containment isolation valves, the pressurizer liquid space and gas space and the pressurizer relief tank. These taps do not allow collection of a representative reactor coolant sample during an accident without operation of the letdown into the makeup and purification system or the pressurizer power operated relief valve (PORV) into the pressurizer relief tank and maybe the reactor building. The licensee has proposed new sample taps for the reactor coolant system cold leg and the decay heat system which will not require the use of letdown or PORV.

The licensee has completed a design and operational review of the plant radiological analysis facility and chemical analysis facility under post-accident conditions. These facilities may not be accessible during accidents which result in significant fractions of the core inventory radioactivity being released from the core. For the short term, the licensee has studied how these facilities can be used during post-accident conditions when these facilities are accessible and has written procedures to minimize personnel exposure during analysis of highly radioactive samples. The critical equipment in the radiological counting facility can be relocated to the designated Plant Assembly Point. The licensee has also studied the terms and logistics for nearby government or private laboratories to analyze post-accident samples.

For the long term, the licensee has proposed a design for new facilities to provide on-site capability to analyze post-accident samples within the NRC exposure guidelines. The licensee has proposed in-line monitors to provide post-accident measurement of hydrogen concentration, oxygen concentration, conductivity and pH for reactor coolant samples and measurement of hydrogen concentration for containment atmosphere samples. Radiological analysis will be by an accessable multichannel analyzer using samples taken from the above system.

We conclude that the licensee has met the Category "A" requirements for this item. Plant modifications are a Category "B" requirement which should be completed by January, 1981. Verification of the adequacy of procedures and the installation of the interim sampling system will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.8.b HIGH RANGE EFFLUENT MONITORS

The licensee has provided an interim method to quantify high level radioactive noble gas effluent from the plant during post-accident conditions. The plant effluent stack, which includes releases from the hydrogen purge system, Auxiliary Building and radwaste service area by way of separate vents, and the secondary side main steam line, through the safety valves and steam dump valves, are the only lines to release radioactivity from the plant during a serious transient or accident which can be monitored. The licensee has designated plant areas for personnel to monitor the gamma radioactivity in the plant effluent stack and the turbine bypass line. These areas are where the plant effluent stack and turbine bypass line are located and are accessible during post-accident conditions from radiation sources other than the lines themselves. The radioactivity in the turbine bypass line will be representative of that in the main steam line when the turbine bypass valve is open. When the safety valves and dump valves are open, the turbine bypass valve is usually also open.

The licensee has procedures to measure the gamma radioactivity in the line and to convert the monitor readings to the rate of noble gas radioactivity being released.

The licensee will install a shielded radiation monitor on the plant effluent stack and the main steam line, in the area of the safety valves and dump valves, to monitor these lines. These monitors should be installed prior to June 1, 1980.

The licensee has a radioiodine/particulate sampling system for the plant effluent stack. The radioiodine/particulates are collected on cartridges which are taken to the plant radiological analysis facility for analysis.

The licensee has developed procedures to collect these cartridges during post-accident conditions; however, this sampling system is next to the Reactor Building, and, therefore, may not be accessible during post-accident conditions. The licensee has proposed a design for a post-accident radioiodine/particulate sampling system to quanitfy such releases from the plant ventilation line during post-accident conditions and keep personnel exposure within NRC exposure guidelines. This is a Category "B" requirement and should be completed by January, 1981.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements for this item. Verification of the adequacy of the procedures and equipment installation to quantify high-level post-accident effluents from the plant will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.8.c IMPRÓVED IN-PLANT RADIOIODINE INSTRUMENTATION

The licensee has a gross radiation monitor and silver zeolite cartridges which are dedicated to the control room/Technical Support Center area to promptly analyze air samples for radioiodine concentrations during an accident. The equipment will be kept in the emergency equipment locker located in a room directly across from the control room.

Based on this, we conclude that the licensee has met the requirements for this item. There are no Category "B" requirements. Verification that the licensee has the above equipment dedicated to the control room/Technical Support Center area, that the equipment is periodically checked and calibrated and the adequacy of the procedures and training of personnel in the use of the equipment will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.2.1.a SHIFT SUPERVISOR RESPONSIBILITIES

The licensee has revised plant procedures as necessary to set forth the responsibilities of the Shift Supervisor such that he can provide direct command oversight of operations and perform review of ongoing operations that are important to safety and not be distracted from these important responsibilities by administrative assignments.

We conclude that the licensee has satisfied our requirements for the Shift Supervisor. Verification of the adequacy of the licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented by an appropriate inspection report.

2.2.1.b SHIFT TECHNICAL ADVISOR

For the interim period of 1980, the licensee has provided an on-shift technical advisor (STA) to assist the shift supervisor in the function of accident assessment. The current personnel are plant staff graduate engineers. They will serve a 24 hour duty day on a rotating basis and will be on-site at all times during this duty. The STA will perform other duties when not required in the control room. A shift technical advisor will be assigned whenever the reactor is not in cold shutdown. He will be accessible to the control room within 10 minutes of notification and will be present during all shift supervisor reliefs and will be called to the control room during unanticipated transients and important plant evolutions. In addition, an on-call Technical Advisor position has been established, separate from the STA. These are senior licensed members of the management staff who, when on designated duty, are available on-call to respond to the STA to provide additional expertise and assistance, by phone or radio. The on-call Technical Advisor will report to the site during emergencies to assist the Shift Supervisor.

The interim training will include courses on small breaks, electrical distribution, reactor simulator, and general reactor coolant system behavior.

For the interim, the operational experience function will be performed by Technical Support staff and the Quality Control organization and the Technical Assistant to the Manager of Nuclear Operations. LER review will be done by contract engineers.

For the long term, the STA position will continue to be filled by graduate engineers with training beginning in summer 1980. Approximately 20 weeks of training will be provided, directed mainly to response to small breaks.

We conclude that the licensee has satisfied the requirements of this item. Verification of the adequacy of the procedures that implement the position will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.1.c SHIFT AND RELIEF TURNOVER PROCEDURES

The licensee has revised plant procedures to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the on-coming shift.

Based on the above considerations, we conclude that the licensee has satisfied the requirements of Item 2.2.1.c to provide new procedures. Verification of the adequacy of the implemented procedures will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.a CONTROL ROOM ACCESS

The licensee has implemented procedures which will limit control room access during an emergency. These procedures establish the authority and responsibility of the person in charge of the control room to control access and establish the line of succession for the person in charge of the control room.

We conclude that the licensee has satisfied the requirements of this item. Verification of the adequacy of the licensee's procedures will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.b ON-SITE TECHNICAL SUPPORT CENTER (TSC)

The licensee has established an onsite technical support center in a conference room adjacent to the control room. A window separates the conference room from the control room. Direct communications between the TSC and the control room and NRC have been established. The control room and the TSC have common air conditioning and airborne radiation monitoring systems. The TSC currently contains drawing files which include piping and instrumentation diagrams and electrical one-line diagrams. Plant parameters can be obtained by going to the terminals in the computer room which is located directly across from the TSC. The terminals provide access to information useful in assessing accident situations and this access is controlled by established procedures. For the long term, the licensee will convert the existing Instrumentation and Control shop into the permanent TSC to meet our requirements. This will be reviewed as a Category "B" item.

Based on our review of the licensee's submittal and our site visit, we have concluded that the TSC at Rancho Seco satisfies our Category A requirements. Verification of the adequacy of the licensee's procedures for activation and use of the short term TSC will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.c OPERATIONAL SUPPORT CENTER

The licensee has established the control room kitchen next to the TSC as the on-site operational support center. Operations Support Personnel will be located in the OSC for support of the control room and/or TSC needs. The on-site operations support center is too small to house all operations support personnel. Additional personnel such as maintenance and health physics technicians will gather at the Plant Assembly Point as described in the Rancho Seco Emergency Plan. Procedures have been written to interface the on-site operations support center and the Plant Assembly Point including specifying the personnel who are to report to each area. Communication with the control room, the on-site operational center and the Plant Assembly Point is provided.

Based on our review of the licensee's submittal and our site visit, we conclude that the licensee has satisfied our requirements for this item. Verification of the adequacy of the licensee's procedures to implement this center will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

REACTOR COOLANT SYSTEM HIGH POINT VENTS (Imposed by September 13, 1979 NRR letter)

The licensee has committed to install remotely operated reactor coolant system vents. The proposed design includes vents on the high points of the hot leg loops and on top of the pressurizer and on the reactor vessel head. The hot let vents will utilize one vent path for each hot leg. The pressurizer vent and reactor vessel head vent will utilize two parallel vent paths each. Each vent path will be controlled by two solenoid operated valves in series. The valves will be seismically qualified and powered from Class 1E power supplies. The valves fail closed on loss of power.

We conclude that the licensee has satisfied the Category "A" requirements of this item.

<u>REFERENCES</u>

1. Letter, NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, dated September 13, 1979.

2. Letter, NRC (Denton) to ALL OPERATING NUCLEAR POWER PLANTS, dated October 30, 1979.

3. Letter, SMUD (Mattimore) to NRC (E/NRR) dated October 18, 1979.

4. Letter, SMUD (Mattimore) to NRC (R/NRR) dated November 19, 1979.

5. Letter, SMUD (Mattimore) to NRC (R/NRR) dated November 26, 1979.

6. Letter, SMUD (Mattimore) to NRC (R/NRR) dated January 7, 1980.

7. Letter, SMUD (Mattimore) to NRC (D/NRR) dated January 16, 1980.

8. Letter, SMUD (Mattimore) to NRC (R/NRR) dated March 5, 1980.

9. Letter, SMUD (Mattimore) to NRC (R/NRR) dated April 11, 1980.

10. Letter, SMUD (Mattimore) to NRC (R/NRR) dated April 17, 1980.