ENCLOSURE 1

SAFETY EVALUATION REPORT

RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

CONFORMANCE TO REGULATORY GUIDE 1.97

1.0 INTRODUCTION AND SUMMARY

The licensee, Sacramento Municipal Utility District was requested by Generic Letter 82-33 to provide a report to the NRC describing how the post-accident monitoring instrumentation meets the guidelines of Regulatory Guide (RG) 1.97 as applied to emergency response facilities. The licensee's response to Regulatory Guide 1.97 was provided by the letters listed in the References at the end of this report.

Detailed reviews and technical evaluations of the licensee's submittals were performed by EG&G Idaho, Inc. under contract to the NRC, with general supervision by the NRC staff. The initial report on this work was transmitted as an enclosure to our memorandum dated June 28, 1985 to J.F. Stolz, Chief, Operating Reactors Branch No. 4, which identified eleven exceptions to RG 1.97 recommendations which were not acceptable. The licensee responded with additional information which was reviewed, evaluated, and reported by EG&G in their Technical Evaluation Report (TER), "Conformance to Regulatory Guide 1.97, Rancho Seco Nuclear Generating Station, Revision 2" dated November 1986 (enclosed). We have reviewed this report and concur with the conclusion that the licensee either conforms to, or has justified deviations taken from the guidance of RG 1.97 for each post-accident monitoring variable except for the qualification of the safety parameter display system (SPDS) to serve as indication for the Category I variables.

2.0 EVALUATION CRITERIA

Subsequent to the issuance of the generic letter, NRC held regional meetings in February and March 1983 to answer the licensee's questions and concerns regarding NRC policy on RG 1.97. At these meetings, it was established that NRC review would address only exceptions taken to the guidance of RG 1.97. Further, where the licensee explicitly stated that instrument systems conform to the provisions of the regulatory guide, no staff review would be necessary. Therefore, the review performed and reported by EG&G only addresses exceptions to the guidance of the regulatory guide. This safety evaluation addresses the licensee's submittals based on the review policy described in the NRC regional meetings and the conclusions of the review as reported by EG&G.

3.0 EVALUATION

In our interim report of June 28, 1985, we identified eleven exceptions to the recommendations of RG 1.97 which were found not acceptable. The licensee was advised of these findings and responded with additional information which was reviewed and evaluated by the contractor. The results of this review were reported in Revision 2 of TER EA-6940 which found that the licensee had either conformed to the R.G. 1.97 recommendations or had provided acceptable justification for deviating from those recommendations for each post-accident monitoring variable except for the radiation level in the circulating primary coolant, accident sampling (primary coolant, containment air and sump) system, and the provision of Category 1 indicators for the Category 1 variables.

For the measurement of radiation level in circulating primary coolant, one of the identified means of measurement was the post-accident sampling system (PASS) which was stated to be under review by the staff as part of the NUREG-0737, Item II.B.3 issue. In their safety evaluation report of July 28, 1983 the staff found the PASS met both the Item II.B.3 requirements and the RG 1.97, Revision 2 recommendations and was, therefore, acceptable. The Revision 2 recommendations requested a maximum sensitivity for the radioactivity determination of 10 μCi/ml. In their submittal of July 13, 1984 (Reference 6) the licensee committed to meet the RG 1.97, Revision 3 recommendations which changed the maximum requested sensitivity for the radioactivity determination from 10 μCi/ml to 1 μCi/ml. In that submittal, the licensee stated that PASS was capable of meeting the new maximum sensitivity, and the staff concluded in their safety evaluation report of March 22, 1985 that the PASS was acceptable for NUREG-0737, Item II.B.3. The staff concludes in this SER that the stated sensitivity of 1 μCi/ml meets the recommendation of RG 1.97, Revision 3 and is acceptable.

For the accident sampling (primary coolant, containment air and sump) variable, the PASS is used to meet this RG 1.97 recommendation and, as above, this system was stated to be under review by the staff as part of the NUREG-0737, Item II.B.3 issue. Since this is the same system with similar requirements, as evaluated above, that evaluation applies for this variable also; we find the PASS meets the recommendations of RG 1.97, Revision 3 for this variable and is, therefore, acceptable.

With respect to provision of continuous real-time display of Category I variables, the licensee proposed to use the SPDS which would be upgraded to Category I requirements to meet this recommendation. The licensee has committed in Reference 6 to provide both hardware and software which will meet the Category I requirements. The staff review of the capability of the licensee's hardware and software to meet the Category I requirements is being performed separately and the review results will be reported in a separate SER. On the basis that the licensee has committed to make the SPDS meet Category I requirements, and the staff will, through their review, ensure that the SPDS meets those requirements, we consider this concern resolved.

4.0 CONCLUSION

Based on the staff's review of the enclosed TER and the licensee's submittals, we find that the Rancho Seco design is acceptable with respect to conformance to RG 1.97, Revision 3.

An appropriate implementation schedule will be developed by the project manager through discussion with the licensee. Once the schedule is established, the licensee is required to inform the Commission, in writing, of any significant changes to the implemented system from that approved in the staff's safety evaluation and when the implementation has actually been completed.

5.0 REFERENCES

1. NRC Letter, D. G. Eisenhut to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, NRC, Office of Standards Development, December 1980.

3. Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737, Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Sacramento Municipal Utility District (SMUD) letter, R. J. Rodriguez to Director of Nuclear Reactor Regulation, NRC, "Generic Letter No. 82-33, Supplement 1 to NUREG-0737," April 15, 1983.

5. SMUD letter, J. J. Mattimoe to Director of Nuclear Reactor Regulation, NRC, "Regulatory Guide 1.97 Comparison Report," September 14, 1983.

6. SMUD letter, R. J. Rodriguez to Director of Nuclear Reactor Regulation, NRC, "NUREG-0737 Supplement 1--Regulatory Guide 1.97", July 13, 1984.

7. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 3, NRC, Office of Nuclear Regulatory Research, May 1983.

8. SMUD letter, R. J. Rodriguez to Director of Nuclear Reactor Regulation, NRC, "NUREG-0737, Supplement 1 - Status of Open Items for Implementation," September 30, 1985, RJR 85-470.

9. SMUD letter, R. J. Rodriguez to Director of Nuclear Reactor Regulation, NRC, "Regulatory Guide 1.97, Request for Additional Information," October 31, 1985, RJR 85-521.

10. SMUD letter, R. J. Rodriguez to Director of Nuclear Reactor Regulation, NRC, "Regulatory Guide 1.97 Request for Additional Information," January 13, 1986, RJR 86-11.

11. SMUD letter, R. J. Rodriguez to Director of Nuclear Reactor Regulation, NRC, "Regulatory Guide 1.97 Implementation Schedule," March 7, 1986, RJR 86-93.

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INFORMAL REPORT

Idaho
National
Engineering
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Managed
by the U.S.
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CONFORMANCE TO REGULATORY GUIDE 1.97, RANCHO SECO

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TECHNICAL EVALUATION REPORT

CONFORMANCE TO REGULATORY GUIDE 1.97
RANCHO SECO

Docket No. 50-312

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FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of PWR Licensing-A, by EG&G Idaho, Inc., NRR and I&E Support Branch.

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CONFORMANCE TO REGULATORY GUIDE 1.97

RANCHO SECO

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement No. 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

The Sacramento Municipal Utility District, licensee for the Rancho Seco Nuclear Generating Station, provided a response to the generic letter on April 15, 1983 (Reference 4). The response to Section 6.2 of the generic letter was submitted on September 14, 1983 (Reference 5), and revised on July 13, 1984 (Reference 6). This last response provides a comparison of the licensee's instrumentation to the recommendations of Revision 3 of Regulatory Guide 1.97 (Reference 7). Additional information was submitted on September 30, 1985 (Reference 8), October 31, 1985, (Reference 9), January 13, 1986 (Reference 10) and March 7, 1986 (Reference 11).

This report provides an evaluation of this material.

this report only addresses exceptions to Regulatory Guide 1.97. The following evaluation is an audit of the licensee's submittals based on the review policy described in the NRC regional meetings.

4. Steam generator level

5. Steam generator pressure

This instrumentation meets the Category 1 recommendations for Type A variables.

3.3 Exceptions to Regulatory Guide 1.97

The licensee identified the following deviations and exceptions to Regulatory Guide 1.97. These are discussed in the following paragraphs.

3.3.1 NUREG-0737 Instrumentation

The licensee has installed instrumentation for several variables in accordance with the requirements of NUREG-0737. For some, the range differs from that recommended by the regulatory guide, others deviate from the regulatory guide in the instrument category. These variables are listed below. The licensee states that this instrumentation has been reviewed and approved by the NRC. The licensee has referred to this approval in addition to his justification for any deviation.

- o Degrees of subcooling
- o Analysis of primary coolant
- o Containment sump water level, wide range
- o Containment hydrogen concentration
- o Primary system safety relief valve position
- o Pressurizer safety/relief valve position
- o Noble gas and vent flow rate--auxiliary building

As the licensee has supplied Category 1 core exit thermocouples, we find this justification for Category 3 RCS cold leg water temperature instrumentation acceptable.

The licensee justifies the upper limit of the range based on the . highest possible temperature of 560°. This takes into account the highest main steam safety relief valve setting of 1102.5 psig. As the instrumentation will remain on scale in the post-accident situation, we find this range acceptable.

3.3.4 RCS Hot Leg Water Temperature

Regulatory Guide 1.97 recommends instrumentation with a range of 50 to 700°F for this variable. The licensee has supplied instrumentation with a range of 120 to 920°F. The licensee states that the low end of the range for RCS hot leg temperature is not important to post-accident monitoring because cold shutdown is defined in the Technical Specifications as less than 200°F. With the RCS temperature between 50 and 120°F, it is cold enough for refueling, therefore, it is in a safe condition.

Additionally, heat removal at these temperatures would be by the residual heat removal (RHR) system rather than the steam generators. This system has instrumentation to monitor the temperature of the RCS in this temperature range. We therefore find this deviation acceptable.

3.3.5 Radiation Level in Circulating Primary Coolant

The licensee indicates that radiation level measurements to indicate fuel cladding failure are provided by the following:

1. Letdown line radiation monitor

2. Radiochemistry analysis

3. Post-accident sampling system.

The licensee notes that the normal water level in these tanks is 13.0 feet (156 inches), a high level alarm is set at 159.6 inches and the tank would not be filled above this level. Because the water level is maintained at less than the upper limit of the range (169 inches), and the lower limit of the range is less than the recommended limit, we find tbat the range is acceptable.

The licensee states that the core flood tank pressure instruments are used as backup instrumentation, and that the key variable to indicate proper operation of the core flood tanks is the level instruments.

The accumulators are passive devices. Their discharge into the reactor coolant system (RCS) is actuated solely by a decrease in RCS pressure. We find that the instrumentation supplied for this variable is adequate to determine that the accumulators have discharged. Therefore, this instrumentation is acceptable.

3.3.9 Accumulator Tank Isolation Valve Position

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The licensee states that these valves are opened during normal plant heatup. Prior to criticality, the valves are verified to be in the operating position, and they do not change position during the course of an accident. Furthermore, the licensee observes that the position indication for these valves is diagnostic in purpose. Therefore, the licensee has supplied Category 3 instrumentation for this variable.

We find that Category 3 instrumentation for this variable is acceptable.

3.3.10 Boric Acid Charging Flow

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The licensee has Category 3 instrumentation. The licensee states that two independent sources and multiple paths exist for lower shell and will maintain steam pressure above the high pressure injection (HPI) system actuation setpoint. Additionally, the licensee states that the RCS can experience a turbine trip without covering the level sensors in the upper shell. Thus, the range allows level monitoring to ensure proper operation of pressurizer heaters. The licensee states that it is adequate for the purpose of determining RCS leakage and voiding.

In Reference 10, the licensee states that the existing range is sufficient to remain on scale for anticipated transients. For severe accidents or transients, the pressurizer will either void or go solid. This would cause the pressurizer level indication to go off-scale low or high depending on the accident or transient, regardless of the span of the range. In these cases of off-scale pressurizer instrumentation, action to be taken must be determined by subcooling margin, reactor coolant system pressure, power operated relief valve status and pressurizer safety valve status. These indications are all available in the control room.

Based on the licensee's justification and the alternate instrumentation available, we conclude that indication of the pressurizer level outside of the supplied range will provide no significant additional information. Therefore, we find this to be an acceptable deviation from Regulatory Guide 1.97.

3.3.13 Pressurizer Heater Status

Regulatory Guide 1.97 recommends instrumentation to monitor the current drawn by the pressurizer heaters. The licensee's instrumentation consists of on/off indication of the redundant emergency pressurizer heaters. In Reference 10, the licensee commits to provide current instrumentation for the emergency pressurizer heaters, prior to Cycle 9 startup. We find this commitment acceptable in meeting the recommendations of Regulatory Guide 1.97.

3.3.14 Quench Tank Level

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the top to the bottom of the tank. The overall height of is 235 psig. The instrumentation for this variable has a range of 0 to 200 psig. The licensee states that this is adequate since the tank rupture disk set pressure is 180 psig.

Based on the rupture disk set pressure, we find that the range of 0 to 200 psig is acceptable and adequate.

3.3.17 Steam Generator Pressure

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from 0 to 20 percent above the lowest safety valve setting. The lowest safety valve setting is 1050 psig; therefore the range should be from 0 to 1260 psig. The instrumentation for this variable has a range of 0 to 1200 psig, 9 percent above the highest safety valve setting.

The licensee states that the upper limit of the range of the instrumentation is 9 percent above the highest setting of the safety relief valves and that the pressure-relief capacity is 20 percent greater than required to relieve the steam flow at maximum power.

Based on this statement, and the maximum range being nearly 100 psi above the highest safety valve setting, we find that the range of 0 to 1200 psig is acceptable.

3.3.18 Containment Spray Flow

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. In Reference 9, the licensee describes the instrumentation for this variable. The safety-grade containment spray system is actuated automatically by high containment pressure. Thus, the licensee considers the Category 1 reactor building pressure instrumentation the key variable to indicate operation of this containment cooling system. Reactor building pressure and reactor building temperature (Category 2) show the effects of the spray. Pump and valve position are monitored to indicate system operation. Finally, Category 3 flow transmitters are indicated in the control room as a backup variable.

3.3.22 Letdown Flow-Out

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The licensee has Category 3 instrumentation. The licensee states that maintaining letdown flow is not essential to the mitigation of any design basis accident. Furthermore, in the event that the Safety Features Actuation System (SFAS) is initiated, letdown flow is isolated.

As this flow is isolated as a result of an accident signal, Category 3 instrumentation for this variable is acceptable.

3.3.23 Volume Control Tank Level

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the top to the bottom of the tank. The licensee does not consider this as post-accident instrumentation; however, the range of this instrumentation covers from 29 to 129 inches of the 153 inch tank height The level is maintained within this range.

The range supplied essentially covers the straight cylindrical shell, not monitoring the hemispherical ends of the tank where the level to volume ratio is not linear. Approximately 78 percent of the tank volume, inclusive of the hemispherical ends is measured for level. Based on this, and the licensee's justification for not requiring this instrumentation in a post-accident situation, we find this deviation in range acceptable.

3.3.24 Component Cooling Water Temperature to Engineered Safety Feature (ESF) System Components

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The licensee has Category 3 instrumentation installed. Cooling water for the ESF components is provided by the Nuclear Service Raw Water (NSRW) system. The heat from this system is transferred to the atmosphere by spray ponds. The Nuclear Service Cooling Water (NSCW) system is cooled by the NSRW system. The spray ponds provide a source of low temperature coolant for the NSCW and NSRW systems.

There is a safety relief valve on this tank, set to relieve any pressure above 145 psig. As the tank pressure will not exceed 160 psig, we find this range acceptable.

3.3.27 Estimation of Atmospheric Stability

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of -9 to +18°F or an analogous range for alternative stability analysis. The licensee has supplied instrumentation with a range of -10 to +10°F. The licensee justifies this, indicating that the range is based on RG 1.23, Rev. 1, Table 1, 'Classification of Atmospheric Stability by Temperature Change with Height'.

Table 1 of Regulatory Guide 1.23 provides seven atmospheric stability classifications based on the difference in temperature per 100 meters elevation change. These classifications range from extremely unstable to extremely stable. Any temperature difference greater than +4°C or less than -2°C does nothing to the stability classification. The licensee's instrumentation encompasses this range. Therefore, we find that the instrumentation is acceptable to determine the atmospheric stability.

3.3.28 Accident Sampling (Primary Coolant, Containment Air and Sump)

The licensee's post-accident sampling system provides sampling and analysis as recommended by the regulatory guide, except that

1. It does not have the capability to analyze for dissolved oxygen, using total gas instead, and

2. It does not have containment air oxygen content analysis on-site, as no action is planned based on this parameter.

The licensee deviates from Regulatory Guide 1.97 with respect to post-accident sampling capability. This deviation goes beyond the scope of this review and is being addressed by the NRC as part of their review of NUREG-0737, Item II.B.3.

4. CONCLUSIONS

Based on our review, we find that the licensee either conforms to or is justified in deviating from Regulatory Guide 1.97. This report does not address the adequacy of the SPDS.