CONFORMANCE TO REGULATORY GUIDE 1.97
LA SALLE COUNTY STATION, UNIT NOS. 1 AND 2

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ABSTRACT

This EG&G Idaho, Inc., report provides a review of the submittals for Regulatory Guide 1.97, Revision 2, for the LaSalle County Station, Unit Nos. 1 and 2. Any exception to the guidelines of Regulatory Guide 1.97 are evaluated and those areas where sufficient basis for acceptability is not provided are also identified.

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 Systems Integration, by EG&G Idaho, Inc., NRC Licensing Support Section.

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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 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

Commonwealth Edison, the licensee/applicant for the LaSalle County Station, provided a response to the generic letter on April 14, 1983 (Reference 4). The letter referred to a previous letter dated June 29, 1982 (Reference 5) for a review of the instrumentation provided for Regulatory Guide 1.97.

This report provides an evaluation of this material.

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement 1, sets forth the documentation to be submitted in a report to the NRC describing how the licensee meets the guidance of Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range

2. Environmental qualification

3. Seismic qualification

4. Quality assurance

5. Redundance and sensor location

6. Power supply

7. Location of display

8. Schedule of installation or upgrade.

Further, the submittal should identify deviations from the guidance in the regulatory guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this matter. At these meetings, it was noted that the NRC review would only address exceptions taken to the guidance of Regulatory Guide 1.97. Further, where licensees or applicants explicitly state that instrument systems conform to the provisions of the guide it was noted that no further staff review would be necessary.

Therefore, this report only addresses exceptions to the guidance of 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.

3. EVALUATION

The licensee provided a response to the NRC generic letter 82-33 on April 15, 1983. This response referred to an earlier submittal of June 29, 1982, which described the licensee's position on post-accident monitoring instrumentation. This evaluation is based on this material.

3.1 Adherence to Regulatory Guide 1.97

The licensee stated that their concurrence with the Revision 2 (Regulatory Guide) guidance is acknowledged unless their submittal discussed a particular variable. Therefore, it is concluded that the licensee has provided an explicit commitment on conformance to the guidance of Regulatory Guide 1.97. Except for those deviations that were justified by the licensee as noted in Section 3.3. The information provided by the licensee (Reference 6) pre-dated Generic Letter No. 82-33 and did not include the information identified as items 1 through 8 (e.g., instrument range etc.) in Section 2 of this report. This information should be provided to document the licensee's commitment on conformance to Generic Letter No. 82-33.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide information required to permit the control room operator to take specific manually controlled safety actions. The licensee has classified the following instrumentation channels as Type A variables.

1. Hydrogen concentration

2. Reactor pressure vessel pressure

3. Suppression cool water temperature

4. Suppression cool water level

5. Drywell pressure.

All of the above variables are also included as Type B, C or D variables and meet Category 1 requirements consistent with the requirements for Type A variables.

3.3 Exceptions to Regulatory Guide 1.97

The licensee identified the following deviations from the recommendations of Regulatory Guide 1.97.

3.3.1 Neutron Flux

The licensee has supplied Category 2 instrumentation for this variable that has a range of 5×10^{-5} to 100 percent of full power. Regulatory Guide 1.97 specifies Category 1 instrumentation with a range of 10^{-6} to 100 percent of full power. Thus, the instrumentation supplied for this variable meets neither the range nor the category requirements of the regulatory guide.

The licensee notes that accident scenarios resulting in an increase in reactivity could only be caused by inadvertent removal of boron that was added by the standby liquid control system or by other effects such as a change in temperature or fission product poisoning. Since these reactivity additions would likely have a slow rate of change, the licensee concludes that power readings in the range of 10^{-5} percent of full power would give the operator sufficient time (21.6 minutes) to identify the problem and take corrective action before the power reached 0.5 percent of full power. The licensee has not shown that this time is within the time permitted by Standard Review Plan 15.4.6 (i.e., 15 minutes from alarm to loss of shutdown margin). We find that the justification provided by the licensee for a deviation in the range recommendations for neutron flux is incomplete, and therefore the deviation is not acceptable.

The licensee has proposed some upgrading of the neutron flux instrumentation; however, they have not identified which of four proposed options they will im... ment.

In the process of our review of neutron flux instrumentation, we note that the mechanical drives of the detectors have not satisfied the environmental qualification requirement of Regulatory Guide 1.97. This deviation is similar to most BWRs. A Category 1 system that meets all the criteria of Regulatory Guide 1.97 is an industry development item. Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The licensee should follow industry development of this equipment, evaluate newly developed equipment, and install Category 1 instrumentation when it becomes available.

3.3.2 Drywell Sump Level Drywell Drain Sumps Level

The licensee has provided Category 3 instrumentation for this variable that provides continuous level indication, high level alarm and high-high level alarm. Regulatory Guide 1.97 requires Category 1 instrumentation with indication from the bottom to the top of each sump. The deviation for this variable is in the category of the supplied instrumentation. This instrumentation refers to the drywell equipment drain tank and floor drain tank levels at the LaSalle station. This instrumentation does not cause any automatic or operator initiated safety related functions. The sump systems are automatically isolated in an accident signal as part of containment isolation. This prevents the pump-out of the drain tank contents.

For small leaks, this Category 3 instrumentation will continue to function as the drywell temperature and pressure will not have changed significantly. Therefore, the sump levels can be used as a leading indicator of reactor coolant system leakage. For larger leaks, the sumps will fill promptly, negating this information because the sumps isolate due to the increase in drywell pressure caused by the accident. The sumps can be assumed full with Category 3 instruments once containment isolation occurs at 2 psig.

In either case, we find the Category 3 instruments provided for this variable acceptable.

3.3.3 Radiation Level in Circulating Primary Coolant

Regulatory Guide 1.97 specifies Category 1 instrumentation for this variable with a range of from 1/2 to 100 times the technical specification limit in units of R/hr. A direct measurement of this variable is not provided. The licensee indicates that radiation level measurements to indicate fuel cladding failure are provided by the following instruments:

1. High radiation sampling system

2. Condenser off-gas radiation monitors

3. Main steamline radiation monitors

4. Primary containment radiation monitors

5. Containment hydrogen concentration monitors

6. Area radiation monitors.

Based on the justification provided by the licensee, we conclude that the instrumentation supplied for this variable is adequate, and therefore, acceptable.

3.3.4 Radiation Exposure Rate

Regulatory Guide 1.97, Revision 2, specifies Category 2 instrumentation for this variable with a range of 10^{-1} to 10^{4} R/hr. The licensee has provided instrumentation for this variable with ranges that vary, dependent on location, from the recommended range. The licensee states that access is not required to service equipment important to safety in secondary containment in a post-accident situation, and that existing Category 3 radiation exposure

rate monitors (rather than Category 2) that have ranges that are lower than recommended by Regulatory Guide 1.97 are used only where radiation levels are mild. These are stated as being influenced by piped radioactive fluids. The licensee concludes that this makes it impractical to detect primary containment breach by use of these monitors, and that Category 3 instrumentation is suitable for this application. Further, the licensee states that the stack gas monitor is adequate to monitor the effluent from the secondary containment. The licensee determines the habitability of secondary containment by a combination of atmosphere sampling and portable radiation survey instruments, not fixed location radiation exposure rate meters.

Regulatory Guide 1.97, Revision 3 (Reference 6), changes this variable to Category 3. Therefore, the only deviation at the LaSalle station for this variable is the range supplied for a given location. The licensee has not shown any analysis of radiation levels expected for the monitor location.

The licensee should show that the existing radiation exposure rate monitors have ranges that encompass the expected radiation levels in their locations.

3.3.5 Suppression Chamber Spray Flow
Drywell Spray Flow

The residual heat removal (RHR) system flow is used for these variables. Both sprays derive their flow from the RHR system, with a throttling valve proportioning the flow between the two sprays. The position of the throttling valve is controlled from the control room. Pressure and temperature changes in the drywell and suppression chamber determine the effectiveness of the spray.

The licensee concludes that the RHR flow, and suppression chamber and drywell temperature and pressure, accurately and reliability measure the effectiveness of the drywell and suppression chamber spray. Additionally, the position of the throttling valve is known in the control room. We find that this instrumentation is adequate for this variable.

3.3.6 Reactor Core Isolation Cooling (RCIC) Flow
High Pressure Core Spray (HPCS) Flow
Low Pressure Core Spray (LPCS) Flow
Low Pressure Coolant Injection (LPCI) Flow

Regulatory Guide 1.97 recommends instrumentation for these variables with a range of 0 to 110 percent of design flow. The licensee notes that flow could be diverted into a test line downstream of the flow element for the RCIC, LPCS and HPCS systems or into RHR branch lines downstream of the flow element for the LPCI system. The concern is that the operator would not have an accurate measurement of flow to the core. The test lines and RHR branch lines have motor-operated valves that are normally closed (two valves in series for HPCS). The valves are aligned automatically, or manually per technical specification requirements. These valves receive a signal to close automatically when the system is actuated. The position of each valve is indicated in the control room. The licensee concludes that the existing flow measurement schemes for these variables are adequate and that they meet the intent of Regulatory Guide 1.97.

Based on the justification supplied by the licensee, we conclude that the instrumentation supplied by the licensee for these variables is adequate.

3.3.7 Status of Standby Power and Other Energy Sources

The licensee states that the regulatory guide recommendations are met for on-site sources only. The regulatory guide recommends "status indication of all standby power ac buses, dc buses, inverter output buses, and pneumatic supplies." The licensee's information does not show compliance with the regulatory guide. The licensee should document the status indication for the above power buses and supplies.

3.3.8 Reactor Building or Secondary Containment Area Radiation

The licensee has determined that the instrumentation for this variable is not needed, as the plant noble gas effluent monitors (which are Category 2

instruments) are more useful and practical in detecting or assessing primary containment leakage. For the LaSalle Mark II containment, the recommended range is 10^{-1} to 10^{4} R/hr. The licensee has not shown how the recommended range is met by the plant noble gas effluent monitors.

We conclude that the licensee should supply additional justification for not implementing this variable.

3.3.9 Plant and Environs Radiation

The licensee has calculated that the maximum range needed for this instrumentation is 500 R/hr for a worst case accident. The licensee has instrumentation with a maximum range of 10^{3} R/hr rather than the recommended 10^{4} R/hr. As the licensee has determined that the instrument range is adequate, the supplied instrumentation is acceptable.

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

The licensee's sample system can obtain samples and provide the analyses within the ranges recommended for this variable, from the reactor coolant, the containment sumps, the residual heat removal sump and the reactor water cleanup sump. The emergency core coolant system room floor drain sumps and auxiliary building sumps do not have grab sample capability.

The licensee takes exception to the guidance of Regulatory Guide 1.97 with respect to post-accident sampling capability. This exception goes beyond the scope of this review and is being addressed by the NRC as part of the 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, the guidance of Regulatory Guide 1.97 with the following exceptions:

1. The licensee should provide the information identified in Section 2 of this report to document their commitment on conformance to Generic Letter No. 82-33 (Section 3.1).

2. Neutron flux--the licensee's present instrumentation is acceptable on an interim basis until Category 1 instrumentation is developed and installed (Section 3.3.1).

3. Radiation exposure rate--the licensee should show that the ranges supplied for this variable encompass the radiation level at the instrument location (Section 3.3.4).

4. Status of standby power and other energy sources--the licensee should show that the status is monitored for all recommended power sources (Section 3.3.7).

5. Reactor building or secondary containment area radiation--the licensee should supply additional justification for this deviation (Section 3.3.8).

5. 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, U.S. Nuclear Regulatory Commission (NRC), Office of Standards Development, December 1980.

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

4. Commonwealth Edison letter, C. Reed to H. R. Denton, NRC, "Response to NUREG 0737--Supplement 1 Generic Letter No. 82-33," April 14, 1983.

5. Commonwealth Edison letter, C. W. Schroeder to A. Schwencer, NRC, "Compliance with Regulatory Guide 1.97," June 29, 1982.

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