CONFORMANCE TO REGULATORY GUIDE 1.97
BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2

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ABSTRACT

This EG&G Idaho, Inc., report provides a review of the submittals for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and identifies areas of full conformance to Regulatory Guide 1.97, Revision 2. Any exceptions to these guidelines are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to Regulatory Guide 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).

Carolina Power and Light Company, the licensee for the Brunswick Steam Electric Plant, provided a response to the generic letter on April 15, 1983 (Reference 4). A review of the instrumentation provided for Regulatory Guide 1.97 was provided in a later submittal of September 30, 1983 (Reference 5). This was revised February 1, 1984 (Reference 6) and May 8, 1984 (Reference 7).

This report provides an evaluation of these submittals.

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 in 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. A letter dated September 30, 1983, and two revisions to this letter dated February 1, 1984 and May 8, 1984, describe the licensee's position on post-accident monitoring instrumentation. This evaluation is based on those submittals.

3.1 Adherence to Regulatory Guide 1.97

Reference 6 provides the licensee's evaluation of Brunswick's position on compliance with Regulatory Guide 1.97. The licensee states that they concur "with the intent of Regulatory Guide 1.97," and they have provided position statements for each variable stating whether or not the recommendations of the regulatory guide have been met, and have provided justification for any nonconformance. Therefore, it is concluded that the licensee has provided an explicit commitment to conform to the recommendations of Regulatory Guide 1.97, except for those deviations noted and evaluated in Section 3.3 of this report.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide information required for operator controlled safety actions. The licensee, therefore, has classified the following instrumentation channels as Type A variables:

1. Reactor pressure vessel (RPV) pressure
2. RPV water level
3. Suppression pool water temperature
4. Suppression pool water level
5. Drywell pressure
6. Drywell temperature
7. Suppression pool pressure
8. Drywell and suppressior cool hydrogen and oxygen concentration.

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 except as noted in Section 3.3.

3.3 Exceptions to Regulatory Guide 1.97

The licensee identified the following exceptions to the requirements of Regulatory Guide 1.97, Revision 2.

3.3.1 Neutron Flux

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. The licensee has Category 2 instrumentation, and states that this meets the intent of the regulatory guide. In their justification, they indicate "there is little probability that there would be, simultaneously, a need for this measurement (in terms of operator action to be taken) and an accident environment in which the neutron monitoring system (NMS) would be rendered inoperable. Additionally, the large number of detectors that are driven into the core soon after shutdown makes it highly probable that one or more of the existing NMS detectors will be inserted and functioning."

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 given the following reason for not providing instrumentation for this variable at the Brunswick Steam Electric Plant. "A LOCA signal will prevent operation of the sump pumps and will close containment isolation valves to eliminate the possibility of radioactive materials leaking outside the primary containment. During and after LOCA, the drywell sumps will overflow into the suppression pool."

The sump level instrumentation is the primary method for determining flow rate resulting from identified and unidentified leakage from the primary coolant system. Operator actions are based on the source and the extent of the leakage.

The licensee should provide information describing how the level of the drywell and the drywell drain sumps are ascertained during and following an accident.

3.3.3 Radioactivity Concentration or Radiation Level in Circulating Primary Coolant

A direct measurement is not provided. The licensee states that during accident situations, primary coolant samples are taken by the Post Accident Sampling System located outside the Reactor Building. The sample is analyzed in the counting room. Results are phoned in to the Technical Support Center (TSC).

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 Suppression Pool Spray Flow

The licensee has stated he does not intend to provide this instrumentation. The licensee indicated that RHR flow can be used to monitor the operation of primary containment related systems. The licensee also indicated that drywell pressure and temperature as well as the suppression pool pressure and temperature measurements are available. The licensee committed to provide instrumentation for drywell temperature and pressure and suppression pool temperature that complies with Regulatory Guide 1.97.

Based on the above discussion we conclude that the justification for the lack of suppression spray flow instrumentation is acceptable.

3.3.5 Drywell Spray Flow

Regulatory Guide 1.97 recommends instrumentation with a range of 0 to 110 percent of design flow for this variable. The licensee has stated he does not intend to provide instrumentation for this variable based on the same justification given for the lack of instrumentation of suppression pool spray flow. As discussed in part 3.3.4 we conclude that the justification for the lack of drywell spray flow instrumentation is acceptable.

3.3.6 High Pressure Core Injection (HPCI) System Flow
Core Spray (CS) System Flow
Low Pressure Coolant Injection (LPCI) System Flow

Regulatory Guide 1.97 recommends instruments with a range of 0 to 110 percent of design flow for this variable. The licensee notes that flow could be diverted into a test line downstream of the flow-measuring element for the HPCI, CS and LPCI systems. The concern is that the operator would not have an accurate measurement of flow to the core. The test lines have motor-operated valves that are normally closed (two valves in series in the case of the HPCI). The valve in the test line closes automatically when the associated emergency core cooling system is activated. Proper valve position can be verified by a direct indication of valve position. The licensee concludes that the existing flow-measurement schemes for the HPCI, CS and LPCI are all adequate and 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 Standby Liquid Control System (SLCS) Flow

Regulatory Guide 1.97 recommends instrumentation with a range of 0 to 110 percent design flow for this variable. The licensee indicates that flow measuring devices for this manually initiated system are not provided. However, the flow could be verified by the following:

1. Observing the pumps discharge header pressure which is indicated in the control room.
2. Decrease in the level of the boric acid storage tank.
3. Reactivity change in the reactor as measured by neutron flux.
4. Squib valve continuity indicating lights.

Based on the above justification, we find that the licensee's position meets the intent of Regulatory Guide 1.97 for this variable.

3.3.8 Standby Liquid Control System Storage Tank Level

The licensee's transmitter for this variable is not environmentally qualified. Environmental qualification has been clarified since Revision 2 of Regulatory Guide 1.97 was issued. The clarification is in the environmental qualification rule, 10 CFR 50.49. It is concluded that the guidance of Regulatory Guide 1.97 has been superseded by a regulatory requirement. Any exception to this rule is beyond the scope of this review and should be addressed in accordance with 10 CFR 50.49.

3.3.9 Cooling Water Temperature to Engineered Safety Feature (ESF) Components

Regulatory Guide 1.97 recommends instrumentation with a range of 32 to 200°F for this variable. The licensee does not intend to provide indication for this variable. Cooling water is provided by an open-loop system designed for 33 to 90°F water temperature. The licensee indicates that the water is taken from the Cape Fear River via the intake canal. Since there are no heat sources between the intake canal and the ESF components, there will be no significant change in water temperature. Also, there are no operator actions based on water temperature. The licensee concludes that there is no need for this indication for post-accident monitoring. There are other indications such as cooling water flow that can be used to monitor system operation.

We concur with the justification for not providing this variable. Therefore, the deviation from the regulatory guide recommendation is acceptable.

3.3.10 Reactor Building or Secondary Containment Area Radiation

Regulatory Guide 1.97 recommends instrumentation with a range of 10^{-1} to 10^{4} R/hr for the Brunswick Mark II containment. The licensee states that high range monitoring of this variable is not required. The justification is that the reactor building vent is closed when the radiation level reaches 5 mr/hr and secondary containment atmosphere is routed through the Standby Gas Treatment (SBGT) system.

The licensee has not identified the range of this instrumentation as required by Section 6.2 of Reference 3. The licensee should identify this range, identify any deviation from the range recommended by the regulatory guide and justify any deviation.

3.3.11 Radiation Exposure Rate

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 10^{-1} to 10^{4} R/hr. The licensee indicates that the Brunswick Station is not designed to allow servicing equipment following an accident. The licensee states that this instrumentation is not required at this time per NUREG 0737, Supplement 1.

Regulatory Guide 1.97, Revision 2, is part of the guidance and requirements contained in NUREG 0737, Supplement 1. Moreover, access to equipment areas could be required after an accident even if the areas are not designed for equipment service. This instrumentation is recommended for the detection of significant releases, release assessment and long term surveillance. We conclude that the justification is not acceptable. The licensee should provide the recommended instrumentation.

3.3.12 Airborne Radiohalogens and Particulates

Regulatory Guide 1.97 recommends instrumentation with a range of 10^{-9} to 10^{-3} µCi/cc for this variable. The licensee provided instrumentation with a range of 10^{-14} to 10^{-2} Ci/cc (10^{-8} to 10^{4} µCi/cc).

We conclude that this deviation is acceptable.

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

Regulatory Guide 1.97 recommends sampling and on-site analysis capability for the reactor coolant system, containment sump, ECCS pump room sumps and other similar auxiliary building pump liquids and containment air. The licensee's post-accident sampling system provides sampling and analysis as recommended by the regulatory guide, except for the following deviations.

The recommended range and the supplied range are listed below.

1. boron content: 0 to 1000 ppm recommended; 20 to 6000 ppm supplied

2. chloride content: 0 to 20 ppm recommended; 0.5 to 20 ppm supplied

3. disolved hydrogen or total gas: 0 to 2000 cc (STP)/kg recommended; range not identified

4. dissolved oxygen: 0 to 20 ppm recommended; range not identified

The licensee indicates that sampling of the containment sump is not necessary because accident conditions will close isolation valves G16-F003, F004, F019 and F020, which prevents release of radioactivity from primary containment.

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. 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).

2. Drywell sump level--the licensee should provide information describing how the level of the drywell sump is ascertained during and following an accident (Section 3.3.2).

3. Drywell drain sumps level--the licensee should provide information describing how the level of the drywell drain sumps are ascertained during and following an accident (Section 3.3.2).

4. Standby liquid control system storage tank level--environmental qualification should be addressed in accordance with 10 CFR 50.49 (Section 3.3.8).

5. Reactor building or secondary containment area radiation--the licensee should identify the range of this instrumentation, identify any deviation from the range recommended by the regulatory guide and justify any deviation (Section 3.3.10).

6. Radiation exposure rate--the licensee should provide instrumentation in accordance with the Regulatory Guide 1.97 recommendations (Section 3.3.11).

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

4. Carolina Power and Light Company letter E. E. Utley to Director of Nuclear Reactor Regulation, "CP&L Response to NRC Generic Letter 82-33," April 15, 1983.

5. Carolina Power and Light Company letter, S. R. Zimmerman to Director of Nuclear Reactor Regulation, NRC, "Emergency Response Capability, Regulatory Guide 1.97," September 30, 1983, SERIAL: LAP-83-408.

6. Carolina Power and Light Company letter, S. R. Zimmerman to Director of Nuclear Reactor Regulation, NRC, "Emergency Response Capability, Regulatory Guide 1.97, Revision 1," February 1984, SERIAL: NLS-84-025.

7. Carolina Power and Light Company letter, S. R. Zimmerman to Director of Nuclear Reactor Regulation, NRC, "Requirements for Emergency Response Capability, Regulatory Guide 1.97, Revision 2," May 1984, SERIAL: NLS-84-202.

8. Clarification of TMI Action Plan Requirements, NUREG-0737, NRC, Office of Nuclear Reactor Regulation, November 1980.