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CONFORMANCE TO REGULATORY GUIDE 1.97 WASHINGTON PUBLIC POWER SUPPLY SYSTEM, NUCLEAR PROJECT NO. 2

A. C. Udy

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EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

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ABSTRACT

This EG&G Idaho, Inc., report provides a review of the submittals for Regulatory Guide 1.97 for the Washington Public Power Supply System, Nuclear Project No. 2. Any exceptions to these guidelines are evaluated.

Docket No. 50-397

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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NUCLEAR PROJECT NO. 2

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 Washington Public Power Supply System, the licensee for Nuclear Project No. 2, provided a response to the generic letter on April 15, 1983 (Reference 4). The letter referred to Section 7.5.2.3e of the Final Safety Analysis Report (Reference 5) for a review of the instrumentation provided for Regulatory Guide 1.97. Additional information was provided on October 8, 1985 (Reference 6) and on January 23, 1986 (Reference 7).

This report provides an evaluation of this material.

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement No. 1, sets forth the documentation to be submitted in a report to the NRC describing how the licensee complies with 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

The submittal should identify deviations from 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 subject. At these meetings, it was noted that the NRC review would only address exceptions taken to Regulatory Guide 1.97. Where licensees or applicants explicitly state that instrument systems conform to the regulatory guide, it was noted that no further staff review would be necessary. Therefore, 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.

3. EVALUATION

The licensee provided a response to NRC Generic Letter 82-33 on April 15, 1983. This response referred to the Final Safety Analysis Report (FSAR) which describes the licensee's position on post-accident monitoring instrumentation. Additional information was provided on October 8, 1985 and on January 23, 1986. This evaluation is based on this material.

3.1 Adherence to Regulatory Guide 1.97

The licensee states, in Section 7.5.2.2.3e of the FSAR, that the FSAR provides an item by item discussion on the instrumentation used to conform to Regulatory Guide 1.97. License condition 16 requires that modifications required to bring about compliance with Regulatory Guide 1.97 be complete by the end of the first refueling outage (approximately June 1986). Equipment procurement problems for the variable neutron flux may extend the schedule for that variable only to the end of the second refueling outage (Reference 8). Therefore, we conclude that the licensee has provided an explicit commitment on conformance to Regulatory Guide 1.97, except for those exceptions that are justified as noted in Section 3.3.

3.2 Type A Variables

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

1. Neutron flux

2. Coolant level in reactor

3. Reactor coolant system pressure

4. Primary containment pressure

These variables either meet or will meet the Category 1 recommendations consistent with the requirements 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 Neutron Flux

The instrumentation presently supplied by the licensee for this variable complies with the range and the Category 1 recommendations except for the four source and the eight intermediate range detector drive units that are not qualified to Category 1 requirements. These drive units remove the detector from the core when operating at power. They are only required post-accident to drive the detectors into the core. The source range detectors cover a range of 10^{-3} to 10 percent of full power in the fully withdrawn position, 10^{-7} to 10^{-3} percent of full power when fully inserted. This, according to the licensee, is sufficient to insure that the reactor is subcritical. There are eight similar intermediate range drive units and detectors which cover higher core power levels. The licensee states that if all the drive units failed, and the source range monitors remained out of core, the indicated range (minimum of 10^{-3} percent of full power) is sufficient to insure the sub-criticality of the reactor.

In the process of our review of the neutron flux instrumentation for boiling water reactors, we note that the mechanical drives of the detectors have not satisfied the environmental qualification requirements of Regulatory Guide 1.97. 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 is following industry development of this equipment, evaluating newly developed equipment, and has proposed to install Category 1 instrumentation prior to the completion of the second refueling outage.

3.3.2 Coolant Level in Reactor

Regulatory Guide 1.97 recommends instrumentation with a range from the bottom of the core support plate to either the top of the vessel or the centerline of the main steamline.

The licensee has Category 1 instrumentation that covers from -310 inches (referenced to instrument zero) to +60 inches. This is from below the bottom of the active fuel, close to the bottom of the core support plate, to 60 inches above the bottom of the dryer skirt. All system trips based on reactor vessel level and manual actions that are the result of the reactor vessel level occur within this range.

The licensee also has two channels of instrumentation, powered by a Class 1E source, that are displayed in the control room. These extend the range of the reactor vessel level instrumentation above the centerline of the main steamlines to +180 and +400 inches.

We conclude that the instrumentation supplied by the licensee for this variable is acceptable.

3.3.3 Drywell Sump Level
Drywell Drain Sumps Level

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. The licensee indicates that in a post-accident situation, the sump drain lines are isolated and the sump overflow goes to the suppression pool via downcomers.

For these variables, the licensee monitors flow between the drywell sump drains and the reactor building sumps. The reactor building sumps are monitored by level instrumentation. During an accident, the line connecting the drywell sump drains and the reactor building sump is isolated. The drywell sump drains then overflow into the suppression pool. The instrumentation cited above is Category 3 instrumentation.

We conclude that the instrumentation provided by the licensee will provide appropriate monitoring of the parameters of concern. This is based on (a) for small leaks, the instrumentation is not expected to experience harsh environments during operation, (b) for larger leaks, the sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, thus negating the drywell sump drains flow instrumentation and (c) this instrumentation neither automatically initiates nor alerts the operator to initiate the operation of a safety related system in a post-accident situation. Therefore, we find the Category 3 instrumentation provided acceptable.

3.3.4 Radiation Level in Circulating Primary Coolant

The licensee indicates that radiation level measurements to indicate fuel cladding failure are provided in the pre-isolation condition by the condenser off-gas radiation monitors and by the main steamline radiation monitors and in the post-accident condition by the post-accident sampling system. The post-accident sampling system is being reviewed by the NRC as part of their review of NUREG-0737, Item II.B.3.

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

3.3.5 Suppression Pool Water Level

The Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the bottom of the emergency core cooling system (ECCS) suction line to five feet above the normal water level. The narrow range instrumentation supplied by the licensee for this variable covers a range of ±25 inches of the normal water level.

Reference 6 also describes the wide range suppression pool water level instrumentation. The range is stated to be from below the ECCS suction lines to five feet above the normal water level. Thus, this instrumentation conforms with the regulatory guide.

3.3.6 Suppression Chamber Spray Flow

The residual heat removal (RHR) system flow is used for this variable. The suppression pool spray derives its flow from the RHR system, with a throttling valve proportioning the flow between the suppression pool spray and the drywell spray. The position of the throttling valve is controlled from the control room. Pressure and temperature changes in the suppression pool determine the effectiveness of the spray.

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

3.3.7 Drywell Atmosphere Temperature

Regulatory Guide 1.97 recommends instrumentation with a range of 40 to 440°F for this variable. The instrumentation supplied by the licensee for this variable covers a range of 50 to 400°F.

The licensee states that the maximum drywell design temperature is 340°F. The actual peak temperature would be less than this and of short duration. Based on this, the licensee's upper limit of 400°F for the post accident period is sufficient. The deviation in the lower limit is 10° out of the upper limit of 400°. This is 2.5 percent. Considering instrument accuracy, this deviation is minor. Therefore, we find the range of the instrumentation supplied for this variable acceptable.

3.3.8 Drywell Spray Flow

The residual heat removal (RHR) system flow is used for this variable. The drywell spray derives its flow from the RHR system, with a throttling valve proportioning the flow between the suppression pool spray and the drywell spray. The position of the throttling valve is controlled from the control room. Pressure changes in the drywell determine the effectiveness of the spray.

The licensee concludes that RHR flow and drywell pressure accurately and reliably measure the effectiveness of the drywell spray. Additionally, the position of the RHR system valves is known in the control room. We find that this instrumentation is adequate for this variable.

3.3.9 Residual Heat Removal (RHR) Heat Exchanger Outlet Temperature

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The licensee has provided instrumentation, that except for environmental qualification in Category 2.

The licensee states that, besides the heat exchanger outlet temperature, the inlet temperature is also monitored, recorded and annunciated in the control room, along with RHR valve position (Category 2 instrumentation) and flow. The licensee states that the RHR system is adequately monitored by this diverse instrumentation.

Additionally, the RHR service water flow (Category 2 instrumentation) is indicated in the control room. The RHR service water flow, when observed, assures that the RHR water is being cooled in the RHR heat exchangers. The heat exchanger bypass valve position is monitored by Category 2 instrumentation. This valve is used to bypass a portion of the water around the heat exchanger to regulate the RHR water temperature, and when fully closed, maximum RHR cooling occurs.

We find the above combination of instrumentation acceptable for this variable.

3.3.10 Cooling Water Temperature to Engineered Safety Features (ESF) System Components

Regulatory Guide 1.97 recommends a range of 32 to 200°F for this variable. The instrumentation supplied by the licensee for this variable has an upper limit of 150°F. The licensee states that the standby service water maximum design temperature is 95°F. Based on this, the range of 0 to 150°F is acceptable.

3.3.11 Plant and Environs Radioactivity (Portable Instrumentation)

Regulatory Guide 1.97 recommends a multichannel gamma-ray spectrometer for this variable. The licensee, in References 4 and 5, did not identify instrumentation for this variable. In Reference 6, the licensee identifies two portable multichannel gamma-ray spectrometers for this variable. We find this acceptable.

3.3.12 Estimation of Atmospheric Stability

The instrumentation supplied by the licensee for this variable covers a range of ±15°F instead of the range recommended by the regulatory guide, -9 to 18°F. The licensee has not justified this deviation from range recommended between +15 to 18°F.

Table 1 of Regulatory Guide 1.23 (Reference 9) 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 or less than -2°C does nothing to the stability classification. The licensee's instrumentation includes this range. Therefore, we find that this instrumentation is acceptable to determine the atmospheric stability.

4. CONCLUSIONS

Based on our review, we find that the licensee either conforms to or is justified in deviating from Regulatory Guide 1.97, with the following exception:

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).

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, 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. Washington Public Power Supply System (WPPSS) letter, G. D. Ouchey to Director of Nuclear Reactor Regulation, NRC, "Emergency Response Capability," April 15, 1983, G02-83-346.

5. WPPSS Nuclear Project No. 2, Final Safety Analysis Report, Amendment No. 23, February 1982.

6. WPPSS letter, G. C. Sorensen to Director of Nuclear Reactor Regulation, NRC, "Nuclear Plant No. 2, Emergency Response Capability - Conformance to R. G. 1.97, Rev. 2," October 8, 1985, G02-85-710.

7. WPPSS letter, G. C. Sorensen to Director of Nuclear Reactor Regulation, NRC, "Emergency Response Capability - Conformance to R. G. 1.97, Revision 2, Clarification," January 23, 1986, G02-86-097.

8. WPPSS letter, G. C. Sorensen to Director of Nuclear Reactor Regulation, NRC, "Nuclear Plant No. 2, Operating License NPF-21, Request for Amendment to License Condition 16, Attachment 2, Item 3(b)," October 14, 1985, G02-85-724.

9. Regulatory Guide 1.23 (Safety Guide 23), Onsite Meterological Programs, NRC, February 17, 1972 or Proposed Revision 1 to Regulatory Guide 1.23, Meteorological Programs in Support of Nuclear Power Plants, NRC, Office of Standards Development, September 1980.