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HQ:S:82:115

OCT 26 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated April 9, 1982

This letter formally responds to your request for additional information contained in the reference letter.

Enclosed are responses to Questions CS 421.36 and 42; which will be incorporated into a future PSAR amendment.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

2 Enclosures

cc: Service List
Standard Distribution
Licensing Distribution

D001

Question CS421.36

Provide a more detailed discussion of the CRBR Leak Detection system and how it meets the provisions contained in the Light Water Reactor Regulatory Guide 1.45. The discussion should include detection methods, detector sensitivity, detector response time, signal correlations and calibration, seismic qualification, testability, and the provisions for technical specifications.

Response:

PSAR Section 7.5.5.1.1 has been revised to provide a more detailed discussion of the CRBRP Leak Detection Instrumentation System. A comparison to the provisions of Regulatory Guide 1.45 is contained in Section 5.3 of WARD-D-185, "Integrity of the Primary and Intermediate Heat Transport System Piping in Containment", (Reference 2 of PSAR, Section 1.6).

Technical Specifications will be developed at the FSAR stage. The Technical Specification will require that the plant will be placed in either the hot shutdown or refueling condition if there is a confirmed leak in either the primary or intermediate heat transport system.

7.5.5.1.1 Design Bases and Design Criteria For the Liquid Metal-To-Gas Leak Detection System

The design bases of the Liquid Metal-to-Gas Leak Detection System arises from the need to protect plant equipment, considerations of maintenance and plant availability, and the corrosion effects of sodium compounds on stainless steels at high temperatures.

Considering the significance of corrosion with respect to piping integrity, it is appropriate that the design criteria assure that the Liquid Metal-to-Gas Leak Detection System provide reliable detection for the Primary and Intermediate Heat Transport In-Containment Systems in a small fraction of the nominal time to penetrate the pipe by local corrosion. The effects of corrosion on the CRBRP PHTS piping have been thoroughly assessed in WARD-D-185 "Integrity of the Primary and Intermediate Heat Transport System Piping In-Containment," Reference 1.6 of the PSAR. In summary, leaks of 100 gm/hr may cause local corrosion in 3600 hrs and general corrosion in 18,000 hours at temperatures near 1000°F. At temperatures less than 700°F, the corrosion rate becomes extremely slow. The Leak Detection System will detect leaks of 100 gm/hr in pipes and components operating at temperatures greater than 700°F in less than 250 hrs.

Design Criteria have been established to guarantee reliable plant operation with pipe temperatures greater than 700°F. These include:

1. The PHTS and in-containment IHTS shall be monitored for leaks by diverse methods each capable of providing the required time response.

2. Capability shall be provided to procure a filter sample for laboratory analysis to provide a highly reliable confirmation method. Filter samples should be analyzed a minimum of once every 1000 hrs.

3. The Liquid Metal-to-Gas Leak Detection System must operate after an operating basis earthquake (OBE).

4. The leak detection system shall be equipped with provisions to readily permit testing for operability and calibration during plant operation.

5. A reliable self-monitoring provision shall be provided to detect component failure.

6. Upon loss of ability to fulfill the specified time response, the plant will be placed in a hot shutdown condition.

7. The system shall be qualified to operate in its environment.

Additional Design Criteria of the Liquid Metal-To-Gas Leak Detection System required to protect plant and capital investment, limit maintenance and protect plant availability are outlined below:

1. The Liquid Metal-To-Gas Leak Detection System shall detect and locate liquid metal-to-gas leaks throughout the plant between the temperatures of 375 to 1000°F as required to fulfill continuous monitoring requirements of Appendix G, "CRBRP Plan For Inservice and Preservice Inspections."

2. The Leak Detection System shall be able to identify the general location of the leak.

This system is not needed for initiation of plant shutdown, for removal of decay heat or for reduction of off-site radiation exposure to acceptable levels; therefore, it is classified as a non-safety system. The safety related instrumentation provided to accommodate liquid metal leaks is described in Section 7.5.3.1.1. The passive engineered safety features provided to mitigate the effects of liquid metal leaks are described in Section 3.8.

7.5.5.1.1.1 Design Description

General

Detection equipment is provided to monitor the primary and intermediate sodium coolant boundaries to identify comparatively small leaks when they occur.

The leak detection methods selected for the following installations are:

1. Particulate monitors (radiation detectors), Sodium Ionization Detectors (aerosol detectors), and chemical analysis for atmosphere monitoring in selected cells.

2. Plugging filter aerosol detectors (PFADs) for Main Heat Transfer System piping and guard vessels, major components, and for inerted cell atmosphere monitoring.

3. Contact detectors in the space between the bellows and the stem packing of the bellows sealed sodium valves.

4. Cable detectors in guard vessels and under major liquid metal components.

Of the types of leak detection devices that comprise the Leak Detection System, only sodium aerosol leak detection devices show a difference in their response when operated in an air atmosphere as opposed to an inert atmosphere. The time for a detector to respond to a leak in air is generally shorter than in an inert atmosphere. The electrical sensing types such as cable and contact detectors show no difference in response due to operating atmospheres. However, the potential for higher moisture content in air can result in greater inhibition to sodium flow when the leak is very small.

Considerations which materially affect detection times include: sodium leak rate, sodium temperature, and cell size. Test data (See Reference 5) confirm that sodium leaks of 100 gm per hour in an air or inert atmosphere can be detected by aerosol detection over the operating temperature ranges, within the detection time periods identified in Figure 5.1.1 of WARD-D-0185, "Integrity of Primary and Intermediate Heat Transport System Piping in Containment", (Reference 2, PSAR Section 1.6). Larger leaks (on the order of kg/min) will be readily detected by two or more systems in minutes.

The electrical sensing types of detectors (cables and contact) respond with an alarm when liquid metal causes an electrical short between the electrode and its protective sheath. The Sodium Ionization Detectors (SIDs) provide an alarm when the aerosol concentration reaches a level of about 10^{-11} gm/cc. The PFADs, which are integrating devices, respond with an alarm when the differential pressure across a filter has increased by 2 inches of water. The time for this response is related to aerosol concentration as shown on Figure 7.5-7. For example, at 1×10^{-11} gm/cc, the time response is approximately 250 hours. Both SIDs and PFADs have filters which are chemically examined for sodium on a monthly basis so that leaks which result in aerosol concentrations lower than 1×10^{-11} gm/cc will also be detected. A leak resulting in a concentration of approximately 2×10^{-13} gm/cc is detectable by chemical examination of the filter pads. The sodium aerosol concentration resulting from a 100-gm/h leak in inerted CRBRP cells ranging in volume from 15,000 to 115,000 ft^3 is shown on Figure 7.5-8. In the operating temperature range of 700-1000°F, the leak detection criteria are easily met with either SIDs or PFADs. In addition, during reactor operation, the radiation particulate monitoring system will detect leaks resulting in aerosol concentration of approximately 10^{-15} gm/cc in those cells containing primary sodium.

The aerosol detectors are connected to the PDH&DS so that the rate at which the signal is changing can be checked after a leak alarm is obtained. A rapid increase in PFAD differential pressure (less than 1 hour from normal reading to alarm) accompanied by leak alarms from other detectors in the same area would indicate a large leak (greater than 1 gpm). Conversely, a leak signal that took 10 to 100 hours or more to reach the alarm level would indicate a small (100-1000 gm/h) leak. The SIDs are calibrated so that aerosol concentration can be related to the signal level. Instruments are set to alarm at specific aerosol concentrations. The liquid metal-to-gas leak detection system is designed to function after an OBE. The radiation particulate monitoring system is designed to function after an SSE. All leak detection equipment will be tested periodically to demonstrate operability.

The increase in cell atmosphere temperature and pressure in the event leaks larger than 20 kg/min as detected by temperature and pressure sensors can provide an additional source of leak detection.

The ability to detect small leaks (100 gm/hr) by several methods in hours plus the ability to detect large leaks (>kg/min) in minutes will provide a highly reliable leak detection system that provides the operator information to enable shutdown to repair defects without extensive time for cleanup operations.

After a sodium or NaK leak has occurred, the Liquid Metal-to-Gas Leak Detection System equipment impacted by the leak will be either replaced or cleaned (pneumatic system rinsed with alcohol) to remove sodium leak residue products. The system will then be acceptance tested and calibrated in accordance with the preoperational test specification criteria utilized prior to inital plant startup.

Table 7.5-3 gives the primary and back-up methods of leak detection for the principal sodium systems and components in the plant. The methods shown in the table are related to the three sizes of leaks defined in Section 7.5.5.1.2. The principal methods of leak detection are described below.

Aerosol Monitoring

Aerosol monitoring will be performed by measuring the pressure drop across a membrane filter with a constant flow of gas sampled from the annular space between major piping and its insulation, from the space within guard vessels, and from cells containing liquid metal systems. Another cell aerosol monitoring method uses a sodium ionization detector. Liquid Metal aerosols or vapor are ionized by a hot filament and the ion current is measured. Increases in the ion current indicate a leak.

Based upon the experimental results, these methods provide for detection of leaks of 100 gm/hr and less, with a response time depending on temperature and the volume being monitored.

The major function of this instrumentation will be to provide indication of the presence of small leaks which do not present a significant contamination hazard, but which might result in undesirable long-term corrosion.

Contact Detectors (Spark-Plug)

Contact detectors consist of a stainless-steel-sheathed, mineral oxide-insulated, two-wire probe with the sensing end open and the wire ends exposed. Contact detectors are installed, for example, on bellows sealed valves with the sensing end between the bellows and the mechanical backup seal. A leak is detected by the reduction in circuit electrical resistance caused by sodium contacting the wire ends.

Cable Detectors

Cable detectors consist of stainless-steel-sheathed, mineral-oxide-insulated, cable with holes penetrating the sheath to permit leaked liquid metal to come in contact with the conductors. Cable detectors will be placed, for example, in the bottom of guard vessels and below large tanks.

Other Detection Methods

Pressure and temperature measurements available in the inerted cells (Section 9.5.1.5) will provide immediate indication of the presence of large leaks over the 20 kg/min size. In the case of systems containing radioactive sodium, the detection of airborne radioactivity arising from Na-24 or Na-22 in the aerosols will be performed by particulate radiation monitoring equipment (Section 11.4.2) which provides a sensitive detection method for aerosol concentrations as low as 10^{-15} gm/cc.

Chemical analysis provides positive detection capability for aerosol concentrations of approximately 10^{-13} gm/cc, depending on the leak integration period.

Other Backup Detection Method

Liquid Sodium Level Sensors in the reactor, the EVST, the IHTS expansion tank, and sodium storage tanks will provide indications of large leaks. Smoke detectors (Fire Protection System) will detect combustion products originating from sodium leaks in air (See Section 9.13.2).

Indication In Control Room

An audible group alarm is sounded in the control room upon indication of a leak or certain failures of contact, cable, or aerosol channels. The channel number producing the alarm and the location of the region covered by this channel are displayed on an annunciator on a local panel. This information will identify the leak as occurring in a specific major comonent or series of pipe sections, or specific bellow-sealed valve, or the cell containing the leaking system. The leak detection system uses the Plant Data Handling System for channel failure monitoring, data and trend logging; the sampling time interval will nominally be approximately 30 seconds.

No automatic isolation functions or reactor scram are initiated by the Liquid Metal-To-Gas Leak Detction System. Isolation or shutdown of a system showing a leak will be performed manually, following verification of the leak and review of the operating conditions.

7.5.5.1.2 Design Analysis

The Liquid Metal-to-Gas Leak Detection System will meet the appropriate requirements of CRBR Design Criterion 30, "Inspection and Surveillance of Reactor Coolant Boundary and Criterion 33, "Inspection and Surveillance of Reactor Coolant Boundary. Criterion 30 requires that means be provided for detecting and identifying the location of the source of reactor coolant leakage from the reactor coolant boundary to the extent necessary to assure that timely discovery and correction of leaks which could lead to accidents whose consequences could exceed the limits prescribed for protection of the health and safety of the public. Criterion 33 requires that means be provided for detecting intermediate coolant leakage from the intermediate coolant boundary. In order to demonstrate how the intent of the criteria will be satisfied, the instrumentation requirements met by this system for three different ranges of leaks are discussed. These ranges have been selected to analyze situations which cover the complete range of leak detection instruments. Section 15.6 discusses the consequences of leaks for the health and safety of the public.

Large Leaks

This category covers failures up to those resulting in a leak of 30 gpm or 100 kg/min. A significant physical characteristic of leaks of this size is that they would result in pressure and temperature changes in the primary cells if the leak occurs in PHTS pipe sections. This feature sets the lower boundary of the leak at about 20 kg/min; this being an estimte of the amount of sodium which would result in measurable changes in cell pressure and temperature. If the leak occurs in a guard vessel, continuity detectors will provide detection of these large leaks. Leaks of this magnitude would be detected in five minutes or less for the primary and intermediate heat transport system. The operator would then be able to initiate and complete plant shutdown within ten minutes after the start of the leak.

The pressure and temperature measurements available in the inerted cells will, in conjunction with the aerosol detectors, continuity detectors and radiation monitors, provide the response required for proper operator action in case of leaks of this magnitude.

Intermediate Leaks

Intermediate leaks were defined as those leaks which would not result in significant changes in cell pressures and temperatures but where the extent of the resulting contamination and plant maintenance makes plant shutdown desirable. The range of leak rates covered extends from the lower limit of the large leaks previously considered down to a leak of 100 gm/hr. The detection times for the wide range of leaks in this group would vary from a few minutes to several hours depending on the rate of leakage. Based upon experimental results, it is concluded that several systems would detect a leak of this magnitude in several hours at least and possibly in minutes.

Instrumentation capable of detecting leaks of this magnitude include radiation monitors, continuity detectors, and the different types of aerosol detectors.

Small Leaks

Small leaks at or below 100 gm/hr were defined as those events resulting in releases of sodium which do not pose a contamination or maintenance problem but might result in undesirable long-term corrosion (see Section 5.3.3). The methods for detecting leaks of this range are aerosol detectors and radiation monitors in the case of the primary system.

In the course of test programs, aerosol concentrations produced by leaks of down to 5 gm/hr were found to be within the detection capability of both a Sodium Ionization Detector and a Plugging Filter Aerosol Detector in test chambers. The test results show that leaks of this size can be detected in the range of one hour to 24 hours by annuli monitors depending upon the sodium temperature and gas environment. It is deduced from the test results that very small leak (<1 gm/hr) will be detected by annuli monitors in several days.

Tests during 1975 and 1976 showed that under environmental conditions typical of LMFBR operation, small leaks from typical piping configurations can be detected by both Sodium Ionization and Plugging Filter Aerosol Detectors. Continuity (cable or contact) detectors did not reliably detect small pipe leaks under these conditions. Testing in 1978 verified the performance of aerosol detectors using prototypic CRBRP cell atmosphere recirculation as well as pipe/insulation design.

It is deduced from the test results that the sodium vapor/aerosol systems will, in conjunction with existing radiation monitoring technology, provide adequate indication of the smallest sizes of leaks of interest.

Sodium Leaks Into an Air Atmosphere

Test results indicate that the methods applicable to sodium leaks in inerted cells will also operate when applied in an air atmosphere. The additional use of smoke detectors and the accessibility of piping located in an air atmosphere to visual inspection assist in the selection of an effective sodium-to-air leak detection system.

7.5.5.2 Intermediate to Primary Heat Transport System Leak Detection

7.5.5.2.1 Design Description

The IHTS pressure is maintained at least 10 psi higher than the Primary Heat Transport System at the IHX to prevent radioactive primary sodium from entering the IHTS in the event of a tube leak. Maintaining a positive pressure differential across the IHX is a limiting condition for operation of the plant (Chapter 16 - Technical Specifications). This provides assurance that a zero or negative differential will not exist during any extended interval. A loss of this pressure or a reversal of it is not expected to occur except during accident conditions. Such an occurance would necessitate an orderly plant shutdown to correct the problem. Since a reverse differential cannot occur for a significant interval. the potential leakage of primary sodium into the intermediate system, through an IHX tube leak, is small.

Leakage of primary sodium into the IHTS, should it occur. will be detected by radiation monitors provided on the IHTS piping within the SGB. The radiation monitor system will provide an indication of the radiation level and will provide alarms for conditions of excessive radiation indicative of ingress of primary sodium. Since the only activity expected in the IHTS is a low level of tritium, the radiation monitors will be very sensitive to the presence of significant amounts of radioactive primary sodium in the intermediate system. For accidents which involve a loss of IHTS boundary integrity the radiological effects have been evaluated. The results of these evaluations are presented in Sections 15.3.2.3, 15.3.3.3 and 16.6.1.5.

References to Section 7.5

1. Ford, J. A., "A Recent Evaluation of Foreign and Domestic Wastage Data from Sodium Water Reaction Investigation", APDA CTS-73-05, January, 1973.

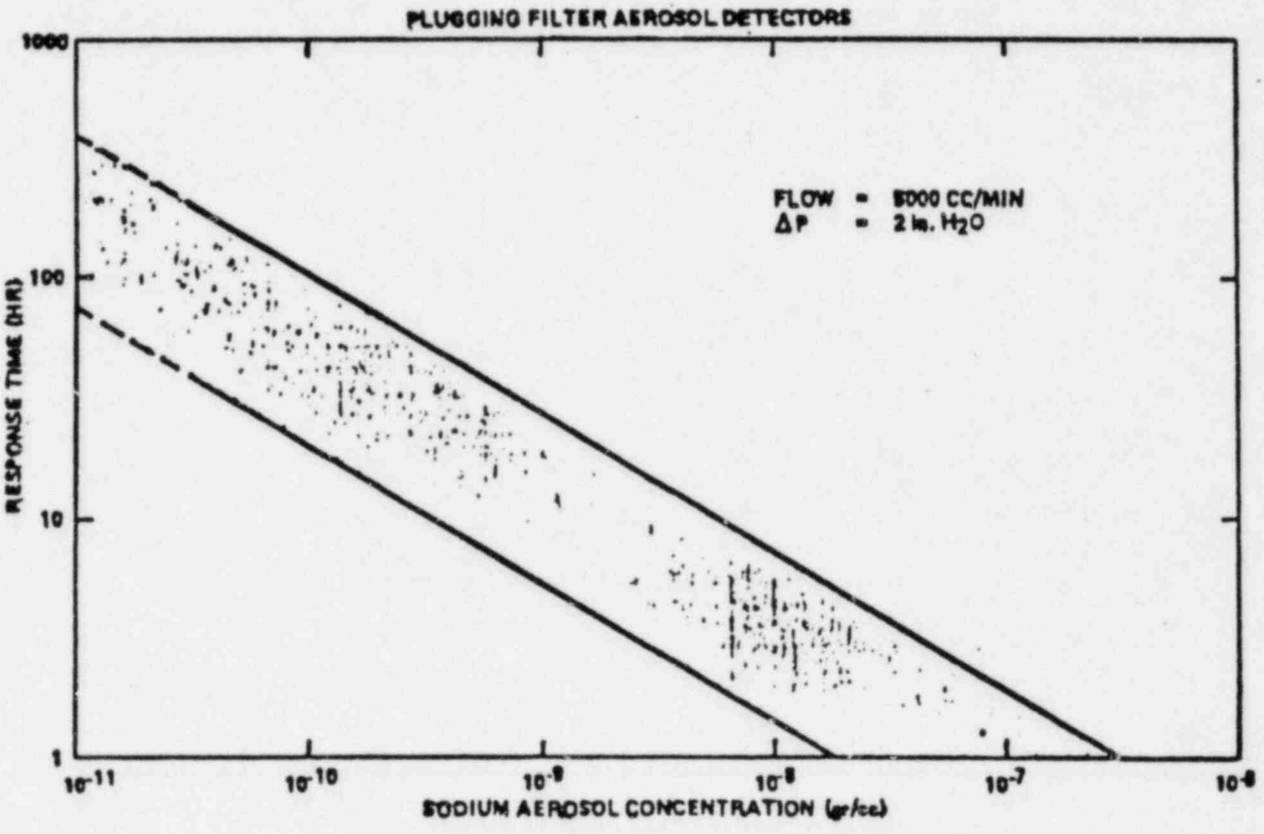
2. Morejon, J. A., "Sodium-to-Gas Leak Detection Mockup Tests", N707-TR-520-004, September 17, 1975. (Atomics International)

3. Greene, D. A., J. A. Gudahl and J. C. Hunsicker, "Experimental Investigation of Steam Generator Materials by Sodium-Water Reactions, Volume 1, GEAP-14094, January 1976.

4. Gudahl, J. A. and P. M. Magee, "Microleak Wastage Test Results", GEFR-00352, March 1978.

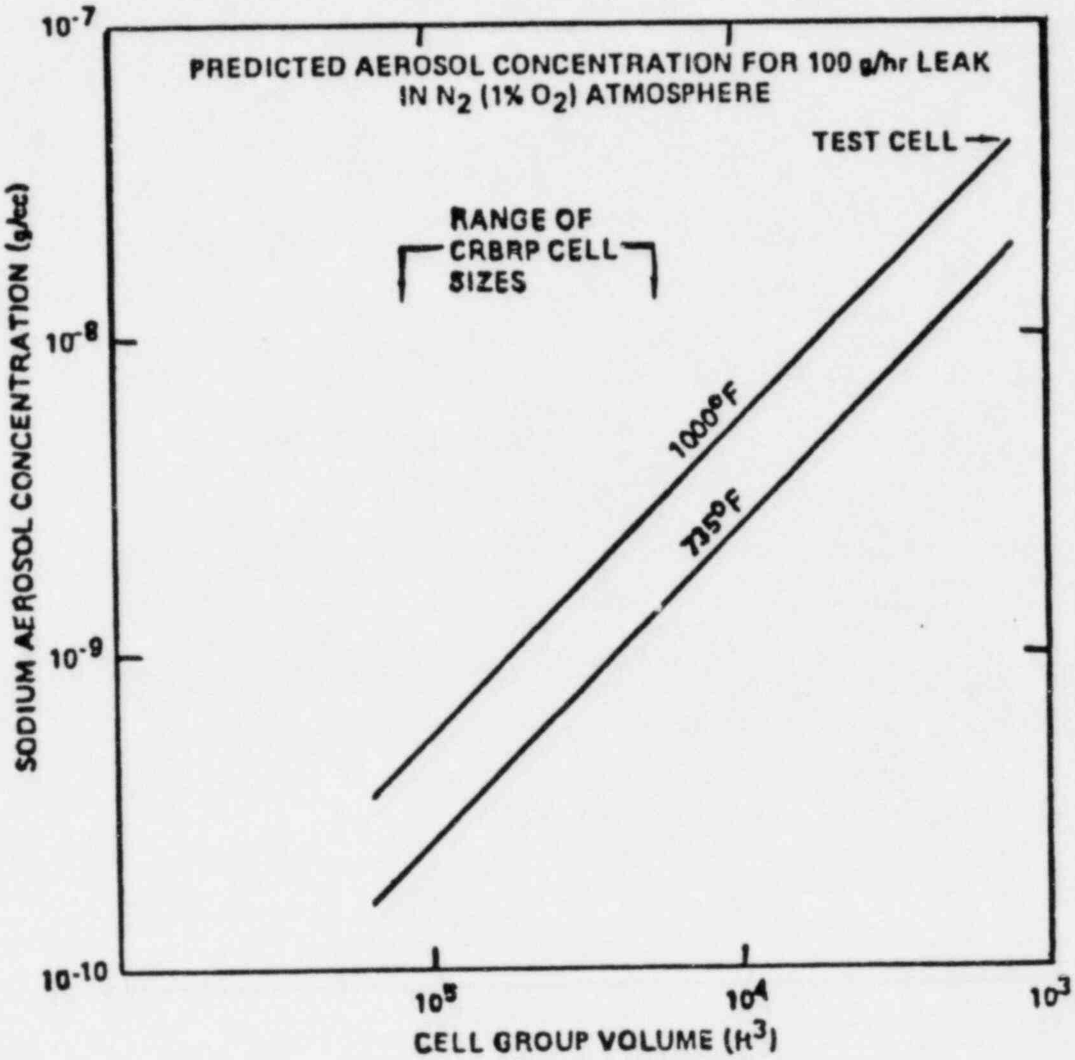
5. Matlin, E., Witherspoon, J. E., Johnson, J. L., "Liquid Metal-to-Gas Leak Detection Instruments".

Figure 7.5-7 Liquid Metal/Gas Leak Detection System
Response Time Vs. Sodium Aerosol Concentration (Inerted Cells)



Amend. 72
Oct. 1982

Figure 7.5-8 Liquid Metal/Gas Leak Detection System
Predicted Aerosol Concentration for 100 g/HR
Leak in N_2(1% O_2) Atmosphere



Amend. 72
Oct. 1982

Question 421.42

Section 7.1.2 and 7.2.2 of Chapter 7 of the PSAR reference the use of IEEE standards. Other sections in Chapter 7 make reference to Section 7.1.2 but do not identify specific IEEE standards which were implemented in the system design. Justify why Section 7.3 through 7.7 of the PSAR do not provide enought information to determine whether the IEEE standards are implemented in the design.

Response:

Chapter 7 has been revised to add specific identification of IEEE standards when appropriate as described below. Compliance with IEEE standards for non-safety related systems is not required and therefore use of IEEE standards for those systems is not discussed.

Section 7.2 - This section is amended to clarify the use of IEEE standards.

Section 7.3 - This section is amended to clarify the use of IEEE standards.

Section 7.4 - This section is amended to clarify the use of IEEE standards.

Section 7.5.1 - The Wide Range and Power Range Flux Monitors discussed in this section are safety related, the IEEE standards of Table 7.1-3 are applied to the designs.

Section 7.5.2 - Addresses the types of functions and the sensors used in the plant and does not specifically identify these instruments as safety related or not. Table 7.5-1 identifies the variables which are safety related as does Section 7.2. Paragraph 7.5.2.2 states that the instruments which are a part of the Protection system comply with the requirements of Section 7.1.2 and 7.2.2 which encompasses the IEEE standards listed in Table 7.1-3.

Section 7.5.3 - The sodium level probes discussed in this section are 1E. The remaining instrumentation is non-1E. Section 7.5.3.2 states that the sodium level probes are part of the Reactor Shutdown system and will comply with PPS Design Requirements (Sections 7.1.2 and 7.2.2). The probes, therefore, will comply with IEEE standards identified in these sections as applicable to PPS.

Section 7.5.4 - The Failed Fuel System is not safety related.

Section 7.5.5 - The leak detection systems discussed in this section are not safety related.

Section 7.5.6 - SWRPRS instrumentation and control has two functions. One is to initiate a reactor trip, the other is to isolate the affected loop. The reactor trip function is part of the Plant Protection system and as stated in 7.5.6.2 complies with Sections 7.1.2 and 7.2.2. Isolation of the affected loop is not safety related since it does not compromise the ability to remove decay heat from the unaffected loops.

Sections 7.5.7, 7.5.8 and 7.5.9 - The instruments discussed in these sections are safety related, the IEEE standards of Table 7.1-3 are applied to the designs.

Sections 7.6.1, 7.6.2, 7.6.4 and 7.6.6 - These Sections have been revised to incorporate applicable IEEE Standards.

Section 7.6.5 - The SGB Flooding Protection System is safety related and section 7.6.5 is amended to clarify the use of IEEE standards.

Sections 7.7 and 7.8 - No IEEE standards are applied in these sections since the systems described therein are non safety related systems.

Section 7.9 - This section has been amended to clarify the use of IEEE standards.

TABLE 7.1-3

LIST OF IEEE STANDARDS APPLICABLE TO
SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE-323-1974	Qualifying Class IE Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Foreword of IEEE 323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1977	Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems
IEEE-344-1975	IEEE Std. 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1975	General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE-383-1974	Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Station.
IEEE-384-1974	IEEE Trial Use Standard Criteria for Separation of Class IE Equipment and Circuits
IEEE-420-1973	Trial-Use Guide for Class IE Control Switchboards for Nuclear Power Generating Stations
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

- Environmental Changes

All electrical equipment is subject to performance degradation due to major changes in the operating environment. Where practical, PPS equipment is designed to minimize the effects of environmental changes; if not, the performance at the environmental extremes is used in the analysis.

Measures have been taken to assure that the RSS electronics are capable of performing according to their essential performance requirements under variations of temperature. The range of temperature environment specified for all the electronic equipment considered here is greater than is expected to occur during normal or abnormal conditions. Electronics do not fail catastrophically when these limits are exceeded even though this is the assumed failure mode. The detailed design of the circuit boards, board mounting and racks includes free ventilation to minimize hot spots. Ventilation is a result of natural convection air flow.

The RSS is designed to operate under or be protected from a wider range of relative humidity than that produced by normal or postulated accident conditions.

Vibration and shock are potential causes of failure in electronic components. Design measures, including the prudent location of equipment, minimize the vibration and shock experienced by RSS electronics. The equipment is qualified to shock and vibration specifications which exceed all normal and off-normal occurrences.

The RSS comparators and protective logic are designed to operate over a power source voltage range of 108 to 132 VAC and a power source frequency range of 57 to 63 HZ. The maximum variation of the source voltage is expected to be $\pm 10\%$. More extreme variations in the power source may result in the affected channel comparator or logic train outputting a trip signal. In addition, testing and monitoring of RSS equipment is used, where appropriate, to warn of impending equipment degradation. Therefore, it is not expected that changes in the environment will cause total failure of an instrument channel or logic train, much less the simultaneous failure of all instrument channels or logic trains.

The majority of the RSS electronics is located in the control building, and is not subjected to a radioactive environment. Any PPS equipment located in the radioactive areas (such as the head access area) will be designed to withstand the level of activity to which it will be subjected, if its function is required.

o Tornado

The RSS is protected from the effects of the design basis tornado by locating the equipment within tornado hardened structures.

o Local Fires

All RSS equipment, including sensors, actuators, signal conditioning equipment, wiring, scram breakers, and cabinets housing this equipment is redundant and separated. These characteristics make any credible fire of no consequence to the safety of the plant. The separation of the redundant components increases the time required for fire to cause extensive damage and also allows time for the fire to be brought to the attention of the operator such that corrective action may be initiated. Fire protection systems are also provided as discussed in Section 9.13.

o Local Explosions and Missiles

All RSS equipment essential for reactor trip is redundant. Physical separation (distance or mechanical barriers) and electrical isolation exists between redundant components. This physical separation of redundant components minimized the possibility of a local explosion or missile damaging more than one redundant component. The remaining redundant components are still capable of performing the required protective functions.

o Earthquakes

All RSS equipment, including sensors, actuators, signal conditioning equipment, wiring, scram breakers and structures (e.g., cabinets) housing such equipment, is classed as Seismic Category I. As such, all RSS equipment is designed to remain functional under OBE and SSE conditions. The characteristics of the OBE and SSE used for the evaluation of the RSS are found in Section 3.7.

7.2.2 Analysis

The Reactor Shutdown System meets the safety related channel performance and reliability requirements of the NRC General Design Criteria, IEEE Standard 279-1971, applicable NRC Regulatory Guides and other appropriate criteria and standards.

The RSS Logic is designed to conform to the IEEE Standards listed in Table 7.2-4.

General Functional Requirement

The Plant Protection System is designed to automatically initiate appropriate protective action to prevent unacceptable plant or component damage or the release or spread of radioactive materials.

TABLE 7.2-4

LIST OF IEEE STANDARDS APPLICABLE TO
THE REACTOR SHUTDOWN SYSTEM LOGIC (1)

IEEE 279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE 308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE 317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE 323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE 323-A-1975	Supplement to the Foreward of IEEE 323-1974
IEEE 336-1971	IEEE Standard: Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE 338-1977	IEEE Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE 352-1975	IEEE Guide for General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE 379-1972	IEEE Trial-Use Guide for the Applicaton of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE 384-1974	IEEE Trial Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE 494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

(1) IEEE Standards applicable to the instrumentation and monitoring systems are listed in Section 7.5.

Head Access Area Radiation

The Head Access Area Radiation Subsystem initiates closure of the containment isolation valves in the event of large radiation releases in the head access area. Three radiation sensors are located in the head access area to provide early initiation and closure of the isolation valves to assure that releases from design basis events do not exceed the guideline values of 10CFR100.

7.3.1.2.2 Essential Performance Requirements

To implement the required isolation function within the specified limits, the CIS must meet the functional requirements specified below:

The closure time requirement for the inlet and exhaust isolation valves is 4 seconds with a three second or less detection time in the heating and ventilating system. A 10 second transport time from sensing point to the valve exists (see Section 15.1.1). The 3 seconds includes sensor time response, comparator and logic time delays.

The CIS is designed to meet these requirements for the environmental conditions described in Section 7.2.1.

7.3.2 Analysis

The design of the CIS provides the necessary design features to meet the functional and performance requirements as described below. The CIS logic is designed to conform to the IEEE Standards listed in Table 7.3-2.

7.3.2.1 Functional Performance

The analyses in Sections 15.5 and 15.6 shows the results of the postulated fault conditions. These analyses assumed a closed containment where the events occurred with the containment hatch closed. For the limiting event, primary drain tank fire during maintenance, scoping analyses have been performed to determine the required closure time of the containment isolation valves. For the primary drain tank fire, closure within 20 minutes is adequate. Further, analyses to determine the required closure time under postulated accident conditions have been performed and are discussed in Section 15.1.1. These analyses are used to determine the available design margin. The results of this assumed condition do not exceed the guideline values of 10CFR100 if the main exhaust and inlet valves are closed within 4 seconds assuming the normal air transport time from the detector to the valve is 10 seconds or more, a 14,000 Cfm normal ventilation rate.

Since the automatic Containment Isolation System is designed to isolate within the above time response requirements, all of the design basis conditions are terminated within the necessary limits for the present design concept.

7.3.2.2 Design Features

The CIS instrumentation, controls and actuators are designed to meet the requirements of IEEE-279-1971. The analyses of compliance with these are summarized below.

TABLE 7.3-2

LIST OF IEEE STANDARDS APPLICABLE TO
THE CONTAINMENT ISOLATION SYSTEM LOGIC

IEEE 279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE 308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE 317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE 323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE 323-A-1975	Supplement to the Foreward of IEEE 323-1974
IEEE 336-1971	IEEE Standard: Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE 338-1977	IEEE Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE 344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE 352-1975	IEEE Guide for General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE 379-1972	IEEE Trial-Use Guide for the Applicaton of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE 384-1974	IEEE Trial Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE 494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

7.4 INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFE SHUTDOWN

The Instrumentation and Control Systems necessary for safe shutdown are those associated with monitoring of core criticality, decay heat removal (SGAHRS portion), outlet steam isolation, and control room habitability.

Monitoring of core criticality is effected by the Flux Monitoring System (Section 7.5.1). The control room habitability is covered in Chapter 6. Thus, this section treats the control and instrumentation needs for decay heat removal by the Steam Generator Auxiliary Heat Removal System (SGAHRS) and outlet steam isolation by the Outlet Steam Isolation System (OSIS); control and instrumentation for Direct Heat Removal Service (DHRS) is discussed in Section 7.6.

7.4.1 Steam Generator Auxiliary Heat Removal Instrumentation and Control System

7.4.1.1 Design Description

7.4.1.1.1 Function

The SGAHRS (fluid system and mechanical components as described in Section 5.6.1, and electrical components as described below) provides the heat removal path and heat sink for the nuclear steam supply system following upset, emergency, or faulted events which render the normal heat sink unavailable.

The SGAHRS Instrumentation and Control System in conjunction with the PPS detects the need for, initiates, and controls the alternate heat removal path when the normal heat sink is unavailable. The SGAHRS nominal control setpoints shown in Table 7.4-2 are discussed in the following subsections.

The SGAHRS Instrumentation and Control System is designed to the IEEE Standards listed in Table 7.4-3.

7.4.1.1.2 Equipment Design

The mechanical system for which the SGAHRS I&C is provided is briefly described below.

When actuated, the SGAHRS draws water from a Protected Water Storage Tank and pumps it to each steam drum. Two supply lines are provided for each steam drum. One line is supplied by two half-sized, motor-driven feedwater pumps while the other is supplied by a full-sized, turbine-driven pump. Each supply line provides a flow control valve and an isolation valve at the inlet to each steam drum. The isolation valves are provided to isolate the auxiliary feedwater system from the steam generator system during power operation and to provide leak isolation during SGAHRS operation.

In addition, a Protected Air Cooled Condenser (PACC) supplied with each steam drum is placed into operation. This system rejects heat to the atmosphere via convection. Saturated steam is supplied to the condenser from the steam drum

7.4.1.2 Design Analysis

To provide a high degree of assurance that the SGAHRS will operate when necessary, and in time to provide adequate decay heat removal, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to SGAHRS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the SGAHRS Instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment where practical and will, in the event of a failure, assume a failed position consistent with its intended safety function.

Because there are three redundant decay heat removal loops, the instrumentation and controls associated with each individual loop (e.g., auxiliary feedwater flow and air cooled condenser control systems) do not independently meet single failure criteria. However, when taken collectively as a system, they provide the single failure capability required.

7.4.2 Outlet Steam Isolation Instrumentation and Control System

7.4.2.1 Design Description

7.4.2.1.1 Function

The Outlet Steam Isolation Subsystem (OSIS) provides isolation of steam system pipe breaks. Steam system isolation is a necessary function for safe shutdown in those pipe break conditions affecting the three steam supply systems and is provided if needed on a per loop basis. By definition, this zone of protection will include the high pressure steam supply system downstream from the individual loop check valves.

The OSIS Controls are designed to the IEEE Standards listed in Table 7.4-3.

TABLE 7.4-3

LIST OF IEEE STANDARDS APPLICABLE TO
SGAHRS AND OSIS INSTRUMENTATION AND CONTROL SYSTEMS

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Foreword of IEEE 323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1977	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1975	General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEE-382-1980	IEEE Standard for Qualification of Safety-Related Valve Actuators
IEEE-384-1974	IEEE Trial Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

7.4.4.2 Design Analysis

The Remote Shutdown System provides the RSMP from which an operator can assess the progress of the plant shutdown and command the local operation of the plant systems (primarily SGAHRS) to effect the shutdown. It should be noted that the PACC subsystem of SGAHRS is automatically initiated by all reactor trips, and it remains in operation for the duration of the plant shutdown or as long as the reactor generates significant decay heat.

The Remote Shutdown System imposes no special requirements on the plant systems, but takes advantage of the following system design features:

- o The ability to operate in both local and remote modes with isolation from and annunciation in the Control Room when operating in the local mode.
- o The redundancy diversity, separation, isolation and reliability of the safety grade systems.
- o The design and location of safety grade systems equipment that minimize the probability and effect of fires and explosions on the ability of the systems to perform their safety function.
- o The redundant safety grade SGAHRS provides the capability to achieve and maintain hot shutdown and, if desired, to cool the plant to and maintain the plant at refueling conditions.
- o When transferring SGAHRS to the local mode, the operator manually starts SGAHRS. Once started, SGAHRS automatically controls those parameters used to remove decay heat.

The RSMP is a non-Class 1E Seismic Class III assembly and therefore, is not subject to the separation requirements of IEEE 384-1974, or to the seismic qualification requirements of IEEE 344-1974, or to any of the other IEEE Standards listed in Table 7.1-3.

7.4.4.1.4 Equipment Design

The RSMP is the only piece of equipment provided by the Remote Shutdown System. It will be a vertical sided, non-Class 1E cabinet assembly containing meters and a phone jack panel. The meters will receive buffered signals from the initiating systems and, thus, do not require transfer switches to isolate them from the Control Room. The phone jack panel will permit the operator at the RSMP to communicate with the five NSSS or Nuclear Island buildings by means of any of the three MCJ circuits provided in each of the buildings. In addition, communications among the buildings can be established through the phone jack panel on the RSMP.

The indications provided on the RSMP are as follows:

- For each primary heat transport system loop,

1 - Pump outlet sodium temperature indicaton	(3 total)
1 - Reactor inlet sodium temperature indication	(3 total)
1 - Sodium pump shaft speed indication	(3 total)

- For each intermediate heat transport system loop,

1 - IHX outlet sodium temperature indication	(3 total)
1 - IHX inlet sodium temperature indication	(3 total)
1 - Sodium pump shaft speed indication	(3 total)

- For each superheated steam loop,

1 - Temperature indication	(3 total)
1 - Steam flow indication	(3 total)

- One reactor vessel sodium level meter (long probe)

- For each Diesel Generator (3 total)

  1 - Wattmeter
  1 - Frequency meter
  1 - Varmeter
  1 - Voltmeter with phase selector switch
  1 - Ammeter with phase selector switch

In addition to the foregoing indications, other indications used during remote shutdown operations that are not on the RSMP will be available as follows:

- SGAHRS

  Controls and indicators used for the operation of each SGAHRS division are located on the three seperate SGAHRS panels in cells 272A, B, and C. Each SGAHRS division is separate and redundant from the other divisions. See the response to Question CS421.04 for additional information about SGAHRS division assignments.

E. Operating Basin Overflow

F. EPSW Makeup Pump Discharge Pressure

G. Emergency Cooling Tower Basin Level

H. EPSW Flow to Emergency Chillers

I. EPSW Temperature at the Discharge of Emergency Chillers

J. EPSW Flow from Diesel Generators Heat Exchangers

K. EPSW Temperature at the Discharge of Diesel Generators Heat Exchangers

L. Diff. Pressure Across Emergency Chillers

M. Transfer of Controlling Capabilities from Control Room to Local Panels

N. Pump and Fan Status

Process variables identified above with 'A' and 'H' are designated as accident monitoring variables to assess plant and environs conditions during and following an accident. Refer to Section 7.5.11 of PSAR for detailed requirements on Accident Monitoring.

7.6.1.3.3 Inputs to PDH&DS

The following process variables are provided as inputs to Plant Data Handling & Display System (Non-Safety System):

A. EPSW Discharge Temperature

B. Emergency Cooling Tower Basin Level

C. EPSW Temperature at the Discharge of Emergency Chiller

D. EPSW Temperature at the Discharge of Diesel Generator Heat Exchanger

E. EPSW Flow to Emergency Chiller

Inoperable status of EPSW Pumps; Makeup Pumps; and Cooling Tower Fans is also monitored through Inoperable Status Monitoring System.

7.6.1.1.3.4 Design Analysis

EPSW System is designed to operate automatically. The system is operated only during emergency conditions. EPSW System components are cascaded to operate in sequence. Starting of EPSW Pumps will operate EPSW Makeup Pumps and Cooling Tower Fans. System will not operate when the EPSW Pump pit level is low or when electrical fault exists.

The design of the EPSW System is in conformance with the following IEEE standards listed in Table 7.6-2.

7.6.2.2.3 Inputs to PDH&DS

The following process variables are provided as inputs to Plant Data Handling & Display System (Non-Safety System):

A. ECW Temperature at the Inlet of Emergency Chiller

B. ECW Temperature at the Discharge of Emergency Chiller

C. ECW Flow from Emergency Chiller

D. ECW Chiller Trip Status

E. ECW Containment Isolation Valves Status

F. Secondary Coolant Expansion Tank DT-J Leakage

7.6.2.2.4 Design Analysis

ECW System is designed to operate automatically. The system is operated only during emergency condition. ECW System components are cascaded to operate in sequence. Low flow of NCW to ECW loop signal will align ECW isolation valves and operate ECW Pumps, Emergency Plant Service Water Loops, and ECW Chillers. System will not operate when the EPSW flow through chiller is not established or when electrical fault exists.

The design of the ECW System is in conformance with the IEEE Standards listed in Table 7.6-2.

3. Unit cooler or HVAC unit supply air temperature high or air temperature entering cooling coil low.

4. Smoke, ammonia, chlorine, fluorine or radiation present in Control Room main or remote air intake.

5. Control switch in the local mode (Control Room only alarm only).

C. Typically, process variables are provided as inputs to the Plant Data Handling & Display System as follows:

1. Control Room and computer room humidity.

2. Containment differential pressure.

3. Annulus differential pressure.

4. RSB confinement differential pressure (four different cells).

5. Control Room differential pressure.

6. Air temperature entering and leaving each filter unit.

7. Air temperature entering and leaving each HVAC unit.

8. Cell temperature of each area being serviced by a unit cooler or HVAC unit.

9. Inoperable or bypass status of components.

D. The following process variables are classified as Accident Monitoring variables and are used to assess plant and environs conditions during and following an accident:

1. Annulus to atmosphere differential pressure.

2. RSB confinement to atmosphere differential pressure.

3. HVAC units discharge air temperature.

4. Filter units adsorbent filter leaving air temperature.

5. HVAC and filter units air flow low.

6. Damper and valve position indication.

7. Fan operation status indication.

7.6.4.3 Design Analysis

The HVAC Instrumentation and Control System is designed to perform the functions described in Section 7.6.4 while meeting the criteria listed in Section 7.6.4.1. All HVAC I&C circuits shall meet the requirements of Section 7.1 with the exception of alarm circuits and inputs to the PDH&DS which are

Non-Class 1E circuits. The design of the HVAC Instrumentation and Control system is in conformance with the IEEE Standards and listed in Table 7.6-2.

Refer to PSAR Section 7.1.2 for conformance to applicable IEEE Standards.

7.6.5 Steam Generator Building (SGB) Flooding Protection Subsystem

7.6.5.1 Design Basis

The SGB Flooding Protection Subsystem is provided to prevent flooding of SGAHRS equipment resulting from postulated SGS water/steam line ruptures, thereby assuring the availability of SGAHRS for reactor decay heat removal following water/steam line rupture events.

The SGB Flooding Protection Subsystem is designed to the IEEE Standards listed in Table 7.6-3.

7.6.5.2 Design Requirements

The SGB Flooding Protection Subsystem is designed to perform the following functions:

a) Detect the presence of large steam/water piping ruptures (see Section 15.3.3.1) by temperature and moisture sensors in each cell.

b) Detect water level flooding conditions in each cell by water level sensing elements.

c) Provide the signals to initiate the alarms and activate the equipment which provides the SGB flooding protection.

7.6.5.3 Design Description

7.6.5.3.1 Instrumentation

Instrumentation provided for this subsystem consists of Class 1E temperature, and moisture transducers. In addition, non-Class 1E level transducers are provided. The transducers and associated control logic are located in the SGB cells containing main feedwater or recirculation piping. Three independent moisture and temperature measurements in each cell are utilized for identifying a major water/steam line rupture. Water level measurements in each cell confirm a flooding condition and are annunciated in the main control room.

7.6.5.3.2 Controls

Each heat removal loop isolates the main feedwater supply upon detection of a major pipe rupture. The start-up and main feedwater control valves close upon activation by a two-out-of-three logic using measurements of moisture and temperature in each cell. The main feedwater isolation valve is independently closed upon activation by a two-out-of-three logic using the same three moisture and temperature measurements from each cell. Separation and isolation is maintained between the control valve and isolation valve activation logic.

Small water/steam leaks are identified in each SGB cell by measuring water level. Manual corrective control of flooding is initiated by the operator upon annunciation in the main control room.

Inoperable status of subsystem fans (MA, MB, EA, EB) and isolation valves (two per subsystem) is also monitored through Inoperable Status Monitoring System.

7.6.6.2.4 Design Analysis

Refer to PSAR Section 7.1.2 for conformance to applicable IEEE Standards. RGC system is designed to operate automatically. The system and its safety-related subsystems are operated during normal as well as emergency conditions. The RGC system components are cascaded to operate in sequence. Starting a fan will open associated supply and return gas isolation valves. A subsystem will not operate when high water vapor or cooler high water level or electrical fault exists.

As discussed in Section 9.16 each subsystem of RGCs supplies cooling to redundant components, so no additional redundancy is provided in its components and instrumentation.

The systems are designed for fail safe operation and control equipment will assume a failed position consistent with its intended safety function.

The coolant supply to safety-related subsystems MA, MB, EA, EB is provided by Emergency Chilled Water System. The fan motors for these subsystems are provided with AC power from Class 1E power sources to continue operating during loss of offsite power, except for the booster fan of the subsystem EB which is not required to operate during loss of power condition. Subsystems MA and EA and the EM pumps cooled by these two subsystems are served by Class 1E power supply Division 1. Also, subsystems MA and EA are served by Emergency Chilled Water Loop "A". Subsystems MB and EB are served by Emergency Chilled Water Loop "B", and Class 1E power supply Division 2. The EM pumps cooled by subsystems MB and EB are also connected to Class 1E power supply Division 2. Automatic isolation valves are designed as fail open valves so as to be in their safety position upon loss of power.

Fan and Isolation Valve control switches are located in the local panels as well as in the back panels for subsystems MA, MB, EA and EB, except for booster fan. Thus, in case of control room evacuation the fans and valves can be controlled from outside the control rooms, using local panels.

The design of the Recirculating Gas System is in conformance with the IEEE Standards listed in Table 7.6-2.

TABLE 7.6-2

LIST OF IEEE STANDARDS APPLICABLE TO EMERGENCY PLANT SERVICE WATER, EMERGENCY CHILLED WATER, HVAC, AND RECIRCULATING GAS INSTRUMENTATION AND CONTROL SYSTEM

a) IEEE Standard 279-1971

IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations

b) IEEE Standard 308-1974

Criteria for Class 1E Power Systems for Nuclear Power Generating Stations

c) IEEE Standard 323-1974

Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations

d) IEEE Standard 338-1977

Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems

e) IEEE Standard 379-1972

IEEE Trial-Use Guide for the Applicability of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems

f) IEEE Standard 383-1974

Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations

g) IEEE Standard 384-1974

IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits.

TABLE 7.6-3

LIST OF IEEE STANDARDS APPLICABLE TO
SGB FLOODING PROTECTION SUBSYSTEM

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Foreword of IEEE-323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1971	IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1972	IEEE Trial-Use Guide: General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single-Failure Criteric to Nuclear Power Generating Station Protection Systems
IEEE-384-1974	IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

Since the Main Control Panel includes safety related equipment, the sections including this equipment are designed to Seismic Category I and qualified in accordance with IEEE Std. 323 and IEEE Std. 344. Structures, wiring, wireways, and connectors are designed and installed to ensure that safety related equipment on the control panel remains operational during and after the SSE. The Main Control Panel is constructed of heavy gauge steel within appropriate supports to provide the requisite stiffness.

Within the boundaries of the Main Control Panel Sections, modules are arranged according to control functions. Fire retardant wire is used. Modular train wiring is formed into wire bundles and carried to metal wire ways (gutters). Gutters are run into metal vertical wireways (risers). The risers are the interface between external wire trays feeding the panel and Main Control Panel wiring. Risers are arranged to maintain the separated routing of the external wire trays. (See Figures 7.9-3 and 7.9-4).

Mutually redundant safety train wiring is routed so as to maintain separation in accordance with the criteria of IEEE Std. 384. A minimum of six inches air separation is maintained between wires associated with different trains. Where such air separation is not available, mechanical barriers are provided in lieu of air space.

The Main Control Panel protection system circuits are designed and selected to ensure that system performance requirements are met and channel integrity and independence are maintained as required by IEEE Std. 279. Power division separation and isolation are maintained in accordance with the requirements of IEEE Std. 308.

7.9.3 Local Control Stations

Local control panels are provided for systems and components which do not require full time operator attendance and are not used on a continuous basis. In these cases, however, appropriate alarms are activated in the Control Room to alert the operator of an equipment malfunction or approach to an off-normal condition.

7.9.4 Communications

Communications are provided between the Control Room and all operating or manned areas of the plant. In addition to public address and interplant communications and the private automatic exchange (used for in-plant and external communications) a sound powered maintenance communication jacking system is provided. Redundant and separate methods of communication between the control room and other TVA generating plants is also provided.

7.9.5 Design Evaluation

Following the Three Mile Island accident, a large task force was formed for the purpose of performing a thorough review of the CRBRP Control Room design. This overall review was divided into three parts; a planning phase, a review phase, and assessment and implementation phase. Following the task force effort, NUREG-0700 was issued. NUREG-0700 is similar in intent to the CRBRP Control Room design evaluation.

7.9.5.1 Planning Phase

In the planning phase the objectives and scope of the task force were identified, and criteria were established for personnel selection. A charter was developed which contained the scope and objectives, and personnel selection was accomplished.

The task force charter required a review of the Control Room design and the operating procedure outlines to ensure that the systems designs, the integration of the systems, and the man-machine interfaces properly supported safe operations of the plant during both normal and abnormal conditions. A task analysis was established for observing the operator conducting various duties. Specific items included in the review are:

1. Overall Control Room and individual panel designs and features, and their interface with the operator.

2. System and overall plant operating procedure outlines.

3. Administrative approaches for plant operations.

4. Recommendations from other Key System Review Task Forces.*

5. Recommendations made by NRC and other parties as a result of the Three Mile Island occurrence.

6. Computer utilization by the operators.

7. Operator training requirements.

8. Remote shutdown capabilities and safety system status indication in the Control Room.

Criteria were established for personnel selection of those to participate on the task force. Nuclear experience was considered necessary in the areas of design, analysis, operations, testing, maintenance, and training. Personnel whose background included sodium plants and light water plants were selected. Licensed and qualified operators were considered mandatory. Personnel with human factors education and experience both inside and outside the nuclear industry were included.

Human factors considerations were emphasized in the planning phase. Previous Control Room design efforts had attempted to optimize the man-machine interface. However, a major objective of the Control Room Task Force was to re-evaluate this interface. Prior to the evaluation effort a seminar was held, under the direction of three leading human factors personnel, to teach the Task Force disciplined methods for considering human factors. Based on this training and further assistance from human engineers, check lists were prepared to evaluate the man-machine interface.

*See Reference 7.9-1

7.9.5.2 Review Phase

In the review phase extensive analysis of plant events were conducted. Functional analyses were made of the operator in his response to automatic equipment actions, manual actions which had to be performed in the Control Room, and manual actions required by operators external to the Control Room. More than 200 walk-throughs of plant events were conducted.

The Control Room design and operating instructions were thoroughly reviewed in four areas:

1. Proper identification of systems to be operated from the Main Control Room.

2. Proper staffing of the Control Room.

3. Proper overall layout of the Control Room to enhance the man-machine interfaces and support the integrated operation of plant's systems.

4. Proper layout and design of individual Control Room panels, instruments, indicators, and controls to enhance the man-machine interface and support the integrated operations of the plant's systems.

A full scale mockup of the Control Room was used. The events chosen to be evaluated were carefully selected so they would umbrella all of the operations that are either expected to occur or might be postulated to occur over the life of CRBRP. The off-normal events include plant responses to single and multiple failures.

The methodology of performing this review consisted of using three groups of people; simulators, operators, and evaluators.

The Simulators analyzed the events which were to be evaluated prior to the walk-throughs and then, during the walk-through evaluations, simulated the control panel indicators. Some of these events had previously been analyzed via computer while other events required additional computer runs to enable mocking up the panel as it would appear to the operator. The control panels were mocked up by the Simulators to represent the changing plant conditions and the information flow into the Control Room during the event. This made the walk-through as realistic as possible.

The Operators played the part of the Control Room operators and carried out the steps of the procedure being evaluated. They touched each switch they were required to operate, and observed each indicator which was part of the particular event.

The Evaluators included a Human Factors Engineer and a Systems Engineer. Their function was to fill out the Operating Sequence Diagram and the evaluation sheets for each procedure and event reviewed.

As problems or concerns were encountered, recommendations were made. These were, in some cases, of a broad nature and reflected the need for reconsideration of decisions made in the four most important evaluation areas described above. Other problems and concerns related to specific details of the Control Room design or the procedure outlines.

7.9.5.3 Assessment and Implementation Phase

The evaluation and implementation of the recommendations started with a check of the consistency of all of the recommendations by the task force. Small models of the overall Control Room and Main Control Panel were made assuming all recommendations were incorporated into the design. The recommendations were modified based on the small model to provide a coordinated and consistent set of final recommendations. Senior Project Management reviewed the final set of recommendations and issued them to the Project line organization for assessment and implementation. The cognizant design engineers have two choices. They can either accept the recommendation if it is valid, and include it into the plant design via normal procedures, or reject the recommendation and provide adequate justification if the recommendation is invalid. Each assessment is reviewed and approved by senior project management.

7.9.5.4 Conclusions

The Control Room Task Force Design Review is documented in further detail in Reference 7.9-1. In September 1981, NUREG-0700 entitled "Guidelines for Control Room Design Review" was issued. A comparison between these two documents leads to the conclusion that although NUREG-0700 was issued after the Control Room Task Force Review, the intent of the NRC in promulgating NUREG-0700 is similar to the Project's intent in performing the Control Room Task Force Review, and the intent of NUREG-0700 is believed met by CRBRP.