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MFN 08-742, Supplement 2

Docket No. 52-010

December 18, 2008

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Document Control Desk
Rockville, MD 20852

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 251 Related to ESBWR Design Certification Application - Instrumentation & Control Systems - RAI Numbers 7.7-7 and 7.7-8**

Enclosures 1 and 2 contain the GE Hitachi Nuclear Energy (GEH) response to RAI Numbers 7.7-7 and 7.7-8 from the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated September 4, 2008.

If you have any questions or require additional information, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 08-687, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request For Additional Information Letter No. 251 Related To ESBWR Design Certification Application*, dated September 4, 2008

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 251 Related to ESBWR Design Certification Application - Instrumentation & Control Systems - RAI Numbers 7.7-7 and 7.7-8

2. Response to Portion of NRC Request for Additional Information Letter No. 251 Related to ESBWR Design Certification Application - DCD Markups for RAI Numbers 7.7-7 and 7.7-8

cc:

AE Cubbage	USNRC (with enclosures)
RE Brown	GEH/Wilmington (with enclosures)
DH Hinds	GEH/Wilmington (with enclosures)
eDRF Section	0000-0094-2768 (RAI 7.7-7)
	0000-0094-2919 (RAI 7.7-8)

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Enclosure 1

Response to Portion of NRC Request for Additional Information Letter No. 251 Related to ESBWR Design Certification Application

Instrumentation & Control Systems

RAI Numbers 7.7-7 and 7.7-8

NRC RAI 7.7-7

ESBWR DCD Tier 2, Revision 5, Section 7.7.1.1.1, Safety Design Basis, identifies the reactor pressure vessel (RPV) water level and RPV pressure measurements and associated instruments as the safety-related portions of the nuclear boiler system (NBS) I&C. These instruments monitor process conditions to provide inputs to the reactor protection system (RPS) to initiate reactor scrams.

DCD Tier 2, Revision 5, Section 7.2.1.2.4.2 identifies six scram initiating circuits associated with NBS. Three of these scram initiating circuits correspond to the discussion of the safety-related portions of the NBS in DCD Tier 2 Section 7.7:

- *High reactor pressure (NBS),*
- *Low reactor pressure vessel (RPV) water level (Level 3) decreasing (NBS),*
- *High RPV water level (Level 8) increasing (NBS),*

However, the remaining three scram initiating circuits are not addressed in DCD Tier 2 Section 7.7:

- *Main steam line isolation valve (MSIV) closure (Run mode only) (NBS),*
- *High simulated thermal power (feedwater temperature biased) (NBS and neutron monitoring system (NMS)),*
- *Feedwater temperature exceeding allowable simulated thermal power vs. feedwater temperature domain (NBS),*

DCD Tier 2, Revision 5, Section 7.2.1.2.4.2 further identifies that the eight MSIVs and the 16 position switches supplied with these valves (for RPS use) and the eight temperature sensors (four on each FW line) associated with the feedwater temperature biased simulated thermal power scram are components of the NBS. Revise DCD Tier 2 Section 7.7 to address the safety-related portions of the NBS, including the items identified above. As appropriate, some safety related portions of the NBS may be addressed by referring to other sections of the DCD.

GEH Response

NBS safety design bases (subsection 7.7.1.1.1) are not related to any scram circuits as described in the RAI. The NBS safety design bases (subsection 7.7.1.1.1) currently discuss only the safety-related indications of RPV water level and reactor dome pressure.

However, in order to increase the clarity of the scope of DCD Section 7.7, GEH will remove the discussions of safety-related NBS indications. The NBS safety-related monitoring design bases are appropriately addressed in Section 7.1, Section 7.2, and Section 7.3, as well as meeting the design requirements of subsection 7.5.1, Post Accident Monitoring Instrumentation.

Furthermore, to assure the safety-related NBS supporting requirements are fully addressed in Sections 7.2 and 7.3 the following design safety evaluation discussions will be added:

- 10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation
- RG 1.151

With these changes, Section 7.7 will reflect only nonsafety-related portions of NBS. All the six scram initiating circuits associated with NBS (High reactor pressure, Low RPV water level, High RPV water level, MSIV closure, High simulated thermal power and Feed water temperature exceeding domain) are addressed in DCD Tier 2, Section 7.2.

DCD Impact

DCD Tier 2, Section 7.2, Section 7.3, and Section 7.7 will be revised in Revision 6 as shown in Enclosure 2.

NRC RAI 7.7-8

In ESBWR DCD Tier 2, Revision 5, Section 7.7, the first sentence states, "This section describes the Instrumentation and Control (I&C) systems for normal plant operation that do not perform plant safety-related functions." However, in the next paragraph, following Revision 5 changes, the first bullet specifically indicates that the Nuclear Boiler System (NBS) has safety-related subsystems which are described in DCD Tier 2, Revision 5, Section 7.7. Revise ESBWR DCD Tier 2 Section 7.7 to correct this inaccuracy.

GEH Response

GEH concurs with staff's request.

The first bullet of Section 7.7 will be revised to delete “and portions of safety-related subsystems ”.

Accordingly, Section 7.7 will be revised to remove the discussions of safety-related NBS indications.

Subsection 7.1.3.2.6 will be revised to add the operator monitoring data for safety-related NBS.

With these changes, Section 7.7 will reflect only nonsafety-related portions of NBS.

DCD Impact

DCD Tier 2, subsection 7.1.3.2.6 and Section 7.7 will be revised in Revision 6 as shown in Enclosure 2.

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Enclosure 2

Response to Portion of NRC Request for Additional Information Letter No. 251 Related to ESBWR Design Certification Application

DCD Markups for RAI Numbers 7.7-7 and 7.7-8

7.1.3.2.4.2 Containment Monitoring System

The CMS instrumentation measures and records radiation levels and the oxygen/hydrogen concentration levels in containment under post-accident conditions. The CMS is designed to operate continuously during normal operation and is automatically put in service upon detection of LOCA conditions. Refer to Subsection 7.5.2 for additional information.

7.1.3.2.4.3 Process Radiation Monitoring System

Safety-related PRMS instrumentation monitors the following for radioactive materials: discharges from the ICS vent, and ventilation discharges. The nonsafety-related PRMS is discussed in Subsection 7.1.5.2.2.1. The MCR display, recording, and alarm capabilities are provided along with controls that provide automatic trip inputs to the respective systems to prevent further radiation release. Refer to Subsection 11.5.3 for additional information.

7.1.3.2.5 Interlock ~~Systems~~Logic

The interlock logic functions are embedded in the DCIS logic, so that a separate interlock system is not required. Refer to Section 7.6 for additional information.

~~A reactor pressure interlock embedded in logic is provided to the GDCS to prohibit inadvertent manual initiation of the system during normal reactor operation.~~

~~Parallel pairs of air-operated, testable check valves, and motor operated on-off valves are provided to protect the FAPCS low pressure piping from over-pressurization during reactor power operation and high pressure transients and accidents. Redundant pressure instruments provide a high-pressure signal to the HP/LP interlock system. The HP/LP interlock system prevents motor operated valves from opening or closes them, if open and prevents the testing of the check valves when the reactor pressure exceeds the FAPCS low pressure setpoint.~~

~~Other than the isolation valves, t~~The design does not have logic that isolates safety-related from nonsafety-related piping during a LOCA. It is not necessary because there are no piping interfaces separating the safety-related and nonsafety-related portions of piping systems.

7.1.3.2.6 Nuclear Boiler System Instrumentation

Redundant NBS safety-related instrumentation provides ~~RPV water level and reactor vessel pressure~~ the following data for operator monitoring.

- RPV water level indicated in the MCR on displays associated with the different water level ranges;
- The reactor pressure indicated in the MCR and at four local instrument racks in the Reactor Building (RB);
- The discharge line temperatures of the SRVs viewed on safety-related video display units (VDUs) in the MCR. Any temperature exceeding the trip setting is alarmed to indicate leakage of a SRV seat;

- RPV temperature is indicated in the MCR, and high bottom head to reactor coolant differential temperature is alarmed in the MCR; and
- Main steam flow rate is indicated in the MCR.

The NBS instrumentation also provides inputs to the safety-related actuation systems during normal, transient, and accident conditions. Refer to Subsection 7.2 and 7.3~~7.7.1~~ for additional information.

7.1.3.2.7 Data Communication Systems

The DCIS data communication functions are embedded within the Q-DCIS and the N-DCIS architectures. Safety-related Q-DCIS internal and external communication protocols are deterministic.

7.1.3.3 Q-DCIS Safety Evaluation

All communication between the Q-DCIS and the N-DCIS is through safety-related CIMs, via datalinks and fiber optic cable. Fiber optic cable is also used for:

- Limited communication between the Q-DCIS divisions (such as the two-out-of-four voting logic),
- Communication within a division,
- Providing data to the VDU monitors, and
- Transferring VDU outputs corresponding to manual initiation actions.

The dual redundant fiber optic cable data networks described below replace the many conventional, long length, copper conductor cables of existing nuclear power plants. This reduces the cost and complexity of divisional cable runs that connect components of the plant protection and safety-related systems such as the RPS, MSIV isolation logic functions, LD&IS containment isolation functions, SSLC/ESF, and safety-related VDUs. The fiber optic cable provides transmission path immune from EMI for plant sensor data and safety-related system control signals.

7.1.3.3.1 Safety-Related Isolation

The use of fiber optic cable provides complete electrical isolation between components and noise free communication pathways, but is not credited for either the safety-related isolation or the safety-related separation. The safety-related fiber optic CIMs are the isolation devices, including data isolation, and convert signals between electricity and light on the safety-related side of the fiber optic cable. These safety-related fiber optic CIMs are powered by the division within which they are physically located. The safety-related fiber optic CIMs, which provide the safety-related isolation and separation, are qualified safety-related components.

The IEEE Std. 603, Sections 5.6 and 6.3, isolation and separation (electrical, physical, data, and communications) occurs in the safety-related fiber optic CIM (transmitter or receiver) where the confirmed by two-out-of-four coincidence of division trip outputs. A separate and diverse manual trip method is provided in the form of two independent manual trip channels. Actuation of both manual trip systems is required for a full reactor scram. Availability is enhanced because any one division can be bypassed at one time to allow online repair without degrading operability. This satisfies the repair requirement of IEEE Std. 603, Section 5.10 while maintaining plant availability.

The RPS consists of four redundant divisions identical in design and independent in operation. Although each division constitutes a separate trip system, normally each division can make two-out-of-four trip decisions with or without a division of sensors being bypassed. There are four instrument channels provided for each process variable being monitored, one for each RPS division. Four sensors, one per division, are provided for each variable. When more than four sensors are required to monitor a variable the outputs of the sensors are combined into only four instrument channels. The logic in each division does not depend on absolute time of day and is asynchronous with respect to the other divisions. No division depends on the correct operation of another division. There is no combination of MCR-initiated bypasses that can unacceptably degrade the RPS.

7.2.1.3 Safety Evaluation

Table 7.1-1 identifies the RPS and the associated codes and standards applied, in accordance with the Standard Review Plan NUREG-0800. This subsection addresses I&C systems conformance to regulatory requirements, guidelines, and industry standards.

7.2.1.3.1 Code of Federal Regulations

10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation:

- Conformance: NBS provides the reactor water level measurement (temperature compensated) inputs to RPS. The reactor water level instrumentation errors due to non-condensable gases in instrument reference legs are addressed in Subsection 7.7.1.2.2.

10 CFR 50.55a(a)(1), Quality Standards for Systems Important to Safety:

- Conformance: The RPS conforms to these standards.

10 CFR 50.55a(h), Protection and Safety Systems, compliance with IEEE Std. 603:

- Conformance: Safety-related systems are designed to conform to Regulatory Guide (RG) 1.153 and IEEE Std. 603. Separation and isolation are preserved both mechanically and electrically in accordance with IEEE Std. 603, Section 5.6 and RG 1.75. The RPS is divisionalized and is designed with redundancy so that failure of any instrument does not interfere with the system operation. Electrical separation is maintained between the redundant divisions.

10 CFR 50.34(f)(2)(v) [I.D.3], Bypass and Inoperable Status Indication:

- Conformance: The RPS design of bypass and inoperable status indication conforms to these requirements and is consistent with the conformance of the RPS design to RG 1.47.

7.2.1.3.4 Regulatory Guides

RG 1.151, Instrument Sensing Lines:

- Conformance: NBS provides the measurement inputs to RPS. The NBS instrument sensing lines conform to the guidelines of RG 1.151 and ISA-67.02.01. Flow restrictors are provided inside containment on instrument lines connected to the RCPB. Accessible manual isolation valves and self-actuating excess flow check valves are provided outside the drywell. The mechanical design guidelines as defined by ISA-67.02.01 and RG 1.151 are met as applicable for each installation.

RG 1.22, Periodic Testing of Protection System Actuation Functions - This includes conformance to BTP HICB-8:

- Conformance: The system is capable of being tested, from sensor device to final actuator device, during plant operation. The tests must be performed in overlapping stages so an actual reactor scram would not occur as a result of the testing. The portions of the protection systems subject to periodic testing are designed in accordance with IEEE Std. 603, Sections 5.7 and 6.5, which supersedes IEEE Std. 279.

RG 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems:

- Conformance: Automatic indication that a system is out of service is provided in the MCR (IEEE Std. 603, Section 5.8). Indicators show which part of a system is not operable and which division is bypassed. Annunciator test switches are provided in the MCR.

  Individual indicators are arranged together in the MCR to indicate which function of the system is out of service, bypassed, or otherwise inoperable. These automatic indicators remain available, and cannot be cleared until the function is operable (IEEE Std. 603, Sections 5.2 and 5.8).

  A manual switch or push button is provided for manual bypass actuation, which annunciates out-of-service conditions (IEEE Std. 603, Section 5.8).

  These display provisions serve to supplement administrative controls and aid the operator in assessing the availability of component and system-level protective actions (IEEE Std. 603, Section 5.8). These displays do not perform a safety-related function (IEEE Std. 603, Section 5.7).

  System out-of-service alarm circuits are electrically isolated from the plant safety-related systems to prevent adverse effects (IEEE Std. 603, Section 5.7).

  Testing is included on a periodic basis when equipment associated with the display is tested.

RG 1.53, Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems:

- Conformance: Compliance with RG 1.53 is satisfied by specifying, designing, and constructing the RPS to meet the single-failure criterion of IEEE Std. 603, Section 5.1,

Table 7.1-1 identifies the ADS and the associated codes and standards applied, in accordance with the SRP. This subsection addresses I&C systems conformance to regulatory requirements, guidelines, and industry standards.

7.3.1.1.3.1 Code of Federal Regulations

10 CFR 50.55a(a)(1), Quality Standards for Systems Important to Safety:

- Conformance: The ADS design complies with 10 CFR 50.55a(a)(1).

10 CFR 50.55a(h), Protection and Safety Systems compliance with IEEE Std. 603:

- Conformance: The ADS design complies with IEEE Std. 603. Separation and isolation are preserved both mechanically and electrically in accordance with IEEE Std. 603, Section 5.6 and RG 1.75. The ADS is divisionalized and designed with redundancy so failure of any instrument will not interfere with the system operation. Electrical separation is maintained between the redundant divisions.

10 CFR 50.34 (f) (2) (v) (I.D.3), Bypass and Inoperable Status Indication:

- Conformance: The ADS design complies with 10 CFR 50.34 (f) (2) (v) (I.D.3).

10 CFR 50.34 (f) (2) (xiv) (II.E.4.2), Containment Isolation Systems:

- Conformance: The ADS design complies with 10 CFR 50.34 (f) (2) (xiv) (II.E.4.2).

10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation:

- Conformance: NBS provides the reactor water level measurement (temperature compensated) inputs to ADS. The reactor water level instrumentation errors due to non-condensable gases in instrument reference legs are addressed in Subsection 7.7.1.2.2.

10 CFR 52.47(a)(1)(iv), Resolution of Unresolved and Generic Safety Issues:

- Conformance: Resolution of unresolved and generic safety issues is discussed in Section 1.11.

10 CFR 52.47(a)(1)(vi), ITAAC in Design Certification Applications:

- Conformance: ITAAC are provided for I&C systems and equipment in Tier 1.

10 CFR 52.47(a)(1)(vii), Interface Requirements:

- Conformance: There are no interface requirements for this section.

10 CFR 52.47(a)(2), Level of Detail:

- Conformance: The level of detail provided for the ADS within the DCD conforms to this requirement.

10 CFR 52.47(b)(2)(i), Innovative Means of Accomplishing Safety Functions:

RG 1.100, Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants:

- Conformance: The ADS design conforms to RG 1.100.

RG 1.105, Instrument Setpoints for Safety-Related Systems:

- Conformance: The setpoints used to initiate the ADS are consistent with RG 1.105. Because the discrete setpoints in the ADS logic do not drift, most of the variation is expected to be in the process transmitters. Setpoints are continuously monitored and alarmed by the PCF. Reference 7.3-2 provides a detailed description of the GEH setpoint methodology.

RG 1.118, Periodic Testing of Electric Power and Protection Systems:

- Conformance: The ADS design conforms to the guidance of RG 1.118. A full functional test of the ADS is not practical, because a LOCA results if the non-reclosable DPVs are opened. Acceptable reliability of equipment operation is demonstrated by alternate test methods. System logic is periodically self-tested, and initiating circuits are continuously monitored. DPV valve initiators periodically are removed and test-fired in a laboratory. RPV level transmitters are located outside containment, so calibration verification can be performed during plant operation.

RG 1.151, Instrument Sensing Lines:

- Conformance: NBS provides the measurement inputs to ADS. The NBS instrument sensing lines conform to the guidelines of RG 1.151 and ISA-67.02.01. Flow restrictors are provided inside containment on instrument lines connected to the RCPB. Accessible manual isolation valves and self-actuating excess flow check valves are provided outside the drywell. The mechanical design guidelines as defined by ISA-67.02.01 and RG 1.151 are met as applicable for each installation.

RG 1.152, Criteria for Digital computers in Safety Systems of Nuclear Power Plants:

- Conformance: The ADS design conforms to RG 1.152.

RG 1.153, Criteria for Power, Instrumentation, and Control Portions of Safety Systems:

- Conformance: The ADS design conforms to RG 1.153.

RG 1.168, Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The ADS design conforms to RG 1.168.

RG 1.169, Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear power Plants:

- Conformance: The ADS design conforms to RG 1.169.

The GDCS has no equipment protective interlocks that could interrupt automatic system operation. To initiate the GDCS injection and equalization systems manually, a RPV low-pressure signal must be present. This prevents system initiation while the reactor is at operating pressure. The GDCS injection and equalizing functions are designed to operate from safety-related power. The system instrumentation is powered by divisionally separated safety-related power. The injection squib valve, and the equalizing squib valve logic and initiation circuitry is powered by divisionally separated, safety-related power (Refer to Section 8.3). The mechanical aspects of the GDCS are discussed in Subsection 6.3.2.

The two deluge system temperature switches and related contacts are safety-related only to prevent the inadvertent actuation of the deluge valves. No single failure within the deluge system control and monitoring equipment causes an inadvertent actuation of the deluge system (IEEE Std. 603, Section 5.1). This is to protect against inadvertently draining the GDCS pools, thereby preventing the injection and equalizing systems from performing their safety functions.

Table 7.1-1 identifies the GDCS and the associated codes and standards applied in accordance with the SRP. This subsection addresses I&C systems conformance to regulatory requirements, guidelines, and industry standards. Any exceptions or clarifications are so noted.

7.3.1.2.3.1 Code of Federal Regulations

10 CFR 50.55a(a)(1), Quality Standards for Systems Important to Safety:

- Conformance: The GDCS design complies with these standards.

10 CFR 50.55a(h), Protection and Safety Systems, compliance with IEEE Std. 603:

- Conformance: Safety-related systems conform to RG 1.153 and IEEE Std. 603. Separation and isolation are preserved both mechanically and electrically in accordance with IEEE Std. 603, Section 5.6 and RG 1.75. The GDCS is divisionalized and designed with redundancy so failure of any instrument will not interfere with the system operation. Electrical separation is maintained between the redundant divisions.

10 CFR 50.34(f)(2)(v)(I.D.3), Bypass and Inoperable Status Indication:

- Conformance: The GDCS design complies by providing automatic indication of bypassed and inoperable status (IEEE Std. 603, Section 5.8).

10 CFR 50.34(f)(2)(xiv)(II.E.4.2), Containment Isolation Systems:

- Conformance: The GDCS design complies with this requirement.

10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation:

- Conformance: NBS provides the reactor water level measurement (temperature compensated) inputs to GDCS. The reactor water level instrumentation errors due to non-condensable gases in instrument reference legs are addressed in Subsection 7.7.1.2.2.

10 CFR 52.47(a)(1)(iv), Resolution of Unresolved and Generic Safety Issues:

- Conformance: The GDCS design complies with RG 1.53, IEEE Std. 603, Section 5.1, and IEEE Std. 379.

RG 1.62, Manual Initiation of Protective Actions:

- Conformance: The GDCS design complies with RG 1.62. Each division of the GDCS has a manual actuation switch in the MCR. Initiation of the system requires actuation of two switches to ensure that manual initiation is a premeditated act. There is an interlock between the manual initiation switches and a low reactor-pressure signal. This interlock prevents manual initiation of the system if the RPV is not depressurized.

RG 1.75, Physical Independence of Electric Systems:

- The GDCS design conforms to RG 1.75 as described in Subsections 8.3.1.3 and 8.3.1.4.

RG 1.89, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants:

- Conformance: The GDCS design conforms to RG 1.89.

RG 1.100, Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants:

- Conformance: The GDCS design conforms to RG 1.100.

RG 1.105, Instrument Setpoints for Safety-Related Systems:

- Conformance: The setpoints used to initiate GDCS are established consistent with RG 1.105. Reference 7.3-2 provides a detailed description of the GEH methodology.

RG 1.118, Periodic Testing of Electric Power and Protection Systems:

- Conformance: The GDCS design complies with the guidance of RG 1.118.

RG 1.151, Instrument Sensing Lines:

- Conformance: NBS provides the measurement inputs to GDCS. The NBS instrument sensing lines conform to the guidelines of RG 1.151 and ISA-67.02.01. Flow restrictors are provided inside containment on instrument lines connected to the RCPB. Accessible manual isolation valves and self-actuating excess flow check valves are provided outside the drywell. The mechanical design guidelines as defined by ISA-67.02.01 and RG 1.151 are met as applicable for each installation.~~The GDCS design complies with the guidance of RG 1.151.~~

RG 1.152, Criteria for Digital computers in Safety Systems of Nuclear Power Plants:

- Conformance: The GDCS design conforms to RG 1.152.

RG 1.153, Criteria for Power, I&C Portions of Safety Systems:

- Conformance: The GDCS design complies with RG 1.153.

- Conformance: Separation and isolation is preserved both mechanically and electrically in accordance with IEEE Std. 603, Section 5.6, and RG 1.75. The LD&IS consists of four redundantly designed divisions so failure of any instrument will not interfere with the system operation. Electrical separation is maintained between the redundant divisions.

10 CFR 50.34(f)(2)(v)(I.D.3), Bypass and Inoperable Status Indication:

- Conformance: The LD&IS design complies by providing automatic indication of bypassed and inoperable status (IEEE Std. 603, Section 5.8).

10 CFR 50.34(f)(2)(xiv)(II.E.4.2), TMI Action Plan Item IIE.4.2, Containment Isolation Systems:

- Conformance: The LD&IS design complies with this requirement.

10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation:

- Conformance: NBS provides the reactor water level measurement (temperature compensated) inputs to LD&IS. The reactor water level instrumentation errors due to non-condensable gases in instrument reference legs are addressed in Subsection 7.7.1.2.2

10 CFR 52.47(a)(1)(iv), Resolution of Unresolved and Generic Safety Issues:

- Conformance: The resolution of unresolved and generic safety issues is discussed in Section 1.11.

10 CFR 52.47(a)(1)(vi), ITAAC in Design Certification Applications:

- Conformance: ITAAC are provided for the I&C systems and equipment in Tier 1.

10 CFR 52.47(a)(1)(vii), Interface Requirements:

- Conformance: There are no interface requirements for this section.

10 CFR 52.47(a)(2), Level of Detail:

- Conformance: The level of detail provided for the LD&IS in the DCD conforms to this requirement.

10 CFR 52.47(b)(2)(i), Innovative Means of Accomplishing Safety Functions:

- Conformance: The I&C design does not use innovative means for accomplishing safety functions.

7.3.3.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 20, 21, 22, 23, ~~and~~ 24 and 29:

- Conformance: The LD&IS design complies with these GDCs.

RG 1.151, Instrument Sensing Lines:

- Conformance: NBS provides the measurement inputs to LD&IS. The NBS instrument sensing lines conform to the guidelines of RG 1.151 and ISA-67.02.01. Flow restrictors are provided inside containment on instrument lines connected to the RCPB. Accessible manual isolation valves and self-actuating excess flow check valves are provided outside the drywell. The mechanical design guidelines as defined by ISA-67.02.01 and RG 1.151 are met as applicable for each installation.

RG 1.152, Criteria for Digital computers in Safety Systems of Nuclear Power Plants:

- Conformance: The LD&IS design conforms to RG 1.152.

RG 1.153, Power Instrumentation & Control Portions of Safety Systems:

- Conformance: The LD&IS design conforms to RG 1.153.

RG 1.168, Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The LD&IS design conforms to RG 1.168.

RG 1.169, Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear power Plants:

- Conformance: The LD&IS design conforms to RG 1.169.

RG 1.170, Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The LD&IS design conforms to RG 1.170.

RG 1.171, Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The LD&IS design conforms to RG 1.171.

RG 1.172, Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The LD&IS design conforms to RG 1.172.

RG 1.173, Developing Software Life Cycle Processes for Digital Computer Software used in Safety Systems of Nuclear Power Plants:

- Conformance: The LD&IS design conforms to RG 1.173.

RG 1.180, Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in safety-related I&C Systems:

- Conformance: The LD&IS design conforms to RG 1.180.

10 CFR 50.34 (f)(2)(xiv) [II.E.4.2], Containment Isolation Systems:

- Conformance: The SSLC/ESF logic controlling containment isolation functions conforms to these criteria.

10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation:

- Conformance: NBS provides the reactor water level measurement (temperature compensated) inputs to SSLC/ESF. The reactor water level instrumentation errors due to non-condensable gases in instrument reference legs are addressed in Subsection 7.7.1.2.2.

10 CFR 50.34 (f)(2)(xxiii) [II.K.2.10], Anticipatory Reactor Trip:

- Conformance: The SSLC/ESF initiates the ICS in response to a Loss of All Feedwater Flow Event. This is an anticipatory trip actuated on loss of power to two of the four main FW pumps.

10 CFR 52.47(a)(1)(iv), Resolution of Unresolved and Generic Safety Issues:

- Conformance: Resolution of unresolved and generic safety issues is discussed in Section 1.11.

10 CFR 52.47(a)(1)(vi), ITAAC in Design Certification Applications:

- Conformance: ITAAC are provided for the I&C systems and equipment in Tier 1.

10 CFR 52.47(a)(1)(vii), Interface Requirements:

- Conformance: There are no interface requirements for this section.

10 CFR 52.47(a)(2), Level of Detail:

- Conformance: The level of detail provided for the SSLC/ESF within the DCD conforms to this requirement.

10 CFR 52.47(b)(2)(i), Innovative Means of Accomplishing Safety Functions:

- Conformance: The I&C design does not use innovative means for accomplishing safety functions.

7.3.5.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 20, 21, 22, 23, ~~and~~ 24 and 29:

- Conformance: The SSLC/ESF design complies with these GDCs.

7.3.5.3.3 Staff Requirements Memorandum

SRM on SECY-93-087, Item II.Q Defense Against Common-Mode Failures in Digital Instrument and Control Systems:

- Conformance: The SSLC/ESF design conforms to RG 1.118 as amplified in IEEE Std. 338. Testing of the SSLC/ESF is performed in conjunction with the Q-DCIS.

RG 1.151, Instrument Sensing Lines:

- Conformance: NBS provides the measurement inputs to SSLC/ESF. The NBS instrument sensing lines conform to the guidelines of RG 1.151 and ISA-67.02.01. Flow restrictors are provided inside containment on instrument lines connected to the RCPB. Accessible manual isolation valves and self-actuating excess flow check valves are provided outside the drywell. The mechanical design guidelines as defined by ISA-67.02.01 and RG 1.151 are met as applicable for each installation.

RG 1.152, Criteria for Programmable Digital Computer System Software in safety-related Systems of Nuclear Power Plants:

- Conformance: The SSLC/ESF design conforms to the guidelines of RG 1.152. Additional discussion is provided in Subsection 7.2.1.3 describing RPS system compliance.

RG 1.153, Criteria for Power, Instrumentation, and Control Portions of Safety Systems:

- Conformance: The SSLC/ESF design, in conjunction with the Q-DCIS, conforms to RG 1.153.

RG 1.168, Verification, Validation, Reviews, and Audits for Digital Computer Software Used In Safety Systems of Nuclear Power Plants:

- Conformance: The SSLC/ESF design complies with RG 1.168.

RG 1.169, Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The SSLC/ESF design complies with RG 1.169.

RG 1.170, Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The SSLC/ESF design complies with RG 1.170.

RG 1.171, Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The SSLC/ESF design complies with RG 1.171.

RG 1.172, Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

- Conformance: The SSLC/ESF design complies with RG 1.172.

RG 1.173, Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants:

7.7 CONTROL SYSTEMS

This section describes the Instrumentation and Control (I&C) systems for normal plant operation that do not perform plant safety-related functions. However, these systems do control plant processes that have a significant effect on plant safety. These systems can affect the performance of safety-related functions either through normal operation or through inadvertent operation. The systems described in this section include:

- The Nuclear Boiler System (NBS) – nonsafety-related subsystems, ~~and portions of safety-related subsystems,~~
- Rod Control and Information System (RC&IS),
- Feedwater Control System (FWCS),
- Plant Automation System (PAS),
- Steam Bypass and Pressure Control (SB&PC) System,
- Neutron Monitoring System (NMS) - nonsafety-related subsystems, and
- Containment Inerting System (CIS).

~~The safety-related monitoring and control portions of the NBS and NMS are part of a group of systems that is collectively referred to as the Safety Related Distributed Control and Information System (Q-DCIS). A functional block diagram of the Q-DCIS is included as part of Figure 7.1-1, and a functional network diagram appears as Figure 7.1-2. These diagrams indicate the relationships of the NBS and NMS with their safety-related peers and with nonsafety-related plant data systems. Section 7.1 contains a description of these relationships.~~

The nonsafety-related monitoring and control for the RC&IS, FWCS, PAS, SB&PC System, NMS, NBS and CIS is part of a group of systems that is collectively referred to as the Nonsafety-Related Distributed Control and Information System (N-DCIS). A functional block diagram of the N-DCIS is included as part of Figure 7.1-1, and a functional network diagram appears as Figure 7.1-2. These diagrams indicate the relationships of RC&IS, FWCS, PAS, SB&PC System, NMS, NBS and CIS with their nonsafety-related peers and with safety-related plant data systems that are collectively referred to as the Q-DCIS. Section 7.1 contains a description of these relationships.

7.7.1 Nuclear Boiler System

The NBS instrumentation provides monitoring and control input for operational variables during normal plant operating modes and during the plant response to accidents. The NBS sensors used for safety-related system actuation and control functions are addressed in other subsections within this chapter. This subsection describes ~~only the safety-related NBS instrumentation used for indication and~~ only those NBS instruments used for actuation and control of nonsafety-related systems.

7.7.1.1 System Design Bases

7.7.1.1.1 Safety Design Bases

Section 7.7 addresses only the non-safety related portion of the NBS instruments.~~The safety-related portions of the NBS I&C meet the following safety-related requirements (IEEE Std. 603, Sections 4.1, 4.2, 4.8, and 4.10).~~

- ~~Provide reactor pressure vessel (RPV) water level and dome pressure measurements over the ranges and to the accuracies necessary for adequate operator monitoring of RPV water level during normal, transient, and accident conditions.~~
- ~~Instrumentation provided for RPV water level and dome pressure measurement is qualified for the design basis loadings of the Safe Shutdown Earthquake (SSE), loadings associated with design basis accidents (DBAs), and the environmental conditions associated with DBAs.~~
- ~~Provide divisionally separated instruments so that a single failure does not result in the loss of level or pressure indication.~~

7.7.1.1.2 Power Generation (Non-safety) Design Bases

The nonsafety-related portions of the NBS instrumentation meet power generation requirements by providing indication of parameters in support of normal plant operations. These parameters are:

- Reactor coolant and RPV temperatures;
- RPV water level:
  - Shutdown range,
  - Narrow range,
  - Wide range, and
  - Fuel zone range.
- RPV pressure;
- Safety relief valve discharge line temperature; and
- Main steam flow rate.

The NBS design provides for periodic calibration and testing of its instrumentation during plant operation.

7.7.1.2 System Description

7.7.1.2.1 Summary Description

The NBS instruments are used to provide the operator with information during normal, transient, accident, and post-accident conditions. The NBS instruments measure the reactor coolant temperature, RPV temperature, RPV water level, RPV pressure, main steam flow rate, and detect SRV leakage.

~~Safety-related instrumentation is powered from a safety-related 120 VAC Uninterruptible Power Supply (UPS).~~ Nonsafety-related instruments are powered from the nonsafety-related instrument power supply buses.

For instruments that are located below the process tap, including the RPV water level measurements, the sensing line slopes downward from the process tap to the instrument to preclude air traps. Where it is impractical to locate the instruments below the process connection, the sensing lines descend below the process connection before sloping upward to a high point vent located at an accessible location with a fill connection. This permits filling and venting of noncondensible gases from the sensing line during calibration procedures.

Level and pressure sensing lines, up to the outboard excess flow check valve, are connected to the Reactor Coolant Pressure Boundary (RCPB) and are classified as Quality Group A, ASME Section III, safety-related, and Seismic Category I. The typical arrangement for these sensing lines is a restricting orifice located inside the containment and a manual isolation valve located outside the containment, which is followed by an excess flow check valve.

7.7.1.2.2 Detailed System Description

Reactor Coolant and Reactor Pressure Vessel Temperature Monitoring System

The reactor coolant temperature is measured at the mid-vessel inlet to the Reactor Water Cleanup and Shutdown Cooling (RWCU/SDC) system and at the bottom head drain. Coolant temperature can also be determined in the steam-filled parts of the RPV and steam-water mixture by measuring the reactor pressure. In the saturated system, reactor pressure connotes saturation temperature. Coolant temperatures (core inlet temperature) can normally be measured by the redundant core inlet temperature sensors located in each Local Power Range Monitor (LPRM) assembly below the core plate elevation.

The RPV outside surface temperature is measured at the head flange and at the bottom head locations. Temperatures needed for operation and for compliance with the Technical Specification operating limits are obtained from these measurements.

Reactor Pressure Vessel Water Level

Figure 7.7-1 shows the water level range and the vessel penetrations for each water level range. The instruments are differential pressure devices calibrated for the specific vessel pressure and liquid temperature conditions. The reactor water level measurement is temperature compensated through the thermocouples installed on the sensing line. As described in Subsection 4.6.1.2.4, the Control Rod Drive Hydraulic Subsystem provides a purge flow that keeps the RPV water level reference leg instrument lines full. These lines are filled to address the effects of noncondensible gases in the instrument lines and to prevent erroneous reference information after a rapid RPV depressurization event. The reactor water level instrumentation is referenced to level zero, which is at the Top of Active Fuel (TAF).

Reactor water level instrumentation that initiates safety-related system functions and engineered safety features (ESF) system functions is discussed in Subsections 7.2.1 and 7.3.1. Reactor water level instrumentation that is used as part of the FWCS is discussed in Subsection 7.7.3. Reactor water level instrumentation used for Diverse Protection System (DPS) functions is discussed in Subsection 7.8.1.

The Shutdown Range Water Level is used to monitor the RPV water level during shutdown conditions when the RPV head is removed, including when the reactor system is flooded for refueling or maintenance. The water level measurement design method is the condensing chamber reference leg type and uses differential pressure devices as its primary elements. The vessel temperature and pressure conditions that are used for the calibration are given in Section 5.1. The two vessel instrument nozzles used for this water level measurement are located at the top of the RPV head and just below the bottom of the dryer skirt.

The Narrow Range Water Level uses the RPV taps near the top of the steam outlet nozzle and the taps near the bottom of the dryer skirt. The instruments are calibrated to be accurate during normal reactor operating conditions. The method of water level measurement is the condensing chamber reference leg type and uses differential pressure devices as its primary elements. The FWCS uses this range for its water level control and indication inputs. Refer to Subsection 7.7.3 for more information on the FWCS.

The Wide Range Water Level uses the RPV taps above the TAF. The upper taps are also used for the Narrow Range Water Level. The instruments are calibrated to be accurate at normal power operating conditions. The water level measurement method is the condensing chamber reference leg type and uses differential pressure devices as its primary elements. Information from the RPV Wide Range Water Level instrumentation is used for safety-related and nonsafety-related applications, for DPS, and is provided for the range of normal, transient, and accident conditions. Separate sensors and indicators are provided for Wide Range Water Level indication.

The Fuel Zone Range Water Level uses the RPV taps near the top of the steam outlet nozzle and the taps below the bottom of the active fuel. The instruments are calibrated to be accurate at zero Pa gauge (0 psig) and saturated conditions. The water level measurement method is the condensing chamber reference type and uses differential pressure devices as its primary elements. The RPV Fuel Zone Water Level instrumentation ~~is safety-related and~~ is provided for post-accident monitoring situations in which the water level is substantially below the normal range. Separate sensors and indicators are provided for ~~Wide~~ Fuel Zone Range Water Level indication.

Reactor Pressure Vessel Pressure

Pressure transmitters detect RPV pressure from the instrument lines used for measuring RPV water level and provide indications in the Main Control Room (MCR).

~~Pressure transmitters and trip actuators for initiating a reactor scram, and pressure transmitters and trip actuators for bypassing the main steam line isolation valve (MSIV) closure scram are discussed in Subsection 7.2.1. High pressure and low pressure (HP/LP) interlocks are discussed in Subsection 7.6.1.1.~~

~~Pressure transmitters that are used for pressure recording are discussed in Subsection 7.2.1.2.~~

Safety Relief Valve Leak Detection

Thermocouples are located in the discharge pipes of ten Safety Relief Valves (SRVs) (Reference Subsection 5.2.5). The temperature signals are recorded, and temperatures indicative of a leaking SRV are alarmed in the MCR.

Main Steam Flow Rate

Differential pressure transmitters are used to infer main steam flow rate. Pressure taps from the throat of the RPV steam outlet nozzles, in conjunction with the RPV dome pressure taps, measure differential pressure. The square root of differential pressure is proportional to the main steam flow rate. ~~Safety-related transmitters provide input to the Leak Detection and Isolation System (LD&IS) logic.~~ Outputs from nonsafety-related transmitters are used for feedwater (FW) control.

7.7.1.3 Safety Evaluation

Section 7.7 addresses only the non-safety related portion of the NBS instruments.~~The safety-related reactor water level and dome pressure instruments are designed to withstand the loads and environmental conditions under which they must function. Sufficient separate sensors and indicators are provided so that a single failure cannot result in the loss of level indication. The combined range of the wide range and fuel zone water level instrumentation ensures that adequate level information is available over the entire range of postulated design basis accident conditions (IEEE Std. 603, Sections 4.8 and 5.1).~~

~~Table 7.1-1 identifies the safety-related NBS and the associated codes and standards applied, in accordance with the SRP. This subsection addresses I&C systems conformance to regulatory requirements, guidelines, and industry standards.~~

The nonsafety-related instruments discussed in this subsection are designed to operate under normal and peak operating conditions of system pressure and at ambient pressures and temperatures. Any mechanical interface between nonsafety-related instruments and safety-related instrument piping or the RCPB is classified as safety-related to avoid compromise of the safety-related sensing capability and/or the RCPB. If a line break occurs in a nonsafety-related portion of a sensing line, the excess flow check valve closes to stop the flow of reactor coolant. If there is a single failure of the excess flow check valve, a restriction orifice limits the flow of coolant to within acceptable bounds.

7.7.1.3.1 Code of Federal Regulations

10 CFR 50.34(f)(2)(xviii)[II.F.2], Inadequate Core Cooling Instrumentation:

- Conformance: Reactor water level instrumentation errors due to non-condensable gases in instrument reference legs are addressed in Subsection 7.7.1.2.2.

10 CFR 50.55a(a)(1), Quality Standards Important to Safety:

- Conformance: The NBS conforms to these criteria, as shown by the following commitments to applicable Regulatory Guides (RG) and standards.

10 CFR 50.55a(h), Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE Std. 603):

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~Safety-related portions of the NBS are designed to conform to IEEE Std. 603, as discussed in Subsection 7.2.1.3.4.~~

10 CFR 50.34(f)(2)(v) [I.D.3], Bypass and Inoperable Status Indication:

- Conformance: The NBS design of bypass and inoperable status indication conforms to this requirement.

10 CFR 52.47(a)(1)(iv), Resolution of Unresolved and Generic Safety Issues:

- Conformance: Resolution of unresolved and generic safety issues is discussed in Section 1.11.

10 CFR 52.47(a)(1)(vi), ITAAC in Design Certification Applications:

- Conformance: ITAAC are provided for the I&C systems and equipment in Tier 1.

10 CFR 52.47(a)(1)(vii), Interface Requirements:

- Conformance: There are no interface requirements for this section.

10 CFR 52.47(a)(2), Level of Detail:

- Conformance: The level of detail provided for the NBS within the DCD conforms to this regulation.

10 CFR 52.47(b)(2)(i), Innovative Means of Accomplishing Safety Functions:

- Conformance: The I&C design does not use innovative means for accomplishing safety functions.

7.7.1.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19 and 24:

- Conformance: The NBS design complies with these GDC.

7.7.1.3.3 Staff Requirements Memorandum

Staff Requirements Memorandum (SRM) on SECY 93-087 II.Q, Defense Against Common-Mode Failures in Digital Instrument and Control Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS.~~In addition to the features already incorporated in the design to provide defense in depth against common mode failures that address this SRM, the NBS ADS function and other ESF designs conform to Item II.Q of SECY-93-087, NRC Branch Technical Position (BTP) HICB-19 through the implementation of an additional Diverse Instrumentation and Control System described in Section 7.8.~~

7.7.1.3.4 Regulatory Guides

RG 1.75, Physical Independence of Electric Systems:

- Conformance: The NBS system design conforms to RG 1.75 as described in Subsections 8.3.1.3 and 8.3.1.4.

RG 1.89, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.89.~~

RG 1.100, Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.100.~~

RG 1.105, Instrument Setpoints for safety-related Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design comply with RG 1.105. Reference 7.7-3 provides a detailed description of the GEH setpoint methodology.~~

RGs 1.151, Instrument Sensing Lines:

- Conformance: The instrument sensing lines for the NBS instrumentation conform to the guidelines of RG 1.151 and ISA-67.02.01. Flow restrictors are provided inside containment on instrument lines connected to the RCPB. Accessible manual isolation valves and self-actuating excess flow check valves are provided outside the drywell. The mechanical design guidelines as defined by ISA-67.02.01 and RG 1.151 are met as applicable for each installation.

RG 1.152, Computer Software Used in Safety-related Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.152.~~

RG 1.153, Criteria for Power, Instrumentation, and Control Portions of Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design comply with RG 1.153.~~

RG 1.168, Verification, Validation, Reviews, and Audits For Digital Computer Software Used In Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.168.~~

RG 1.169, Configuration Management Plans For Digital Computer Software Used In Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.169.~~

RG 1.170, Software Test Documentation For Digital Computer Software Used In Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.170.~~

RG 1.171, Software Unit Testing For Digital Computer Software Used In Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.171~~.

RG 1.172, Software Requirements Specifications For Digital Computer Software Used In Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.172.~~

RG 1.173, Developing Software Life Cycle Processes For Digital Computer Software Used In Safety Systems:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.173.~~

RG 1.180, Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems.

- Conformance: The NBS design conforms to RG 1.180.

RG 1.204, Guidelines for Lightning Protection of Nuclear Power Plants:

- Conformance: The NBS design conforms to RG 1.204.

RG 1.209, Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants:

- Conformance: Not applicable to the nonsafety-related portions of NBS. ~~The safety-related portions of the NBS design conform to RG 1.209~~.

7.7.1.3.5 **Branch Technical Positions**

BTP HICB-11, Guidance on Application and Qualification of Isolation Devices:

- Conformance: The NBS design complies with BTP HICB-11.

BTP HICB-12, Guidance on Establishing and Maintaining Instrument Setpoints

- Conformance: Not applicable to the nonsafety-related portions of NBS.~~The safety-related portions of the NBS design comply with BTP HICB-12. Reference 7.7-3 provides a detailed description of the GEH setpoint methodology.~~

BTP HICB-16, Guidance on Level of Detail Required for Design Certification Applications Under 10 CFR Part 52:

- Conformance: The level of detail provided for this system complies with BTP HICB-16.

BTP HICB-14, 17, 18, 19, and 21 are discussed in association with the Safety System Logic and Control/Engineered Safety Features (SSLC/ESF) in Subsection 7.3.5.3.

7.7.1.4 Testing and Inspection Requirements

Calibration and testing of the various instruments are performed during preoperational testing to confirm that the instrumentation is installed correctly and performs as intended.

Pressure, differential pressure, water level, and flow instruments are located outside the drywell so that calibration and test signals can be applied during reactor operation. Temperature elements located inside the drywell can be tested and calibrated from junction boxes located outside the drywell ~~(IEEE Std. 603, Section 5.7)~~.

7.7.1.5 Instrumentation and Control Requirements

The information available to the reactor operator from the NBS instrumentation is discussed in sections 7.1.~~this subsection (IEEE Std. 603, Section 5.8) consists of~~:

- ~~RPV water level indicated in the MCR on displays associated with the different water level ranges;~~
- ~~The reactor pressure indicated in the MCR and at four local instrument racks in the Reactor Building (RB);~~
- ~~The discharge line temperatures of the SRVs viewed on safety-related video display units (VDUs) in the MCR. Any temperature exceeding the trip setting is alarmed to indicate leakage of a SRV seat;~~
- ~~RPV temperature is indicated and recorded in the MCR, and high bottom head to reactor coolant differential temperature is alarmed in the MCR; and~~

~~□Main steam flow rate is indicated in the MCR.~~

7.7.2 Rod Control and Information System

The main objective of the RC&IS is to control the Fine Motion Control Rod Drive (FMCRD) motors of the Control Rod Drive (CRD) (explained in Subsections 4.6.1 and 4.6.2) to permit changes in core reactivity so that reactor power level and power distribution can be controlled. The RC&IS acquires status and control rod position information from the CRD FMCRD instrumentation. The RC&IS sends purge water valve control signals to and acquires status signals from the Hydraulic Control Units (HCUs) of the CRD. The RC&IS also sends and receives status and control signals to and from other plant systems and RC&IS modules.

7.7.2.1 System Design Bases

7.7.2.1.1 Safety Design Bases

The RC&IS has no functional safety-related design basis and is designed so that it does not adversely affect functional capabilities of safety-related systems.

7.7.2.1.2 Power Generation (Non-safety) Design Bases

The RC&IS performs the following functions.

- Controls changes to core reactivity, and thereby reactor power, by moving neutron absorbing control rods within the reactor core as initiated by the following.
  - The plant operator, when the RC&IS is in a manual or semiautomatic mode of operation.
  - The automatic rod movement mode of the PAS, when the RC&IS is in an automatic mode of operation.
- Displays summary information to the plant operator about positions of the control rods in the core and the status of the FMCRDs and RC&IS. This summary information is provided by a RC&IS dedicated operator interface in the MCR. There are dual redundant measurements of the absolute rod position during normal FMCRD conditions. If one position detector fails for an individual FMCRD, the failed position detector can be bypassed, and the reactor can continue to operate without power restrictions.
- Provides RC&IS and FMCRD status data and control rod position data to other plant systems that require this data, such as the N-DCIS.
- Provides automatic, electric motor Run-in of all operable control rods, following detection of activation of the hydraulic insertion of the control rods, by a reactor scram. This is called the scram-follow function.
- Automatically enforces rod movement blocks to prevent potentially undesirable rod movements. These blocks do not affect a hydraulic scram insertion function, the scram-