MFN 09-096

Enclosure 2

Response to NRC Request for Additional Information Letter No. 261 Related to ESBWR Design Certification Application

Instrumentation & Control Systems

DCD Markups for RAI Number 7.5-8

condensation in tubes in the presence of noncondensable gases. A test program was conducted to secure this information, reported to the NRC in Reference 1.5-9.

The Single Tube Condensation Test Program was conducted to investigate steam condensation inside tubes in the presence of noncondensable gases. The work was independently conducted at the University of California at Berkeley and at the Massachusetts Institute of Technology (MIT). The work was initiated in order to obtain a data base and a correlation for heat transfer in similar conditions as would occur in the SBWR/ESBWR PCCS tubes during a DBA LOCA. Three researchers utilized three separate experimental configurations at the University of California at Berkeley, while two researchers utilized one configuration at MIT. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and noncondensable mass fractions. The experimenters found the system to be well behaved for all tests, with either of the noncondensable gases, for forced flow conditions similar to the ESBWR design. The results of the tests at the University of California at Berkeley have become the basis for the condensation heat transfer correlation used in the TRACG computer code.

While all SBWR/ESBWR features are extrapolations from current and previous designs, two features (specifically, the Passive Containment Cooling System and the Gravity-Driven Cooling System) represent the two most challenging extrapolations. Therefore, it was decided, for these two cases, to obtain additional test data, which could be used to demonstrate the capabilities of TRACG to successfully predict SBWR/ESBWR performance over a range of conditions and scales. Blind (in some cases double blind) predictions of test facility response use only the internal correlations of TRACG. No "tuning" of the TRACG inputs was performed, and no modifications to the coding were anticipated as a result of these tests.

For the case of the PCCS, the steady state heat exchanger performance was predicted in full-vertical-scale 3-tube (GIRAFFE), 20-tube (PANDA), and prototypical 496-tube (PANTHERS) configurations, over the range of steam and noncondensable conditions expected for the SBWR. This process addresses scale and geometry differences between the basic phenomena tests performed in single tubes, and larger scales including prototype conditions. Transient performance was similarly investigated at two different scales in both GIRAFFE and PANDA.

TRACG GDCS performance predictions were performed against the GIST and GIRAFFE/SIT test series. Pre-test predictions have also been performed for the PANTHERS and PANDA steady state tests.

~~Compliance with 10 CFR 52.47~~ Additional Design Requirements

~~10 CFR 52.47(b)(2)(i)(A) requires in part that:~~

- The performance of each safety feature of the design has been demonstrated through analysis, appropriate test programs, experience, or a combination thereof.
- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.
- Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

The ESBWR meets the above requirements, as discussed below:

- ESBWR plant features have been used in earlier BWR designs and most continue in operation today after many years and over a very large number of combined plant operating years of service. While the details of the particular plant feature design for the ESBWR may differ somewhat from those in current plants, the function of each feature is substantially the same. ~~This experience constitutes a sufficient database to meet the requirements of 10 CFR 52.47(b)(2)(i)(A)(1).~~
- In those scenarios in which ESBWR safety features come into operation, no other systems are required and, therefore, system interactions are not an issue, or the system designs are similar in the ESBWR and the operating plants having the feature. The operating plant feature(s) perform under the same general conditions and for the same scenarios as are anticipated to occur in the ESBWR. ~~The operating plant database is sufficient to meet requirements of 10 CFR 52.47(b)(2)(i)(A)(2) and (3).~~
- Feature performance has been predicted with the TRACG computer program. TRACG has been qualified by comparison to data from experiments and operating BWRs over a wide range of reactor conditions, including temperatures and pressures during which the features are expected to operate. The TRACG analyses add to the confidence that the features would perform as expected ~~and reinforce the GEH position that the requirements of 10 CFR 52.47(b)(2)(i)(A)(1), (2) and (3) have been met~~. The NRC safety evaluation report for Reference 1.5-3 concludes that no further testing in support of the thermal hydraulic behavior of the design is necessary.

The detailed design of specific ESBWR plant equipment is, in some cases, not specified in the ESBWR DCD; in some instances, only the design requirements of the equipment are given. When this is the case, a requirement for hardware testing is not appropriate under the certification program. However, because the ESBWR-specific hardware design differs from that currently in use, GEH believes that testing before application of a specific equipment design in a plant should be planned. Therefore, testing of plant hardware is done prior to or during startup testing of the plant.

For any ESBWR constructed, equipment performance will be demonstrated. For example, overall testing of the heat rejection capability of the isolation condensers is to be included as part of the plant startup test program. No ESBWR plant will operate until plant-specific tests confirm that each isolation condenser meets the performance requirements. Full-scale tests of an isolation condenser module in the PANTHERS test facility, as well as experience with condensing heat exchangers in many industries give high confidence that the requirements will be met.

1.5.3.1 Major ESBWR Unique Test Programs

As noted previously, the vast majority of data supporting the ESBWR design were generated using the design of the previous BWR product lines. ESBWR-unique certification and confirmatory tests applicable to its design are listed below.

GIST (Confirmatory)

GIST is an experimental program conducted by GE to demonstrate the Gravity-Driven Cooling System (GDCS) concept and to collect data to qualify the TRACG computer code for ESBWR

1.9 CONFORMANCE WITH STANDARD REVIEW PLAN AND APPLICABILITY OF CODES AND STANDARDS

1.9.1 Conformance with Standard Review Plan

This subsection provides the information required by 10 CFR 52.47(a)(9)~~50.34(h)~~ showing conformance with the Standard Review Plan (SRP). The summary of differences from requirements in each SRP section is presented on a section by section basis in Tables 1.9-1 through 1.9-19. If no difference is indicated, the ESBWR design does not deviate from the requirements in the SRP section. For SRP sections where there are deviations, a reference location is provided for additional information.

1.9.2 Applicability to Regulatory Criteria

Standard Review Plans, Branch Technical Positions, Regulatory Guides and Industrial Codes and Standards, which are applicable to the ESBWR design, are provided in Tables 1.9-20, 1.9-21 and 1.9-22. Applicable revisions are also shown. The applicability column of Tables 1.9-20 and 1.9-21 refers to whether or not the requirement is applicable during Design Certification of the ESBWR. Standard Review Plans, Branch Technical Positions, and Regulatory Guides that apply only during detailed design, construction, fabrication and erection are indicated by a dash in the applicability column and a comment.

1.9.3 Applicability of Experience Information

Table 1.9-23 lists NUREGs that have been included as references in the ESBWR DCD. Appendix 1C addresses applicability of US NRC Generic Letters and Bulletins.

1.9.4 COL information

1.9-1-A SRP Deviations (deleted)

1.9-2-A Experience Information (deleted)

1.9-3-A SRP and Regulatory Guide Applicability

COL applicant will address the applicability of SRPs and Regulatory Guides that refer to "BSP" or "COL" in the Comments column. (Tables 1.9-20 and 1.9-21)

1.9.5 References

1.9-1 Electric Power Research Institute, “Advanced Light Water Reactor Utility Requirements Document,” Revision 6, May 1997.

1.9-2 GE Nuclear Energy; “GE Nuclear Energy Quality Assurance Program Description,” NEDO-11209-04a, Class I (non-proprietary), Revision 8, March 31, 1989.

1.11 TECHNICAL RESOLUTIONS OF TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, NEW GENERIC SAFETY ISSUES AND CHERNOBYL ISSUES

Consistent with 10 CFR 52.47, this section provides technical resolutions of Unresolved Safety Issues (USIs) and New Generic Issues, medium and high priority Generic Safety Issues (GSIs) that are identified in Table II of Reference 1.11-1, which are technically relevant to the ESBWR.

1.11.1 Approach

Each item and/or issue in Table II of Reference 1.11-1 is addressed in Table 1.11-1. 10 CFR 52.47(a)(21)~~(iv)~~ requires the "Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues that are identified in the version of Reference 1.11-1 current on the date six months prior to application and that are technically relevant to the design," be included in a DCD. In accordance with 10 CFR 52.47(a)(21)~~(iv)~~, those issues that are not technically relevant to the ESBWR design are not necessarily addressed in detail.

Table 1.11-1 uses a series of notes, which are consistent with the 10 CFR 52.47(a)(1)(iv) requirement and the Legend and Notes of Table II of Reference 1.11-1, to disposition many of the items/issues.

- For issues that are not applicable to the 10 CFR 52.47(a)(21)~~(iv)~~ requirement, Table 1.11-1 only provides notes explaining those conclusions.
- For issues specifically addressed elsewhere in Tier 2, Table 1.11-1 only provides cross-references to the applicable Tier 2 locations.
- For issues whose technical concerns are adequately addressed elsewhere in Tier 2, Table 1.11-1 only provides cross-references to the applicable Tier 2 locations.
- For issues whose technical concerns are only partially addressed elsewhere in Tier 2, Table 1.11-1 provides cross-references to the applicable Tier 2 locations and the additional information to provide their resolutions.

For issues whose technical concerns are not addressed elsewhere in Tier 2, Table 1.11-1 provides their technical resolutions.

1.11.2 COL Information

1.11-1-A Address Table 1.11-1 Items That Refer to Notes (2) and (7)

COL applicant will provide information to supplement the listings for all issues in Table 1.11-1 that refer to Notes (2) and (7). This includes items A-33, B-1, B-28, B-37 through B-43, C-16 and 184. (Table 1.11-1, Notes (2) and (7))

1.11.3 References

1.11-1 U.S. Nuclear Regulatory Commission, "A Prioritization of Generic Safety Issues," NUREG-0933 and its Supplements through Supplement 30, October 2006.

The configuration of the pressure suppression containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the containment atmosphere following an accident is thereby substantially reduced compared to the Reg. Guide 1.183 source terms. The Passive Containment Cooling System (PCCS) condensing function contributes to reduce many of the airborne fission products.

Containment leakage is limited to less than 0.4% of the weight in the containment free volume per day.

1B.3.2 Post-Accident Access of Areas and Systems

This section addresses any area that may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas that must be accessible after an accident are the control room and technical support center.

Areas requiring post-accident access also include consideration (in accordance with NUREG-0737, II.B.2) of the containment isolation reset control area, manual ECCS alignment area, motor control centers and radwaste control panels. However, the ESBWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room or are not applicable for the passive ECC systems. Areas requiring post-accident access that are normally areas of mild environment allowing unlimited access are not reviewed for access.

Systems specific to the ESBWR that may require post-accident access are those for long-term core cooling, fission product control and combustible gas monitoring, as well as the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1B.3.3 Post-Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10 CFR ~~50.34(a)~~52.47(a)(2)(iv) ~~guidelines~~limits.

Safety-related systems are required for scram and to achieve a safe shutdown condition. However, they are not necessarily needed to maintain safe shutdown. The systems identified in Section 1B.5 are the systems used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the manual external connections for the IC/PCCS and fuel pools makeup.

1B.4 DESIGN REVIEW BASES

1B.4.1 Radioactive Source Term and Dose Rates

The radioactive source term used is equivalent to the source terms recommended in Reg. Guide 1.183 and Standard Review Plan 15.0.1 with appropriate decay times. Depressurized coolant is assumed to contain no noble gas.

Dose rates for areas requiring continuous occupancy may be averaged over 30 days to achieve the desired <0.15 mSv/hr (15 mrem/hr).

Design dose rates for personnel in areas requiring post-accident access are such that the guidelines of General Design Criterion (GDC) 19 [i.e., <0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2] are not exceeded for the duration of the accident, based upon expected occupancy and protection.

1B.4.2 Accidents Used as the Basis for the Specified Radioactivity Release

The various accidents and associated potential for fuel rod failure are addressed in Sections 15.3 and 15.4. This chapter also provides the accident parameters. Of those accidents, only the design basis accident (DBA) LOCA is assumed to produce 100% failed fuel rods under NRC worst-case assumptions. The fuel handling accident is the only other DBA postulated as leading to failed fuel rods with the potential consequence of radioactivity releases comparable to the 10 CFR ~~50.34(a) guidelines~~52.47(a)(2)(iv) limits.

For the fuel handling accident, the reactor is either shutdown and cooled or is operating normally if the accident is in the spent fuel storage pool. The total activity released to the environment and the calculated exposures are provided in Subsection 15.4.1. The exposures are within the ~~guidelines~~limits of 10 CFR 52.47(a)(2)(iv)~~50.34(a)~~. Thus, recovery is possible well within the specified 100-day equipment qualification period defined in Appendix 3H. ECCS equipment is not affected by this accident. This accident is not considered further.

Although a DBA-LOCA would not actually uncover the core or lead to fuel damage (see Section 6.3), core wide fuel failure is assumed such that this accident produces the limiting conditions of interest for this design review. In this accident the reactor is depressurized and reactor water mixes with suppression pool water in the process of keeping the fuel covered and cooled.

1B.4.3 Availability of Offsite Power

The availability of offsite power is not influenced by plant accident conditions. Various 10 CFR 50 Appendix A General Design Criteria (i.e., GDCs 33, 34, 35, 38, 41 and 44) require that safety functions be capable of being performed from either onsite or offsite power sources while the other power source is assumed unavailable. Consequently, loss of offsite power is typically assumed as occurring coincident with the beginning of many accident sequences.

However, continued absence of offsite power for the accident duration is not realistic. While restoration of offsite power is not a necessary condition for maintaining core cooling, its availability permits operation of other plant systems that would not otherwise be permitted by emergency power restrictions (e.g., operation of the pneumatic air system, nonsafety-related HVAC systems and other systems useful to plant cleanup and recovery).

2. SITE CHARACTERISTICS

2.0 INTRODUCTION

This chapter defines the envelope of site-related parameters that the ESBWR Standard Plant is designed to accommodate. These parameters envelope most potential sites in the U.S. A list of the site envelope design parameters is given in Table 2.0-1.

Table 2.0-2 references the guidance in NUREG-0800 Standard Review Plan (SRP). Table 2.0-2 defines the limits imposed on the acceptance criteria in Section II of the various SRPs by (1) the envelope of site-related parameters that the ESBWR plant is designed to accommodate, and (2) the assumptions, both implicit and explicit, related to site parameters that were employed in the evaluation of the ESBWR design.

The requirements for site parameters for a standard design are contained in 10 CFR 52.47(a)(1)~~(iii)~~. A design certification applicant provides postulated site parameters for the design, and an analysis and evaluation of the design in terms of such parameters. The following demonstrate that the standard design meets the above criteria.

The specified site parameters are the top-level bounding site parameters useful in the selection of a suitable site for a facility referencing the ESBWR certified design. Because they were used in bounding evaluations of the certified design, they define the envelope of site parameters used for the design that must be considered for a site. When the site characteristics fall within the site parameter values, a facility built on the site is in conformance with the design certification. Appropriate values for site parameters have been selected that make the design suitable for many sites. All site parameters specified in Tier 1 have the same values as those presented in this chapter.

The analyses and evaluations of the design, considering the site parameters of Table 2.0-1, are contained in the various sections of this document. For example, the safe shutdown earthquake parameters are used in structural and piping analyses in various sections of Chapter 3, atmospheric dispersion parameters are used in radiological analyses throughout Chapter 15, and the elevation parameter is used in the flooding analyses in Section 3.4.

Site parameters are specified for the following parameters:

- Maximum Ground Water Level
- Maximum Flood (or Tsunami) Level
- Precipitation (for roof design)
- Ambient Design Temperature
- Extreme Wind
- Tornado (maximum speed, pressure drop, missile spectrum, etc.)
- Maximum Settlement Values for Seismic Category I Buildings
- Soil Properties (minimum static bearing capacity, minimum dynamic bearing capacity, minimum shear wave velocity, liquefaction potential, angle of internal friction)
- Seismology (SSE response spectra, using figures)

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

ESBWR structures, systems and components (SSCs) are categorized as safety-related (as defined in 10 CFR 50.2) or nonsafety-related. The safety-related SSCs are those relied upon to remain functional during and following design basis events to ensure:

- The integrity of the reactor coolant pressure boundary (RCPB);
- The capability to shut down the reactor and maintain it in a safe condition; or
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable ~~guidelines~~ exposure limits set forth in 10 CFR ~~50.34(a)(1)~~52.47(a)(2)(iv).

Safety-related SSCs conform to the quality assurance requirements of Appendix B to 10 CFR 50. Nonsafety-Related SSCs have quality assurance requirements applied commensurate with the importance of the item's function. The quality assurance program is described in Chapter 17.

The ESBWR complies with 10 CFR 50, Appendix A, General Design Criterion (GDC) 2, as the safety-related SSCs are designed to withstand the effects of earthquakes without loss of capability to perform their safety-related functions. Specific requirements for seismic design and quality group classifications are identified for these ESBWR items commensurate with their safety classification. Table 3.2-1 identifies these classifications for ESBWR SSCs.

3.2.1 Seismic Classification

The ESBWR meets the acceptance criteria of Standard Review Plan (SRP) 3.2.1 (Reference 3.2-1). Structures that must remain integral with systems and components (including their foundations and supports) that must remain functional or retain their pressure integrity in the event of a safe shutdown earthquake (SSE) are designated Seismic Category I. These include safety-related items and fuel storage racks.

The Seismic Category I structures, systems, and components are designed to withstand the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads without loss of function or pressure integrity. The seismic classifications indicated in Table 3.2-1 are consistent with the guidelines of Regulatory Guide (RG) 1.29 (Reference 3.2-2).

SSCs that perform no safety-related function, but whose structural failure or interaction could degrade the functioning of a Seismic Category I item to an unacceptable level of safety or could result in incapacitating injury to occupants of the main control room, are designated Seismic Category II. These items are designed to structurally withstand the effects of an SSE. Seismic Category II SSCs that are also classified as RTNSS Criterion B in Tables 19A-2 and 19A-3 are required to remain functional following a seismic event.

Structures, systems, and components that are not categorized as Seismic Category I or II are designated Seismic Category NS.

Seismic Category NS structures and equipment are designed for seismic requirements in accordance with the International Building Code (IBC) Reference 3.2-6. The building structures are classified as Category IV (Power Generating Stations) with an Occupancy Importance Factor of 1.5. Either of the methods permitted by the IBC, simplified analysis or dynamic analysis, is acceptable for determination of seismic loads on Seismic Category NS structures and equipment~~,~~. Refer to Subsection 19A.8.3 for seismic design requirements applicable to Seismic Category NS SSCs ~~including those~~ designated as ~~Regulatory Treatment of Non-Safety Systems (~~RTNSS~~)~~, and~~.~~ ~~Refer~~ to ~~Appendix 19A,~~ Table 19A-~~1~~2 for a list of RTNSS SSCs.

3.2.2 System Quality Group Classification

The ESBWR meets the acceptance criteria of SRP 3.2.2 (Reference 3.2-3). NRC RG 1.26 (Reference 3.2-4) describes a quality group classification method for fluid systems and relates it to industry codes. Items are classified by Quality Group A, B, C or D, as indicated in Table 3.2-3. Table 3.2-3 tabulates the design and fabrication requirements for each quality group, as defined in RG 1.26.

Table 3.2-1 shows the quality group classifications for ESBWR components. Although not within the scope of RG 1.26, the containment boundaries that are within the scope of ASME Code, Section III, are assigned quality group classifications in accordance with Table 3.2-2.

Due to the use of many passive safety-related systems in ESBWR, the definitions of the quality groups provided in RG 1.26 can be somewhat misleading when trying to apply them directly to the ESBWR design. The following definitions in this section are consistent with the definitions in RG 1.26, but have been modified to more accurately describe their application to the ESBWR design.

3.2.2.1 Quality Group A

Quality Group A applies to pressure-retaining portions and supports of mechanical items that form part of the RCPB and whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability. These items are designed to meet the ASME Boiler and Pressure Vessel (B&PV) Code, Section III. Remaining portions of the RCPB are classified in accordance with Subsection 3.2.2.2.

3.2.2.2 Quality Group B

Quality Group B applies to pressure-retaining portions and supports of containment and other mechanical items, requirements for which are within the scope of ASME B&PV Code, Section III. These items are not assigned to Quality Group A and are relied upon to accomplish one or more of the following safety-related functions:

- Maintain the pressure integrity of RCPB items that are not Quality Group A.
- During or following design basis accidents whose consequences could result in potential offsite exposures comparable to the ~~guidelines~~limits of 10 CFR ~~50.34(a)(1)~~52.47(a)(2)(iv). These items include those that:
  - Maintain the pressure integrity of the containment, containment isolation, or extension of containment.
  - Maintain the pressure integrity of items that are (1) exterior to the containment; (2) communicate with the RCPB or containment interior; and (3) are not isolated normally, cannot be automatically isolated, or are not isolated following a design basis accident or anticipated operation occurrence (transient).

3.5 MISSILE PROTECTION

The missile protection design basis for Seismic Category I structures, systems and components (SSCs) is described in this section. A tabulation of SSCs (both inside and outside containment), their location, seismic category, and quality group classification is given in Table 3.2-1. General arrangement drawings showing locations of the SSCs are presented in Section 1.2.

Missiles considered are those that could result from a plant-related failure or incident including failures within and outside of containment, environmental-generated missiles and site-proximity missiles. The structures, shields, and barriers that are designed to withstand missile effects, the possible missile loadings, and the procedures to which each barrier is designed to resist missile impact are described in detail.

3.5.1 Missile Selection and Description

Components and equipment are designed to have a low potential for generation of missiles as a basic safety precaution. In general, the design that results in reduction of missile-generation potential promotes the long life and usability of a component and is well within permissible limits of accepted codes and standards.

Seismic Category I structures are analyzed and designed to be protected against a wide spectrum of missiles. For example, failure of certain rotating or pressurized components of equipment is considered to be of sufficiently high probability and to presumably lead to generation of missiles. However, the generation of missiles from other equipment is considered to be of low enough probability and is dismissed from further consideration. Tornado-generated missiles and missiles resulting from activities particular to the site are also discussed in this section. The missile protection criteria to which the plant has been designed consider Criterion 4 of 10 CFR 50 Appendix A, General Design Criteria for Nuclear Power Plants.

Potential missiles that have been identified are listed and discussed in later subsections.

After a potential missile has been identified, its statistical significance is determined. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or violation of the ~~guidelines~~limits of 10 CFR 52.47(a)(2)(iv)~~100 (and 10 CFR 50.34(a))~~.

The examination of potential missiles and their consequences is done in the following manner to determine statistically significant missiles:

- If the probability of occurrence of the missile, P_1, is determined to be less than 10^{-7} per year, the missile is dismissed from further consideration because at that likelihood it is considered not to be a statistically significant risk.
- If P_1 is found to be greater than 10^{-7} per year, it is examined for its probability of impacting a design target P_2.
- If the product of P_1 and P_2 is less than 10^{-7} per year, the missile is dismissed from further consideration.
- If the product of P_1 and P_2 is greater than 10^{-7} per year, the missile is examined for its damage probability P_3. If the combined probability (i.e., $P_1 \times P_2 \times P_3 = P_4$) is less than 10^{-7} per year, the missile is dismissed.

- Finally, measures are taken to design acceptable protection against missiles with P_4 greater than 10^{-7} per year to reduce P_1, P_2, and/or P_3, so that P_4 is less than 10^{-7} per year.

Many practices used in the fabrication, construction and inspection of equipment as well as conservative design criteria result in very robust components that are inherently missile resistant. These practices are used in making the design missile-proof.

Protection of SSCs is afforded by one or more of the following practices:

- Location of the system or component in an individual missile-proof structure;
- Physical separation of redundant systems or components of the system from the missile trajectory path or calculated range;
- Provision of localized protection shields or barriers for systems or components;
- Design of the particular structure or component to withstand the impact of the most damaging missile;
- Provision of design features on the potential missile source to prevent missile generation; and/or
- Orientation of the potential missile source to prevent unacceptable consequences caused by missile generation.

The following criteria are adopted to provide an acceptable design basis for the plant's capability to withstand the statistically significant missiles postulated inside the Reactor Building:

- No loss of containment function as a result of missiles generated internal to containment.
- Reasonable assurance that a safe plant shutdown condition can be achieved and maintained.
- Off-site exposure within the 10 CFR ~~50.34(a)~~52.47(a)(2)(iv) limits~~guidelines~~ for those potential missile damage events resulting in radiation activity release.
- The failure of nonsafety-related equipment, components, or structures whose failure could result in a missile, do not cause failure of more than one division of safety-related equipment.
- No high energy lines are located near Off-Gas Charcoal Bed Adsorbers (located in the Turbine Building).

The systems requiring protection are as follows:

(1) Reactor coolant pressure boundary;

(2) Automatic Depressurization System relief valves;

(3) Passive Containment Cooling System;

(4) Isolation Condenser;

(5) Gravity Driven Cooling System;

(6) Control Rod Drive scram system (hydraulic and electrical);

(7) Reactor Protection System;

3.5.1.6 Aircraft Hazards

The probability of aircraft hazards impacting the ESBWR Standard Plant and causing consequences greater than 10 CFR 52.47(a)(2)(iv)~~100 (and 10 CFR 50.34(a))~~ exposure ~~guidelines~~limits is ≤ about 10^{-7} per year.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

This subsection discusses the SSCs to be protected from externally generated missiles and includes all safety-related SSCs on the plant site that are provided to support the reactor facility.

The sources of external missiles, which could affect the safety of the plant, are identified in Subsection 3.5.1. Certain items in the plant are required to safely shut down the reactor and maintain it in a safe condition assuming an additional single failure. These items, whether they are structures, systems or components, must all be protected from externally generated missiles.

These items are the safety-related items listed in Table 3.2-1; appropriate safety classes and equipment locations are given in this table. All of the safety-related systems listed are located in buildings that are designed as tornado resistant. Because the tornado missiles are the design basis missiles, the SSCs listed are adequately protected. Provisions are made to protect the Off-Gas Charcoal Bed Adsorbers, Seismic Category I portions of the Fire Protection System (FPS) and components of Fuel and Auxiliary Pool Cooling System that transport makeup water to Spent Fuel Pool and Isolation Condenser/Passive Containment Cooling Pools from the FPS against tornado missiles.

3.5.3 Barrier Design Procedures

The procedures by which structures and barriers are designed to resist the missiles described in Subsection 3.5.1 are presented in this section. The following procedures are in accordance with Subsection 3.5.3 of NUREG-0800 (Standard Review Plan) and ensure that the design of structures, shields, and barriers that must withstand the effects of environmental and natural phenomena meet the relevant requirements of GDC 2 and GDC 4.

3.5.3.1 Local Damage Prediction

The prediction of local damage in the impact area depends on the basic material of construction of the structure or barrier (i.e., concrete or steel). The corresponding procedures are presented separately.

3.5.3.1.1 Concrete Structures and Barriers

Sufficient thickness of concrete is provided to prevent perforation, spalling or scabbing of the barriers in the event of missile impact. The (modified) National Defense Research Committee formula (Reference 3.5-5) is applied analytically for missile penetration in concrete. To prevent perforation, ACI-349 Appendix C Section C.7 is used. The resulting thickness of concrete required to prevent perforation, spalling or scabbing is no less than that for Region I listed in Table 1 of SRP 3.5.3.

3.6.1.1 Design Bases

Criteria

Pipe break event protection conforms to 10 CFR 50 Appendix A, General Design Criterion 4, Environmental and Dynamic Effect Design Bases, as it relates to safety-related Structures, Systems and Components (SSCs) being designed to accommodate the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The design bases for this protection are in compliance with NRC BTP SPLB 3-1 (Formerly BTP ASB 3-1), and BTP 3-4 included in Subsections 3.6.1 and 3.6.2, respectively, of NUREG-0800 (Standard Review Plan). BTP 3-4 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Subsections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.

The design of the containment structure, component arrangement, pipe runs, pipe whip restraints, and compartmentalization are done in consonance with the acknowledgment of protection against dynamic effects associated with a pipe break event. Analytically sized and positioned pipe whip restraints are engineered to preclude damage based on the pipe break evaluation.

Objectives

Protection against pipe break event dynamic effects is provided to fulfill the following objectives:

- Assure that the reactor can be shut down safely and maintained in a safe shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits with Loss of Preferred Power (LOPP).
- Assure that containment integrity is maintained.
- Assure that the radiological doses of a postulated piping failure remain below the limits~~guidelines~~ of 10 CFR ~~50.34(a)~~52.47(a)(2)(iv).

Assumptions

The following assumptions are used to determine the protection requirements:

- Pipe break events may occur during normal plant conditions (i.e., reactor startup, operation at power, normal hot standby (Reference 3.6-1) or reactor cooldown to a cold shutdown conditions but excluding test modes).
- A pipe break event may occur simultaneously with a seismic event; however, a seismic event does not initiate a pipe break event. This applies to Seismic Category I and non-Seismic Category I piping (seismically analyzed).
- A Single Active Component Failure (SACF) is assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted below. A SACF is the malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, or electrical malfunction but not the loss of component structural integrity. The direct consequences

- Safety-related structures, systems, and components are not impaired so as to preclude safety-related functions. For any given postulated pipe break and consequent jet, those safety-related structures, systems, and components needed to safely shut down the plant are identified.
- Safety-related SSCs which are not necessary to safely shut down the plant for a given break, are not protected from the consequences of the fluid jet.
- Safe shutdown of the plant caused by postulated pipe ruptures within the RCPB is not aggravated by sequential failures of safety-related piping and the required emergency cooling system performance is maintained.
- Off-site doses comply with 10 CFR ~~50.34(a)~~52.47(a)(2)(iv).
- Postulated breaks resulting in jet impingement loads are assumed to occur in high-energy lines at 102% power operation of the plant.
- Through-wall leakage cracks are postulated in moderate-energy lines and are assumed to result in wetting and spraying of safety-related structures, systems, and components.
- Reflected jets are considered only when there is an obvious reflecting surface (such as a flat plate) which directs the jet onto safety-related equipment. Only the first reflection is considered in evaluating potential targets.
- Potential targets, or portion of a large target, which are close enough to the jet boundary of the model assumed such that with reasonable variations in the jet geometry or pipe movement parameters they could be impinged upon, are assumed to be impinged upon.

The analytical methods used to determine which targets could be impinged upon by a fluid jet and the corresponding jet impingement load include:

- The direction of the fluid jet is based on the arrested position of the pipe during steady-state blowdown.
- The impinging jet proceeds along a straight path.
- The total impingement force acting on any cross-sectional area of the jet is time and distance invariant with a total magnitude equivalent to the steady-state fluid blowdown force given in Subsection 3.6.2.2 and with jet characteristics shown in Figure 3.6-1.
- The jet impingement thrust force on the target is calculated for certain cases according to ~~Appendix D of~~ ANSI/ANS 58.2. For cases where a magnitude of a jet thrust force are only important for pipe reaction load, a detailed jet evaluation is not necessary. Simple load calculation may be applicable for a pipe break where absence of an energy reservoir upstream or downstream of the break will not result in a continuous jet blowdown. A detailed jet impingement analysis is not significant for smaller pipe breaks if a design or an analysis of larger size pipe break loads envelop these pipe break jet impingement loads affecting the same target. The jet shield, barrier and an enclosure designed for a large pipe breaks may bound smaller pipe jet impingement and pipe whip effects and loads calculated based on simplified method may be sufficient to justify these cases.
- On a case by case basis a quantative analysis approach of the dynamic jet force determination is necessary where jet characteristics such as, jet nonlinearity, turbulence,

3.7 SEISMIC DESIGN

For seismic design purposes, all structures, systems, and components of the ESBWR standard plant are classified into Seismic Category I, Seismic Category II, or ~~non-seismic~~Seismic Category NS in accordance with the requirements to withstand the effects of the Safe Shutdown Earthquake (SSE) as defined in Section 3.2. For those Seismic Category I and Seismic Category II structures, systems and components (SSCs) in the Reactor Building (RB) complex, the effects of other dynamic loads caused by Reactor Building vibration (RBV) caused by suppression pool dynamics are also considered in the design. Although this section addresses seismic aspects of design and analysis in accordance with Regulatory Guide (RG) 1.70, the methods of this section are also applicable to RBV dynamic loadings, unless noted otherwise.

The SSE is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is the earthquake that produces the maximum vibratory ground motion for which Seismic Category I SSCs are designed to remain functional and within applicable stress, strain, and deformation limits. These systems and components are those necessary to ensure the following:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe condition; or
- The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable ~~guidelines~~ exposure~~s~~ limits set forth in 10 CFR 100 (10 CFR 52.47(a)(2)(iv)~~50.34(a)~~).

ESBWR response to an earthquake up to SSE may achieve shutdown of the reactor and maintenance of it in a safe condition using the Automatic Depressurization System and Gravity Driven Cooling System as described in the Probabilistic Risk Assessment. In this case, depressurization is accomplished in part with Depressurization Valves that remain open in order for the Gravity Driven Cooling System and the Passive Containment Cooling System to perform their safety functions.

Seismic Category II includes all plant SSCs which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system or component to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room. Thus, this category includes the SSCs whose structural integrity, not their operational performance, is required. Seismic Category II SSCs that are also classified as RTNSS Criterion B in Tables 19A-2 and 19A-3 are required to remain functional following a seismic event. The methods of seismic analysis and design acceptance criteria for Seismic Category II SSCs are the same as Seismic Category I; however, the procurement, fabrication and construction requirements for Seismic Category II SSCs are in accordance with industry practices. Seismic Category II items are those corresponding to position C.2 of RG 1.29.

The Operating Basis Earthquake (OBE) is a design requirement. For the ESBWR OBE ground motion is chosen to be one-third of the SSE ground motion. Therefore, no explicit response or design analysis is required to show that OBE design requirements are met. This is consistent with Appendix S to 10 CFR 50. The effects of low-level earthquakes (lesser magnitude than the

- One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
- One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Certain systems require alternative containment isolation arrangements that are an exception to the above requirements. These exceptions are listed in Table 1.9-6 and are qualified on a case-by-case basis.

- GDC 57 as it relates to lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere being provided with at least one locked closed, remote-manual, or automatic isolation valve outside containment. This valve is to be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.
- Appendix K to 10 CFR 50 as it relates to the determination of the extent of fuel failure (source term) used in the radiological calculations.

6.2.4.1 Design Bases

Safety Design Bases

- Containment isolation valves provide the necessary isolation of the containment in the event of accidents or other conditions and prevent the unfiltered release of containment contents that cannot be permitted by 10 CFR ~~50.34(a)(1)~~52.47(a)(2)(iv) limits. Leak-tightness of the valves shall be verified by Type C test.
- Capability for rapid closure or isolation of pipes or ducts that penetrate the containment is performed by means or devices that provide a containment barrier to limit leakage within permissible limits.
- The design of isolation valves for lines penetrating the containment follows the requirements of General Design Criteria 54 through 57. Exemptions from these GDCs are listed in Table 1.9-6.
- Isolation valves for instrument lines that penetrate the DW/containment conform to the requirements of RG 1.11.
- Isolation valves, actuators and controls are protected against loss of their safety-related function from missiles and postulated effects of high and moderate energy line ruptures.
- Design of the containment isolation valves and associated piping and penetrations meets the requirements for Seismic Category I components.
- Containment isolation valves and associated piping meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1 or 2, in accordance with their quality group classification.

include remote manual isolation valves or check valves. In addition, the SLC System and ICS are ESF systems that have fluid paths through containment penetrations. The SLC penetrations are not automatically isolated and do not contain remote manual isolation valves. Instead, the SLC penetrations are isolated if necessary by process-actuated check valves, but only after the SLC flow into the reactor pressure vessel/containment has ceased following an accident. The ICS penetrations listed in Tables 6.2-23 through 6.2-30 consist of various system process lines, all of which may be open or required to be opened following an accient in order to perform the required ESF function. The ICS penetration flow paths contain remote manual isolation valves, process-actuated flow control valves, or automatic isolation valves that only close for the applicable ICS train if leakage outside of containment is detected through IC/Passive Containment Cooling (PCC) pool high radiation or IC lines high flow.

6.2.4.2.1 Containment Isolation Valve Closure Times

Containment isolation valve closure times are established by determining the isolation requirements necessary to keep radiological effects from exceeding guidelines in 10 CFR 52.47. For system lines, which can provide an open path from the containment to the environment, a discussion of valve closure time bases is provided in Chapter 15. However the design values of closure times for power-operated valves is more conservative than the above requirement. For valves above 80 mm (3 inches) up to and including 300 mm (12 inches) in diameter, the closure time is at least within a time determined by dividing the nominal valve diameter by 300 mm (12 inches) per minute. Valves 80 mm (3 inches) and less generally close within 15 seconds. All valves larger than 300 mm (12 inches) in diameter close within 60 seconds unless an accident radiation dose calculation is performed to show that the longer closure time does not result in a significant increase in offsite dose.

6.2.4.2.2 Instrument Lines Penetrating Containment

Sensing instrument lines penetrating the containment follow all the recommendations of RG 1.11, as follows:

- Each line includes a 6 mm (¼ inch) diameter orifice such that in the event of a piping or component failure, leakage is reduced to the maximum extent practical consistent with other safety requirements. The rate of coolant loss is within the makeup capability, the integrity and functional performance of secondary containment and associated safety systems is maintained and the potential offsite exposure is substantially below the ~~guidelines~~limits of 10 CFR ~~100~~52.47(a)(2)(iv).
- Each line is provided with a self-actuated excess flow check valve located outside containment, as close as practical to the containment. These check valves are designed to remain open as long as the flow through the instrument lines is consistent with normal plant operation; however, if the flow rate is increased to a value representative of a loss of piping integrity outside containment, the valves close. These valves reopen automatically when the pressure in the instrument line is reduced.
- The instrument lines are designated as Quality Group B up to and including the isolation valve, located and protected to minimize the likelihood of damage, protected or separated to prevent failure of one line from affecting the others, accessible for inspection and not so restrictive that the response time of the connected instrumentation is affected.

from that point forward in the accident, the amount of total hydrogen never exceeds this 72 hour maximum due to the operation of the PARs. The number and size of PARs specified will provide the minimum safety factor of two for each containment compartment (Drywell and Wetwell). There will be a minimum capacity of the equivalent of one full size PAR unit specified for each containment compartment, however due to other design considerations, more, smaller capacity units (with equivalent total capacity) will be specified. The nominal hydrogen depletion rates for the full size PAR will be a minimum of 0.8 kg/h. The half, quarter, and eighth size PARs have nominal depletion rates as a direct ratio to the full size PAR. ~~and~~ Additionally PARs are sited to incorporate items such as protection from jet impingement, protection from containment spray and cooling fan discharge, protection from flooding and pool swell, discharged exhaust impacts and accessibility for testing. The PARs units are placed in azimuthally diverse locations which support adequate containment atmosphere mixing.

6.2.5.1.1 Containment Purging Under Accident Conditions

In accordance with 10 CFR 50.34(f)(2)(xv), (NUREG-0933 Item II.E.4.4), the capability for containment purging/venting is designed to minimize the purging time consistent with As Low As Reasonably Achievable (ALARA) principles for occupational exposure. The piping, valves and controls in the Containment Inerting System can be used to control containment pressure (that is, purge the containment), and can reliably be isolated under accident conditions.

The Containment Inerting System (CIS) is used to establish and maintain an inert atmosphere within the containment during all plant operating modes except during plant shutdown for refueling or maintenance and during limited periods of time to permit access for inspection and maintenance during reactor low power operation. The system is designed to permit deinerting the containment for safe operator access and minimizing personnel exposure.

6.2.5.2 Containment Inerting System

The objective of the CIS is to preclude combustion of hydrogen and prevent damage to essential equipment and structures by providing an inerted containment environment. This is the method of combustible gas control for the ESBWR, as required by 10 CFR 50.44.

6.2.5.2.1 Design Bases

Safety (10 CFR 50.2) Design Bases

The CIS does not perform any safety-related function. Therefore, the CIS has no safety design bases other than provision for safety-related containment penetrations and isolation valves, as described in Subsection 6.2.4.

Power Generation Design Bases

- The CIS is designed to establish an inert atmosphere (i.e., less than 4% oxygen by volume) throughout the containment in less than 4 hours and less than 2% oxygen by volume in the next 8 hours following an outage.
- The CIS is designed to maintain the containment oxygen concentration below the maximum permissible limit (~~5~~4%) during normal~~, abnormal, and accident conditions~~ power operations to assure an inert atmosphere.

- Conformance: The N-DCIS design conforms to these requirements. See Table 3.11-1 (Electrical and Mechanical Equipment for Environmental Qualification).

10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants:

- Conformance: The design has ATWS mitigation functions, as described in Section 7.8.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety-Related Issues:

- Conformance: The N-DCIS is nonsafety-related. Resolution of unresolved and generic safety issues is discussed in Section 1.11.

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

- Conformance: Inspection, Test, Analyses, and Acceptance Criteria (ITAAC) for the N-DCIS are identified in Tier 1.

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

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

10 CFR 52.47, Level of Detail:

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

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

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

7.1.5.3.2 General Design Criteria

GDC 1, 2, 4, 12, 13, 19, ~~and~~ 24, 25, 26, 27, 28, 29, 33, 38, 41, 42, 43, and 64:

- Conformance: The N-DCIS design conforms to these GDCs. Refer to Subsections 3.1.2 and 3.1.3 for a general discussion of each GDC.

7.1.5.3.3 Staff Requirements Memorandum

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

- Conformance: SRM on SECY 93-087, II.Q, states that if a postulated common mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

- Conformance: Inspection, Test, Analyses, and Acceptance Criteria (ITAAC) for the N-DCIS are identified in Tier 1.

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

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

7.1.5.3.2 General Design Criteria

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

- Conformance: The N-DCIS design conforms to these GDCs. Refer to Subsections 3.1.2 and 3.1.3 for a general discussion of each GDC.

7.1.5.3.3 Staff Requirements Memorandum

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

- Conformance: SRM on SECY 93-087, II.Q, states that if a postulated common mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

  The N-DCIS provides diverse functionality via the DPS and associated interface systems. The nonsafety-related portions of the systems that conform to this guidance are~~,~~ further discussed in Section 7.8 and in Reference 7.1-4.

SRM, SECY-93-087, Item II.T, Control Room Alarm Reliability:

- Conformance: The N-DCIS AMS follows guidance in the above document for redundancy, independence, and separation so that the "alarm system" is considered redundant, has its own redundant processors and uses signals from distributed and redundant controllers. Alarm points are sent through a dual network to redundant processors that have dual power feeds. The alarm processors are dedicated, redundant, and conservatively sized. The alarms can be displayed on multiple independent VDUs, each with dual power supplies. Alarms are driven by redundant data links to the AMS. The alarm processors are redundant. There is one horn and one voice speaker. Test buttons test the horn and the lights.

7.1.5.3.4 Regulatory Guides

RG 1.151, Instrument Sensing Lines:

- Conformance: RG 1.151 is not applicable to the N-DCIS. The N-DCIS receives signals from sensors in various systems in the plant that are from instrument sensing lines from

7.1.6.1 Code of Federal Regulations

10 CFR 50.55a(a)(1), Quality Standards for Systems Important to Safety, and10 CFR 50.55a(h) Protection and Safety Systems, compliance with IEEE Std. 603:

- Conformance: The Q-DCIS design complies with the above requirements.

10 CFR 50.34(f), Conformance to Three Mile Island (TMI) Action Plan Requirements:

- The response to TMI related matters is generically addressed in Appendix 1A. TMI action plan requirements are identified for the systems in Table 7.1-1. The applicable systems are designed to conform. However, because of the design features, several of these requirements are not applicable. These are identified as follows:
  - II.K.3.13 – HPCI and RCIC Initiation Levels,
  - II.K.3.15 - Isolation of HPCI and RCIC (Turbine Driven),
  - II.K.3.21 - Automatic Restart of LPCS and LPCI, and
  - II.K.3.22 - RCIC Automatic Switchover of Suction Supply.

The TMI action items applicable to the I&C systems are:

- 10 CFR 50.34(f)(2)(iv) [I.D.2], Safety parameter display system, (see Subsection 7.1.5.2.4.1),
- 10 CFR 50.34(f)(2)(v) [I.D.3], Bypass and Inoperable Status Indication,
- 10 CFR 50.34(f)(2)(xvii) [II.F.1], Accident Monitoring Instrumentation,
- 10 CFR 50.34(f)(2)(xviii) [II.F.2], Inadequate Core Cooling Instrumentation,
- 10 CFR 50.34(f)(2)(xiv) [II.E.4.2], Containment Isolation Systems,
- 10 CFR 50.34(f)(2)(xix) [II.F.3], Instruments for Monitoring Plant Conditions Following Core Damage,
- 10 CFR 50.34(f)(2)(xxiii) [II.K.2.10], Anticipatory Reactor Trip,
- 10 CFR 50.34(f)(2)(xxiv) [II.K.3.23], Central Reactor Vessel Water Level Recording,

10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants:

- Conformance: The design has ATWS mitigation functions, as described in Section 7.8.

10 CFR 52.47(a)~~(1)(iv),~~ (21)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)~~(b)(1), 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)~~(25), 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 within the DCD conforms to this regulation.

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

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

10 CFR 50.49, Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants

- Conformance: The Q-DCIS systems are designed to meet the equipment qualification requirements set forth in 10 CFR 50.49. Details are discussed in Section 3.11

7.1.6.2 General Design Criteria

In accordance with Table 7.1-1, the following GDC are addressed for the Q-DCIS:

GDC 1, 2, 4, 13, 19, 20, 21, 22, 23, 24, 25, and 29.

- Conformance: The Q-DCIS design complies with these GDC. Specific conformance of the I&C systems themselves is addressed in Sections 7.2 through 7.8.

7.1.6.3 Staff Requirements Memorandum

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

- Conformance: To minimize exposure to common-mode failures, the digital I&C systems are designed for high reliability, with the application of quality assurance requirements as specified in 10 CFR 50.55a(a)(1). Additionally, ~~T~~the digital I&C is designed ~~with~~ applying principles of defense-in-depth and diversity ~~for~~ defense against common mode failures. Section 7.8 includes the description of the diverse I&C systems that specifically addresses the requirements of this SRM.

SRM on SECY 93-087 II.T, Control Room Annunciator/Alarm Reliability:

- Conformance: The AMS follows guidance in the above document for redundancy, independence, and separation because the "alarm system" is considered redundant. Alarm points are sent through dual networks to redundant message processors on dual power supplies. The processors are dedicated to only doing alarm processing. The alarms are displayed on multiple independent VDUs that each have dual power supplies. The alarm tiles, or their equivalent, are driven by redundant datalinks (with dual power). There are redundant alarm processors. There are no alarms that require manually

Table 7.1-1

I&C Regulatory Requirements Applicability Matrix

	Q-DCIS																			N-DCIS			
	RTIF - NMS Platform							SSLC/ESF Platform										Independent Control Platform		Network Segments			
	RTIF						NMS																
Applicable Criteria Guidelines: SRP NUREG-0800, Section 7.1	RTIF	RPS	LD&IS (MSIV Only)[6]	CMS (includes SPTM)[6]	NBS[6]	CRD[6]	NMS	SSLC/ESF[3]	LD&IS (non-MSIV)[1][6]	PRMS	CMS[6]	NBS (includes ADS)[6]	GDCS	ICS	SLC[6]	CBVS[2]	CRD[5][6]	VBIF	ATWS/SLC[4][6]	GENE	PIP A/B	BOP	PCF
50.49	Refer to Table 3.11-1 (Electrical and Mechanical Equipment for Environmental Qualification)																						
50.62							X	X						X					X	X	X	X	
50.63	X	X	X			X		X	X				X	X		X							
52.47(a)(21)	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
52.47(b)(1)	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
52.47(a)(25)	NA																						
52.47	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
52.47(c)(2)	NA																						
GENERAL DESIGN CRITERIA																							
1	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
2	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

- Conformance: The reactor will trip in response to a Loss of All Feedwater Flow Event. This is an anticipatory trip actuated on a power generation bus loss event~~loss of power to two of the four main FW pumps~~. The reactor will also trip on a turbine trip only if an insufficient number of bypass valves opens within a prescribed time period.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(1)~~(vi)~~, ITAAC in Design Certification Applications:

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

10 CFR 52.47(a)~~(1)(vii)~~(25), 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 RPS within the DCD documents conforms to this requirement.

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

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

7.2.1.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 20, 21, 22, 23, 24, 25, and 29:

- Conformance: The RPS design conforms to these GDC.

7.2.1.3.3 Staff Requirements Memorandum

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

- Conformance: The Reactor Trip (Protection) System design conforms to Item II.Q of SECY-93-087 NRC Branch Technical Position (BTP HICB-19) by the implementation of an additional Diverse Instrumentation and Control System described in Section 7.8.

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

There are 64 LPRM assemblies uniformly distributed in the core. There are four LPRM detectors within each LPRM assembly, equally spaced from near the bottom of the active fuel region to near the top of the active fuel region (Figure 7.2-8). The 256 detectors are assigned to four divisions comprising the four APRM channels. Any single LPRM detector is assigned to one APRM division. Each set of 64 LPRM detector signals is assigned to one APRM channel with these signals averaged and normalized to form an APRM signal representing the average core power. Electrical and physical separation of the division is maintained and optimized to fulfill the safety-related system requirement.

With the four divisions redundancy requirements are met, because a scram signal still can be initiated with a postulated single failure of one APRM channel under allowable APRM bypass conditions.

Components used for the safety-related functions are qualified for the environments in which they are located.~~ Additional information on NMS equipment qualification is included in Reference 7.2-2.~~

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

- Conformance: The NMS design of bypass and inoperable status indication conforms to this requirement, consistent with conformance of the NMS design to RG 1.47. In addition, the design conforms to the requirements of control and protection system interaction as described in IEEE Std. 603, Sections 5.8 and 6.3.

  The 12 SRNM channels are divided into four divisions and are independently assigned to four bypass groups such that bypass of up to four SRNM channels at any one time is allowed while still providing the required monitoring and protection capability (with any three of the four divisions of safety-related power available).

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), 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 NMS within the DCD documents conforms to this requirement.

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

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

- Conformance: Safety-related systems are designed to 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 SPTM portion of CMS consists of four redundant divisions so failure of any single temperature element 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 SPTM function complies by providing automatic indication of bypassed and inoperable status.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), 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 SPTM function within the DCD documents conforms to this requirement.

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

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

7.2.3.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 20, 21, 22, 23, 24, 25, and 29:

- Conformance: The SPTM function complies with these GDC.

7.2.3.3.3 Staff Requirements Memorandum

Item II.Q, (Defense Against Common-Mode Failures in Digital Instrument and Control Systems):

- Conformance: The SPTM subsystem design conforms to item II.Q of SECY-93-087 (BTP HICB-19) by the implementation of diverse instrumentation and control, described in Section 7.8.

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) (l.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) (lI.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)~~(21), 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(b)~~(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)~~(25), 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)~~(c)(2)~~(i)~~, Innovative Means of Accomplishing Safety Functions:

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)~~(21), 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)~~(b)(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)~~(25), 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 GDCS within the DCD conforms to this requirement.

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

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

7.3.1.2.3.2 General Design Criteria

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

- Conformance: The GDCS design complies with these GDCs.

7.3.1.2.3.3 Staff Requirements Memorandum

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

- Conformance: The GDCS design conforms to these criteria by providing diverse I&C, as described in Section 7.8.

7.3.1.2.3.4 Regulatory Guides

RG 1.22, Periodic Testing of Protection System Function:

- Conformance: System logic is tested continually as described in Subsection 7.3.1.2.4. Components are tested periodically during refueling outages. The GDCS design complies with RG 1.22.

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

- Conformance: The GDCS design complies with RG 1.47. Automatic indication is provided in the MCR to inform the operator that the system is inoperable or a division is bypassed.

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

- 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)~~(21), 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(b)~~(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)~~(25), 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)~~(c)(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.

- Conformance: Separation and isolation is preserved both mechanically and electrically in accordance with IEEE 603 and RG 1.75. The CRHS consists of four redundantly designed divisions so failure of any instrument will not interfere with 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 CRHS design complies by providing automatic indication of bypassed and inoperable status.

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

- Conformance: The CRHS design complies with this requirement.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), 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 CRHS in the DCD conforms to this requirement.

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

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

7.3.4.3.2 General Design Criteria

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

- Conformance: The CRHS design complies with these GDCs.

7.3.4.3.3 Staff Requirements Memorandum

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

- Conformance: The CRHS and ESF designs conform to these criteria, as described in Subsection 7.8.2.2.

design adequacy is considered during detail equipment design by analysis of heat loads (per circuit module, per bay, and per module).

Table 7.1-1 identifies the SSLC/ESF 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.

7.3.5.3.1 Code of Federal Regulations

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

- Conformance: The SSLC/ESF design 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 RG 1.153 and 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 SSLC/ESF complies by providing automatic indication of bypassed and inoperable status ~~(IEEE Std. 603, Sections 5.8, 6.2, and 7.2)~~.

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)~~(21), 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)~~(b)(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)~~(25), 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)~~(c)(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 Reactor Trip (Protection) System and ESF designs conform to Item II.Q of SRM on SECY-93-087 (BTP HICB-19) in conjunction with the implementation of the DPS, described in Section 7.8.

7.3.5.3.4 Regulatory Guides

RG 1.22, Safety Guide 22 Periodic Testing of Protection System Actuation Functions:

- Conformance: The SSLC/ESF design complies with the guidance of RG 1.22.

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

- Conformance: The SSLC/ESF provides bypass capability and status that complies with RG 1.47.

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

- Conformance: The SSLC/ESF design complies with the guidance of RG 1.53, IEEE Std. 603, Section 5.1 and IEEE Std. 379.

RG 1.62, Manual Initiation of Protective Actions:

- Conformance: The SSLC/ESF design complies with the guidance of RG 1.62. Signals for manual initiation of protective actions are hardwired to the SSLC/ESF equipment.

RG 1.75, Physical Independence of Electric Systems:

- Conformance: The SSLC/ESF design conforms to RG 1.75 as described in Subsections 8.3.1.3 and 8.3.1.4.

7.3.6.3 Safety Evaluation

Section 6.2 evaluates the VB isolation function and shows that for the entire range of nuclear process system pipe break sizes, the opening of a single VB ensures containment structure functional integrity.

Table 7.1-1 identifies the VB isolation function 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.

7.3.6.3.1 Code of Federal Regulations

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

- Conformance: The VB isolation function design complies with these standards.

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

- Conformance: Safety-related systems are in conformance with RG 1.153 and IEEE Std. 603. Separation and isolation is preserved both mechanically and electrically in accordance with IEEE Std. 603, Section 5.6 and RG 1.75. The VB isolation function is divisionalized and designed with redundancy so failure of any instrument will not prevent 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 VB isolation function 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 VB isolation function design complies with this requirement.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), 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 design of the VB and VB isolation function within the DCD complies with this requirement.

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

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

7.3.6.3.2 General Design Criteria

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

- Conformance: The VB isolation function design complies with these GDCs.

7.3.6.3.3 Staff Requirements Memorandum

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

- Conformance: The VB isolation function design complies with these criteria through demonstration that no postulated common-mode failure of the control system could disable the VB isolation function. The discrete logic and solid state controls used in this design are not subject to the vulnerabilities described by SECY-93-087, Item II.Q.

7.3.6.3.4 Regulatory Guides

RG 1.22, Periodic Testing of Protection System Function:

- Conformance: The VB isolation function design conforms to RG 1.22. System logic and components are tested periodically during refueling outages.

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

- Conformance: The VB isolation function design conforms to RG 1.47. Automatic indication is provided in the MCR to inform the operator that the system is inoperable or a division is bypassed.

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

- Conformance: The VB isolation function design conforms to RG 1.53, IEEE Std. 603, Section 5.1, and IEEE Std. 379.

RG 1.62, Manual Initiation of Protective Actions:

- Conformance: The VB isolation function design complies with RG 1.62. Each division 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.

RG 1.75, Physical Independence of Electric Systems:

- The VB isolation function design conforms to RG 1.75 as described in Subsections 8.3.1.3 and 8.3.1.4.

- Conformance: The SLC system design conforms to IEEE Std. 603. Separation and isolation is maintained both mechanically and electrically in accordance with IEEE Std. 603 and Regulatory Guide (RG) 1.75. The SLC is designed so a single failure does not interfere with system operation. Electrical separation is maintained between safety-related divisions, and between safety-related and nonsafety-related portions.

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

- Conformance: The SLC system design conforms to this requirement.

10 CFR 50.62, Requirements for Reduction of Risk from ATWS Events for light-water Cooled Nuclear Power Plants:

- Conformance: The SLC is automatically initiated, and is designed to perform its mitigation function reliably.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

- Conformance: The SLC system design complies with this criterion. Reference Section 1.11 for resolution of unresolved and generic safety issues.

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

- Conformance: ITAAC are provided for I&C Equipment in Tier 1.

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

- Conformance: There are no interface requirements for the SLC system.

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

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

10 CFR 52.47~~(b)~~(c(2)~~(i)~~, Innovative Means of Accomplishing Safety Function:

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

7.4.1.3.2 General Design Criteria.

In accordance with Table 7.1-1, the following General Design Criteria (GDC) are addressed for the SLC system:

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

- Conformance: The SLC system design conforms to these GDC.

provides 50% capacity for residual heat removal, and each RCCW and plant service water train provides approximately 50% capacity for shutdown cooling. The RWCU/SDC, RCCW, and plant service water systems, in conjunction with the ICS, can bring the plant to cold shutdown within 36 hours assuming the most restrictive single active failure.

In the event that one CRD train fails or is out of service for maintenance, the remaining pump can provide sufficient makeup to maintain RPV water level during reactor cooldown.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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(b)~~(a)~~(1)~~(vi)~~, ITAAC in Design Certification Applications:

- Conformance: ITAAC for the RSS are identified in Tier 1.

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

- Conformance: There are no interface requirements for RSS.

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

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

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

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

7.4.2.3.2 General Design Criteria

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

- Conformance: The RSS design conforms to these GDC.

7.4.2.3.3 Regulatory Guides

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

- Conformance: The RSS design conforms to RG 1.53, IEEE Std. 603, Section 5.1, and IEEE Std. 379.

In addition, separation and isolation is preserved both mechanically and electrically in accordance with IEEE 603, Sections 5.6 and 6.3, and RG 1.75.

RG 1.75, Physical Independence of Electric Systems:

Density compensated system mass flow is measured in the process lines (by mid-vessel nozzles with venturi-type flow elements in each line) from the reactor bottom, located inside the containment. Flow elements also are provided in the Seismic Category I RWCU/SDC return lines to the feedwater lines and in the overboarding lines. The flow transmitters for all of these flow elements are arranged in a two-out-of-four logic configuration used to detect high RWCU/SDC differential mass flow due to a break outside the containment and to close the inboard and outboard containment isolation valves of the affected RWCU/SDC train. The containment isolation function on detection of RWCU/SDC high differential mass flow (due to a break outside the containment) is part of the LD&IS described in Subsection 7.3.3. See Figures 7.4-2a through 7.4-2e for the logic for detection of a RWCU/SDC pipe break outside containment.

Flow orifices are used for flow monitoring of demineralizer inlet flow and to open the demineralizer bypass control valve if the flow exceeds the demineralizer capacity.

7.4.3.3 Safety Evaluation

The RWCU/SDC system functions are nonsafety-related, with the exception of containment isolation functions and providing instruments to detect high differential mass flow following a RWCU/SDC break outside the containment. Refer to Subsection 6.2.4 for the containment isolation functions, and Subsection 7.3.3 for the containment isolation and leak detection functions performed by the LD&IS.

Table 7.1-1 identifies the RWCU/SDC system 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.

7.4.3.3.1 Code of Federal Regulations

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

- Conformance: The RWCU/SDC system design conforms to these requirements.

10 CFR 50.55a(h), Protection and Safety System Compliance with IEEE 603:

- Conformance: The RWCU/SDC system is nonsafety-related, 10CFR 50.55a(h) and IEEE Std. 603 are not applicable to this system.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

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

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

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

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

- Conformance: There are no interface requirements for the RWCU/SDC system.

7.4.4.3.1 Code of Federal Regulations

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

- Conformance: The ICS design conforms to these standards.

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

- Conformance: Separation and isolation is preserved both mechanically and electrically, in accordance with IEEE Std. 603, Section 5.6 and 6.3, and RG 1.75. The ICS is divisionalized and redundantly designed so 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 ICS design conforms to this requirement because it is an ECCS.

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

- Conformance: The ICS will initiate 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 feedwater pumps.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), Interface Requirements:

- Conformance: There are no interface requirements for ICS.

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

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

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

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

7.4.4.3.2 General Design Criteria

In accordance with the SRP for Section 7.4 and Table 7.1-1, the following GDC are addressed for the ICS:

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

- Conformance: The PAM instrumentation design complies with this requirement. The direct water-level instrument system provides for the detection of conditions indicative of inadequate core cooling (Refer to Table 1A-1 of Appendix 1A, Three Mile Island [TMI] Action Plan Items).

10 CFR 50.34(f)(2)(xix) [II.F.3], Instrumentation for Monitoring Plant Conditions Following Core Damage:

- Conformance: The PAM instrumentation design conforms to this requirement. The PAM instrumentation design conforms to RG 1.97.

10 CFR 50.34(f)(2)(xxiv) [II.K.3.23], Recording of Reactor Vessel Water Level:

- Conformance: The PAM instrumentation design conforms to this requirement. (Refer to Table 1.A-1 of Appendix 1A TMI Action Plan Items).

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(1)~~(vi)~~, Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) in Design Certification Applications:

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

10 CFR 52.47(a)~~(1) (vii)~~(25), 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 PAM instrumentation within the DCD conforms to this requirement.

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

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

7.5.1.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 24, and 64:

- Conformance: The PAM instrumentation design conforms to these GDC.

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

7.5.2.3.1 Code of Federal Regulations

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

- Conformance: The CMS design complies with these standards.

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

- Conformance: Separation and isolation is preserved both mechanically and electrically in accordance with IEEE Std. 603, Section 5.6 and RG 1.75. The CMS safety-related subsystems are divisionalized and are 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 CMS design complies by being able to provide automatic indication of bypassed and inoperable status.

10 CFR 50.34(f)(2)(viii)[II.B.3], Compatibility to Promptly Obtain and Analyze Containment Atmosphere Samples:

- Conformance: The CMS design complies with this requirement.

10 CFR 50.34(f)(2)(xxvii)[III.D.3.3], Monitoring of In-plant Airborne Radioactivity:

- Conformance: The CMS design complies with this requirement.

10 CFR 50.34(f)(2)(xvii)[II.F.1], Accident Monitoring Instrumentation:

- Conformance: The CMS design complies with this requirement.

10 CFR 50.34(f)(2)(xix)[II.F.3], Instrumentation for Monitoring Plant Conditions Following Core Damage:

- Conformance: The CMS design complies with this requirement.

10 CFR 50.44(c)(4), Combustible Gas Control For Nuclear Power Reactors, Monitoring:

- Conformance: The CMS design complies with this requirement.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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(b)~~(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)(25)~~(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 CMS within the DCD conforms to this requirement.

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

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

7.5.2.3.2 General Design Criteria

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

- Conformance: The CMS design complies with these GDC.

7.5.2.3.3 Staff Requirements Memorandum

SRM on SECY-93-087, Item II.T, Control Room Annunciator (Alarm) Reliability:

- Conformance: The CMS AMS meets the requirements of SECY-93-087, Item II.T for redundancy, independence, and separation in that the "alarm system" is considered redundant as follows:
  - Alarm points are sent via dual networks to redundant message processors using dual power supplies. The processors are dedicated to alarm processing.
  - The alarms are displayed on multiple independent Video Display Units (VDUs) (dual power supplies on each).
  - The alarms are driven by redundant datalinks to the AMS (dual power). There are redundant alarm processors.
  - There is one horn and one voice speaker. Test buttons are available to test the horn and all the lights.
  - There are no alarms requiring manually controlled actions for safety systems to accomplish their safety-related functions.

7.5.2.3.4 Regulatory Guides

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

- Conformance: The CMS design conforms to RG 1.47.

10 CFR 50.34(f)(2)(xix)[II.F.3], Instrumentation for Monitoring Plant Conditions Following Core Damage:

- Conformance: The PRMS design complies with this requirement.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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(b)~~(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)(25)~~(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 PRMS within the DCD documents conforms to this requirement.

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

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

7.5.3.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 24, and 64:

- Conformance: The PRMS design complies with these GDC.

7.5.3.3.3 Staff Requirements Memorandum

SRM on SECY-93-087, Item II.T, Control Room Annunciator (Alarm) Reliability:

- Conformance: The PRMS AMS design meets the requirements of SECY-93-087, Item II.T for redundancy, independence, and separation in that the “alarm system” is considered redundant as follows:
  - Alarm points are sent via dual networks to redundant message processors using dual power supplies. The processors are dedicated to alarm processing.
  - The alarms are displayed on multiple independent VDUs (dual power supplies on each).
  - The alarms are driven by redundant data links to the AMS (dual power). There are redundant alarm processors.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), Interface Requirements:

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

7.5.4.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 24, and 64:

- Conformance: The ARMS design complies with these GDC.

7.5.4.3.3 Regulatory Guides

RG 1.97 - Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants:

- Conformance: The ARMS area radiation level instrumentation conforms to RG 1.97.

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

- Conformance: The ARMS design conforms to RG 1.180.

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

- Conformance: The ARMS design conforms to RG 1.204.

7.5.4.3.4 Branch Technical Positions

BTP HICB-10, Guidance on Application of Regulatory Guide 1.97:

- Conformance: The ARMS design conforms to BTP HICB-10.

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

- Conformance: The level of detail provided for the ARMS complies with BTP HICB-16.

7.5.4.3.5 Three Mile Island Action Plan Requirements

In accordance with SRP 7.5 and with Table 7.1-1, 10 CFR 50.34(f)(2)(xvii) [II.F.1], 10 CFR 50.34(f)(2)(xix) [II.F.3], and 10 CFR 50.34(f)(2)(xxvii)[III.D.3.3] apply to the ARMS. The ARMS design complies with these requirements, as indicated above. TMI action plan requirements are addressed generically in Appendix 1A.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(1)~~(vi)~~, Inspections, Tests, Analyses and Acceptance Criteria (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)~~(25), 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 interlock logic~~systems~~ conforms to this criterion.

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

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

7.6.1.3.2 General Design Criteria

GDC 1, 2, 4, 13, 19, 24 and 25:

- Conformance: Because the HP/LP interlock logic~~system~~ does not involve reactivity control, GDC 25 is not applicable. The interlock logic~~system~~ design complies with the remaining GDC listed above.

7.6.1.3.3 Regulatory Guides

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

- Conformance: The HP/LP interlock logic~~system~~ does not have a bypass feature.

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

- Conformance: The HP/LP interlock logic~~system~~ is nonsafety-related. RG 1.53 is not applicable~~ to this system~~.

RG 1.75, Physical Independence of Electrical Systems:

- Conformance: The HP/LP interlock logic~~system~~ is nonsafety-related. The physical and electrical separations maintained between safety-related and nonsafety-related systems conform to RG 1.75 as described in Subsections 8.3.1.3 and 8.3.1.4.

- 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)~~(21), 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)~~(b)(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)~~(25), 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)~~(c)(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.

update process. All other setpoints are established prior to plant startup operations and only adjusted, if needed, as a result of plant startup testing results. It is anticipated that none or very few of the RC&IS setpoints (besides the continual ATLM rod block setpoint updates) require adjustment as a result of startup testing results.

7.7.2.2.7.11 Environmental Considerations

The RC&IS is not required for safety-related purposes, nor is it required to operate after a design basis accident. This system is required to operate in the normal plant environmental conditions at the locations of the RC&IS equipment, in the back-panel area of the MCR and in applicable areas of the RB.

7.7.2.3 Safety Evaluation

The circuitry described for the RC&IS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the RC&IS circuitry from affecting the scram circuitry. The scram circuitry is discussed in Subsection 7.2.1. Because the RC&IS directly controls movement of each control rod as an individual unit, a failure that results in inadvertent movement of a control rod affects only one control rod. The malfunctioning of any single control rod does not impair the effectiveness of a reactor scram. Therefore, no single failure in the RC&IS prevents a reactor scram. Repair, adjustment, or maintenance of the RC&IS components does not affect the scram circuitry.

Chapter 15 examines the various failure mode considerations for this system. The expected and abnormal transients and accident events analyzed in subsections 15.2.3.1, 15.2.3.2, 15.3.1, 15.3.7, 15.3.8, and 15.3.9 envelope the failure modes associated with RC&IS ~~components~~ digital controls.

Table 7.1-1 identifies the RC&IS 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.

7.7.2.3.1 Code of Federal Regulations

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

- Conformance: The RC&IS design conforms to these requirements.

10 CFR 50.62, Requirements for reduction of risk from Anticipated Transients Without Scram (ATWS) events for light-water-cooled nuclear power plants:

- Conformance: The ATWS mitigation functions are designed in accordance with the requirements of 10 CFR 50.62.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

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

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

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

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

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

7.7.2.3.2 General Design Criteria

GDC 1, 13, 19, 24, 28 and 29:

- Conformance: The RC&IS complies with these GDC.

7.7.2.3.3 Regulatory Guides

RG 1.180, Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in safety-related Instrumentation and Control Systems

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

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

- Conformance: The RC&IS design conforms to RG 1.204.

7.7.2.3.4 Branch Technical Positions

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 the RC&IS design conforms to BTP HICB-16.

BTP HICB-19, Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems:

- Conformance: The portions of RC&IS that provide interface support for DPS conform to BTP HICB-19.

7.7.2.3.5 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: The portions of RC&IS that provide interface support for DPS conform to Item II.Q of SECY-93-087.

7.7.2.4 Testing and Inspection Requirements

The RC&IS equipment is designed with consideration for online testing capabilities. The system can be maintained on line while repairs or replacement of hardware take place without causing any abnormal upset condition. The single-channel bypass capabilities support having continued

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

- Conformance: Unresolved and generic safety issues are discussed in Section 1.11.

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

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

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

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

7.7.3.3.2 General Design Criteria

GDC 1, 13, 19, and 24:

- Conformance: The FWCS design complies with these GDC.

7.7.3.3.3 Regulatory Guides

RG 1.151, Instrument Sensing Lines:

- Conformance: The FWCS receives signals from sensors on vessel instrument lines in the NBS. Refer to Subsection 7.7.1.3 for a discussion of the guidance of RG 1.151 in relation to the NBS.

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

- Conformance: The FWCS design conforms to RG 1.180.

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

- Conformance: The FWCS design conforms to RG 1.204.

7.7.3.3.4 Branch Technical Positions

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

- Conformance: The level of detail in this subsection conforms to BTP HICB-16.

BTP HICB-19, Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems:

- Conformance: The portions of FWCS that provide interface support for DPS conform to BTP HICB-19.

- Conformance: The PAS design conforms to these requirements.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

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

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

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

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

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

7.7.4.3.2 General Design Criteria

GDC 1, 13, 19, and 24:

- Conformance: The PAS design complies with these GDC.

7.7.4.3.3 Regulatory Guides

RG 1.180 – Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in safety-related Instrumentation and Control Systems

- Conformance: The PAS design conforms to RG 1.180.

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

- Conformance: The PAS design conforms to RG 1.204.

7.7.4.3.4 Branch Technical Positions

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 the PAS conforms to BTP HICB-16.

7.7.4.4 Testing and Inspection Requirements

The FTDC input and output communication interfaces function continuously during normal power operation. Abnormal functioning of these components can be detected during operation. In addition, the FTDC is equipped with self-test and on-line diagnostic capabilities for identifying and isolating failures of input/output signals, buses, power supplies, processors, and inter-processor communications. These on-line tests and diagnostics can be performed without interrupting the normal control operation of the PAS.

The pressure control function forces the TCVs to remain under pressure control supervision to provide automatic load following. This enables fast bypass opening for transient events that require fast reduction in turbine steam flow.

The steam bypass function controls reactor pressure by responding to the bypass flow demand signal. It modulates the regulating bypass valves, which are automatically operated. This control mode is assumed under the following conditions:

- During RPV heatup to rated pressure,
- While the turbine is brought up to speed and synchronized,
- During power operation when the reactor steam generation rate exceeds the turbine steam flow rate requirements,
- During plant load rejection and turbine/generator trips, and
- During cooldown of the nuclear reactor.

7.7.5.3 Safety Evaluation

The SB&PC System is classified as a primary power generation system. It is not safety-related, and is not required to operate during or after any DBAs. The system is required to operate in the normal plant environment and is required for the power production cycle. The SB&PC System equipment is located in both the MCR area of the CB and the turbine building (TB); and each SB&PC System component is subject to the environment of the applicable area. The SB&PC System FTDC panel and its components are designed to retain structural integrity during and after DBEs so that safety-related equipment in its area are able to perform their safety functions.

Chapter 15 examines the various failure mode considerations for this system. The expected and abnormal transients and accident events analyzed in subsections 15.2.5.1, 15.3.3, and 15.3.4, 15.3.5, and 15.3.6 envelope the failure modes associated with the SB&PC digital controls.

Table 7.1-1 identifies the nonsafety-related SB&PC System 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.

7.7.5.3.1 Code of Federal Regulations

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

- Conformance: The SB&PC design conforms to these requirements.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

- Conformance: The SB&PC System is nonsafety-related and conforms in that there are no unresolved issues for the SB&PC System. Resolution of unresolved and generic safety issues is discussed in Section 1.11.

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

- Conformance: Inspection, test, analyses, and acceptance criteria of the SB&PC System FTDC are identified in Tier 1.

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

7.7.6.2.2.4 Environmental Considerations

The MRBM is located in the MCR. It is physically and electrically isolated from the safety-related NMS subsystems. All interfaces with the safety-related NMS subsystems are through fiber optic isolators.

7.7.6.3 Safety Evaluation

Table 7.1-1 identifies the nonsafety-related control systems 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.

7.7.6.3.1 Code of Federal Regulations

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

- Conformance: The AFIP and MRBM subsystem designs conform to these requirements.

10 CFR 52.47(a)~~(1)(iv)~~(21), Resolution of Unresolved and Generic Safety Issues:

- Conformance: Unresolved and generic safety issues are discussed in Section 1.11.

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

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

10 CFR 52.47(a)~~(1)(vii)~~(25), Interface Requirements:

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

7.7.6.3.2 General Design Criteria

GDC 1, 13, 19, 24, and 28:

- Conformance: The AFIP and MRBM subsystem designs comply with these GDC.

7.7.6.3.3 Regulatory Guides

RG 1.180, Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in safety-related Instrumentation and Control Systems:

- Conformance: The AFIP and MRBM subsystem designs conform to RG 1.180.

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

- Conformance: The AFIP and MRBM subsystem designs conform to RG 1.204.

7.7.6.3.4 Branch Technical Positions

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

flexible interface design to accommodate either metal wire or fiber optic communication links. The MRBM instrument is provided with necessary operator interface functions meeting NMS man-machine interface requirements.

The MRBM includes basic logic such as continuous LPRM data collection, MRBM rod block algorithm calculation, MRBM setpoint comparison, and communication protocol with the N-DCIS. The MRBM subsystem is located within the nonsafety-related equipment rooms of the CB having acceptable environmental conditions and physical and electrical separation from the safety-related NMS instruments..

7.7.7 Containment Inerting System

7.7.7.1 System Design Bases

The CIS design bases are discussed in Subsection 6.2.5.2.1.

7.7.7.2 System Description

The CIS system description is discussed in Subsection 6.2.5.2.

7.7.7.3 Safety Evaluation

The CIS safety evaluation is discussed in Subsection 6.2.5.2.3.

Table 7.1-1 identifies the CIS 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.

7.7.7.3.1 Code of Federal Regulations

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

- Conformance: The CIS design conforms to these requirements.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(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)~~(25), Interface Requirements:

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

Table 7.1-1 identifies the diverse I&C 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.

7.8.3.1 Code of Federal Regulations

10 CFR 50.55a(a)(1), Quality Standards for Systems Important to Safety:0

- Conformance: The diverse I&C systems design conforms to these standards.

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

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

For the diverse I&C systems, the applicable requirements are from IEEE Std. 603, Section 5.6, 'Independence'. Q-DCIS inter-divisional and cross-platform signal transmission is performed via fiber optic cables. Signal transmission between the systems of the Q-DCIS and the nonsafety-related control systems, including the DPS, is performed via fiber optic cables. The safety-related fiber optic communication interface module (CIM) provides the required isolation.

The diverse I&C have electrical surge withstand capability and can withstand the electromagnetic interference, radio frequency, and electrostatic discharge conditions that exist at their locations in the plant.

The diverse I&C equipment withstands the room ambient temperature, humidity conditions, radiation levels, and seismic accelerations that exist at their locations at the times for which they are required to be operational or required to fail in a safe mode.

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

- Conformance: The diverse I&C systems conform to these requirements by providing automatic indication of bypassed and inoperable status.

10 CFR 50.62, Requirements for reduction of risk from ATWS events for light-water cooled nuclear power plants:

- Conformance: The ATWS mitigation functions described in Subsection 7.8.1.1 are designed in accordance with the requirements of 10 CFR 50.62.

10 CFR 52.47(a)~~(1)(iv)~~(21), 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)~~(b)(1)~~(vi)~~, ITAAC in Design Certification Applications:

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

10 CFR 52.47(a)~~(1)(vii)~~(25), 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 diverse I&C functions within the DCD conforms to this requirement.

7.8.3.2 General Design Criteria

General Design Criteria (GDC) 1, 2, 4, 13, 19, 20, 21, 22, 23, and 24:

- Conformance: The diverse I&C systems design conforms to these GDC.

The design of the diverse I&C systems does not compromise the ability of the RPS and SSLC/ESF actuation system to meet the requirements of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Section III, "Protection and Reactivity Control Systems."

7.8.3.3 Staff Requirements Memorandum

Item II.Q, (Defense Against Common-Mode Failures in Digital Instrument and Control Systems) of SECY-93-087 and SRM on SECY 93-087 (Policy, Technical, and Licensing Issues Pertaining to Evolutionary and ALWR Designs):

- Conformance: The SRM requirements applicable to the diverse I&C functions state that, "If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure as the safety system shall be required to perform either the same function as the safety system function that is vulnerable to common mode failure or a different function." It also states, "The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary functions under the associated event conditions." With respect to manual control and display functions, it states, "A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer systems."

  The implementation of the DPS and the ATWS mitigation features as described in Subsection 7.8.1, in conjunction with the RPS and ESF designs, conforms to the above SRM requirements.

- Conformance: The safety-related ATWS mitigation logic conforms to the guidance in HICB-18. This BTP is not applicable to the nonsafety-related DPS.

BTP HICB-19, Guidance for evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems:

- Conformance: Reference 7.8-1 details the echelons of defense used in the design that conforms to BTP HICB-19. This document also discusses the basis for selection of the DPS functions used as backups for the RPS and SSLC/ESF. A failure modes and effects analysis based on the Guidance in NUREG/CR-6303 (Reference 7.8-2) is performed to ensure the radiation guidelines from ~~10 CFR 100~~ 10 CFR 52.47(a)(2)(iv) are not exceeded in the event of a common mode failure of the RPS or SSLC/ESF software platform during the design basis events discussed in the Safety Analyses.

BTP HICB-21, Guidance on Digital System Real-Time Performance:

- Conformance: The safety-related ATWS mitigation logic conforms to the guidance in HICB-21. This BTP is not applicable to the nonsafety-related DPS.

7.8.4 Testing and Inspection Requirements

Periodic testing to verify proper operation of the ATWS/SLC logic is performed. Periodic testing to verify proper operation of the DPS logic is also performed.

7.8.5 Instrumentation and Control Requirements

The ATWS/SLC uses logic that is diverse from the RPS. Logic and controls for ATWS/SLC are located in divisional RTIF cabinets. Operating status is available to the operator in the MCR. Division of sensors bypass capability is provided for the ATWS/SLC logic. Communication with external interfaces is through isolation devices. Provisions are made to allow testing of the ATWS/SLC logic and maintenance of the ATWS/SLC equipment.

The DPS uses triple redundant microprocessor-based automatic actuation logic that is diverse from the RPS and SSLC/ESF automatic actuation logic.

The information available to the operator from the diverse I&C systems is described in Subsection 7.8.1.3.

7.8.6 COL Information

None.

7.8.7 References

7.8-1 GE-Hitachi Nuclear Energy, “ESBWR I&C Defense-In-Depth and Diversity Report”, NEDO-33251, Class I (Non-proprietary), Revision 2~~1, August 2007~~.

All sampling lines have the process isolation block valves located as close as practical to the process taps. These valves may be closed if sample line rupture occurs downstream of the valves.

9.3.2.3 Safety Evaluation

The operation of the PSS is not required for any of the following:

- Integrity of the reactor coolant pressure boundary;
- Capability to shut down the reactor and maintain it in a safe shutdown condition; and
- Ability to prevent or mitigate the consequences of accidents that can result in potential offsite exposures comparable to the ~~guideline~~ exposure~~s~~ limits of 10 CFR ~~100~~52.47(a)(2)(iv).

However, the system incorporates features that improve operator safety. The sampling stations are closed systems and the grab samples taken at the sampling stations have a chemical fume hood to preclude the exposure of operating personnel to contamination hazards. A constant air velocity is maintained through the working face of the hoods to ensure that airborne contamination does not escape to the room under operating conditions.

In the event of a loss of cooling water to a sample cooler or sample flow in excess of sample cooler capacity, the sampling system valves are interlocked to prevent high-temperature water flow through the lines.

Safety/relief valves, vented to the drain headers, are provided in the stations for high-pressure process streams.

9.3.2.4 Tests and Inspections

The sample stations are in continuous use during normal plant operation, therefore PSS functionality is continuously demonstrated during normal plant operation. The sample stations are tested and calibrated at frequencies in accordance with the sample station supplier's operation and maintenance requirements.

9.3.2.5 Instrumentation Requirements

PSS instrumentation is provided in each sample station for the following:

- Provisions are made to stop sample flow upon detection of high-temperature sample flow leaving the sample cooler;
- Pressure and temperature indication is provided for all high-pressure and high-temperature samples, respectively;
- Process conditions are measured and recorded for each sample flow;
- Alarms are provided for necessary measurement indications; and
- Provisions are made for sample flow to be indicated.

Additional monitoring equipment is provided within the panels to meet the process stream monitoring conditions listed in Table 9.3-1.

10.3.2.3 System Operation

At low plant power levels, the TMSS may be used to supply steam to the Turbine Gland Seal System (TGSS).

At normal reactor power, steam generated in the reactor is supplied to the second stage reheater of the steam MSRs. Main steam supply pressure to the MSRs is regulated at low power levels.

If a large, rapid load reduction occurs, steam is bypassed directly to the condenser via the turbine bypass system (Subsection 10.4.4 for a description of the turbine bypass system).

10.3.3 Evaluation

All components and piping for the TMSS are designed in accordance with the codes and standards listed in Section 3.2. This ensures that the TMSS accommodates operational stresses resulting from static and dynamic loads, including water (steam) hammer and relief valve discharge loads, normal and abnormal environmental conditions, and includes provisions to limit water entrainment. Operating and maintenance procedures include adequate precautions to minimize the potential for water (steam) hammer.

The break of a main steam line or any branch line does not result in offsite radiation exposures in excess of the limits of 10 CFR ~~100~~52.47(a)(2)(iv) because of the safety features designed into the plant. The main steam line pipe break accident outside containment is addressed in Chapter 15, and high energy pipe failure is discussed in Section 3.6.

The TMSS complies with applicable General Design Criteria (GDC) in Appendix A to 10 CFR 50. GDC 2, 4, 5, and 34 are not applicable to the TMSS since the TMSS is classified as nonsafety-related, has no safety-related functions, does not share SSCs with other units, and is not required to provide residual heat removal functions. Additional information regarding compliance with the above GDCs is provided in Section 3.1.

10.3.4 Inspection and Testing Requirements

The preservice and inservice inspection programs for the ASME Section III, Class 2 portions of the system are in accordance with the requirements set forth in Section 6.6 and Subsection 3.9.6. Other piping and components are inspected and tested in accordance with the requirements of ASME B31.1.

Accessibility for inservice inspections is provided by appropriate arrangement of piping and major equipment and accessible arrangement of vents and drains in the system to comply with ASME Code Section XI [Paragraph 2.2.1(c)] requirements for the performance of inservice inspection and testing for assessing operational readiness. Areas that require inspection are provided with access space and removable insulation.

10.3.5 Water Chemistry (PWR)

This section applies to a Pressurized Water Reactor (PWR), and is therefore not applicable.