February 7, 2008
NRC:08:012

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

U.S. EPR Final Safety Analysis Report, Supplement 1

Ref. 1: Letter, Sandra M. Sloan (AREVA NP Inc.) to Document Control Desk (NRC), "Application for Standard Design Certification of the U.S. EPR (Project No. 733)," NRC:07:070, December 11, 2007.

On December 11, 2007, AREVA NP Inc. (AREVA NP) tendered an application for a standard design certification for the U.S. EPR (Reference 1). The application included a Final Safety Analysis Report (FSAR). Since the application was tendered, the NRC has requested that AREVA NP provide additional information in the FSAR to support the acceptance review of the design certification application in three areas. AREVA NP is hereby transmitting the following attached information to further support the NRC's acceptance review:

- FSAR Tier 2 Section 9.3.1, *Compressed Air System* (pp. 9.3-1 – 9.3-6) - mark-up providing additional information

- ANP-10293, *U.S. EPR Design Features to address GSI-191 Technical Report*, and respective FSAR markups:
    - FSAR Tier 2 Section 6.2.2, *Containment Heat Removal Systems* (pp. 6.2-137 and 6.2-138)
    - FSAR Tier 2 Section 6.3, *Emergency Core Cooling System* (pp. 6.3-1 – 6.3-19) and
    - FSAR Tier 2 Table15.0-60, *NRC Generic Letters* (pp. 15.0-126 –15.0-128)

- FSAR Tier 2 Section 14.3, *Inspection, Test, Analysis, and Acceptance Criteria* (pp. 14.3-1 – 14.3-8) - markup to clarify compliance with Part 20

Attachment 1 provides the revised pages in a redline/strikeout format and supplements the FSAR submitted by the reference. These pages provide the staff with the information to support the U.S. EPR design certification review. This supplemental information will be included in revision 1 to the U.S. EPR FSAR.

Attachment 2 provides ANP-10293, *U.S. EPR Design Features to address GSI-191 Technical Report*. This report does not contain any information that AREVA NP considers to be proprietary.

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If you have any questions related to this submittal, please contact Ms. Sandra M. Sloan, Regulatory Affairs Manager for New Plants Deployment. She may be reached by telephone at 434-832-2369 or by e-mail at sandra.sloan@areva.com.

Sincerely,

Ronnie L. Gardner, Manager
Site Operations and Corporate Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: J. Rycyna
G. Tesfaye
Project 733

ATTACHMENT 1

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The compressed air system (CAS) ~~provides compressed air to instruments and devices requiring instrument or service quality air. For the U.S. EPR, the only safety-related function of the CAS is containment isolation. Containment isolation for the CAS is provided in Section 6.2.4.~~ consists of compressors, dryers, filters, receivers and other equipment required for performing its non-safety-related functions.

9.3.1.1 Design Bases

The CAS provides compressed air for the following services:

- Instrument air for non-safety-related valves and other equipment located in the Conventional Island (CI).
- Instrument air for opening the containment ventilation purge dampers.
- Instrument air to valves, pumps and other equipment located in the radioactive waste, decontamination, blowdown demineralization, fuel handling and other systems for non-safety-related functions.
- Service air throughout the plant (for using air-operated tools and purging tanks).
- The containment isolation features for the containment penetrations in the CAS are described in Section 6.2.4.
- There are no air-operated valves (AOV) or air-operated equipment required to function in response to an accident where the compressed air is provided by the CAS.
- The design of the CAS is in compliance with the resolution of NUREG-0933, Generic Safety Issue 43, Reliability of Air Systems (Reference 1).
- The CAS is designed for a single unit and is not shared with other units.

9.3.1.2 System Description

9.3.1.2.1 General Description

The CAS consists of a compressed air generation system and a compressed air distribution system. The compressed air generation system is located entirely in the Turbine Building (TB). It supplies compressed air to the compressed air distribution systems in the Nuclear Island (NI) and CI. The location of the compressed air generation system in the TB minimizes the likelihood of leakage from radioactive systems being ingested into the CAS.

Figure 9.3.1-1—Compressed Air Generation System, shows a schematic diagram of the compressed air generation system while Figure 9.3.1-2—Compressed Air Distribution System, shows a schematic diagram of the NI and CI compressed air distribution system.

Component Description

Table 3.2.2-1 provides the quality group and seismic design classification of components and equipment in the CAS. The containment isolation valves (CIV) and penetrations are the only safety-related components in the CAS.

Instrument Air

Two oil-free rotary screw compressors are provided. They are connected for parallel operation and provide clean, dry, oil-free instrument air. During normal plant operation, one instrument air compressor operates continuously, loaded and unloaded depending on the system demand. The other instrument air compressor is in standby and is started in the event the operating compressor fails or if the system pressure drops below a preset value. Each compressor is equipped with an inlet air filter, aftercooler and moisture separator to condition the compressed air.

Two instrument air receivers serve as a storage volume to supply a limited amount of compressed air following a compressor failure. Overpressure protection is provided via pressure relief valves located on the air receivers.

Duplex prefilters are provided at the instrument air dryer inlet in order to protect the adsorption dryer units. Prefilter elements are constructed of corrosion-resistant materials.

Duplex afterfilters are provided at the instrument air dryer outlet to prevent the carryover of desiccant dust. These filters also remove rust, scale and dirt. Afterfilter elements are constructed of corrosion resistant materials. The duplex afterfilters have an automatic drain trap to remove accumulated condensate.

An air dryer is installed downstream of each instrument air compressor to remove moisture from the air.

Service Air

A single oil-free rotary screw compressor provides service air. During normal operation, the service air compressor operates continuously, loaded and unloaded depending on the system demand. The compressor is equipped with an inlet air filter, aftercooler and moisture separator to condition the compressed air.

A service air aftercooler–moisture separator is provided immediately downstream of the compressor to cool the flow of air from the air compressor and remove entrained moisture.

The service air receiver located directly after the aftercooler–moisture separator serves as a storage volume to supply a limited amount of compressed air following a compressor failure. Overpressure protection is provided via pressure relief valves located on the air receiver.

A duplex filter is provided downstream of the service air receiver discharge to remove rust, scale and dirt that might be present in the service air distribution system.

9.3.1.2.2 System Operation

Normal Operation

During normal plant operation, one instrument air compressor provides the required pressure in the instrument air receivers.

The service air compressor is in operation and provides the required pressure to the service air receiver. Service air is not required inside containment during normal plant operation.

The CAS supplies the opening function of the containment ventilation dampers for the containment building ventilation system (CBVS). However, compressed air is not required to perform the safety-related function in this system; for these containment dampers, air is required to keep the dampers open, a non-safety-related function. The containment dampers close on spring pressure. Refer to Section 9.4.7 for a description of the CBVS.

Accident Conditions

The CAS is not required to operate during or following an accident condition.

The CIVs maintain containment integrity, including an accident with a loss of offsite power and a loss of an emergency diesel generator. The instrument air motor-operated CIVs are actuated by a safety-related I&C System following a containment isolation signal (CIS). The service air manual CIVs are locked in the closed position during normal plant operation and remain closed during any accident condition.

The CAS does not provide compressed air to the diesel generator starting air system (DGSAS). Refer to Section 9.5.6 for a description of the DGSAS.

9.3.1.3 Safety Evaluation

The safety evaluation of the containment isolation system is given in Section 6.2.4.

The U.S. EPR does not use air operators on safety-related valves, except for the non-safety function of opening normally-closed containment ventilation dampers. This practice effectively avoids the problems noted in NUREG-1275 (Reference 2).

The CAS is not required for any safety function. Failure of the CAS does not affect any accident mitigation function. Failure of the CAS does not cause degradation of barriers to radiation releases during normal operation.

9.3.1.4 Inspection and Testing Requirements

The CAS components are inspected and tested during initial plant startup as part of the initial test program. Refer to Section 14.2 (test abstract #054 and #179) for initial plant startup test program.

9.3.1.5 Instrumentation Requirements

The instrument air CIVs are actuated by a safety-related I&C System. The actuator control logic of the CIVs is periodically tested. The CIVs are actuated from the main control room or the remote shutdown station. Section 7.3 describes the safety-related instrumentation associated with containment isolation. All other I&C functions are performed by non-safety-related I&C Systems described in Section 7.1.

9.3.1.6 References

1. NUREG-0933, "A Prioritization of Generic Safety Issues," U.S. Nuclear Regulatory Commission, Revision 21, September 2007.

2. NUREG-1275, Volume 2, "Operating Experience Feedback Report - Air Systems Problems," U.S. Nuclear Regulatory Commission, December 1987.

Figure 9.3-1—Compressed Air Generation System

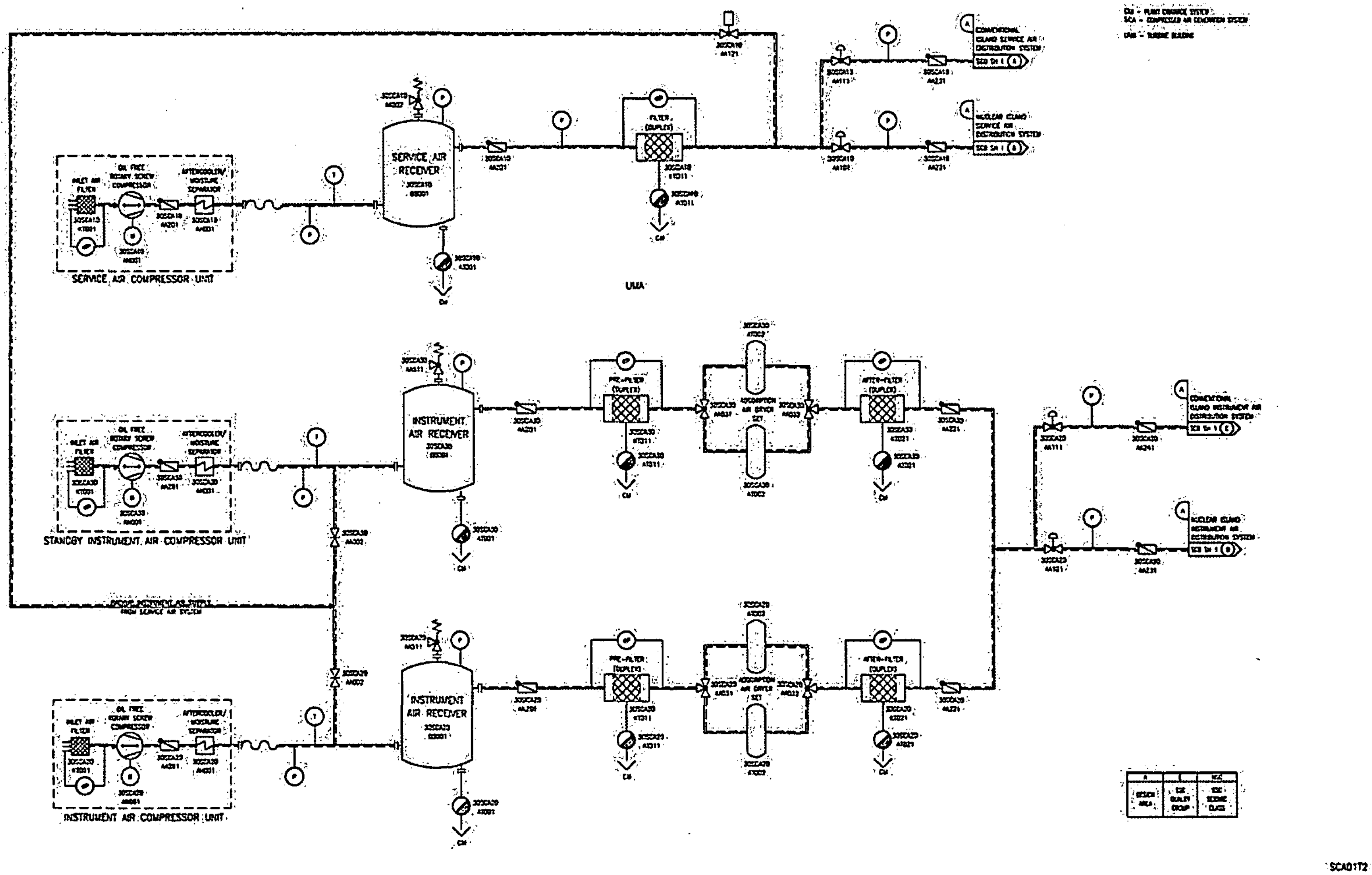
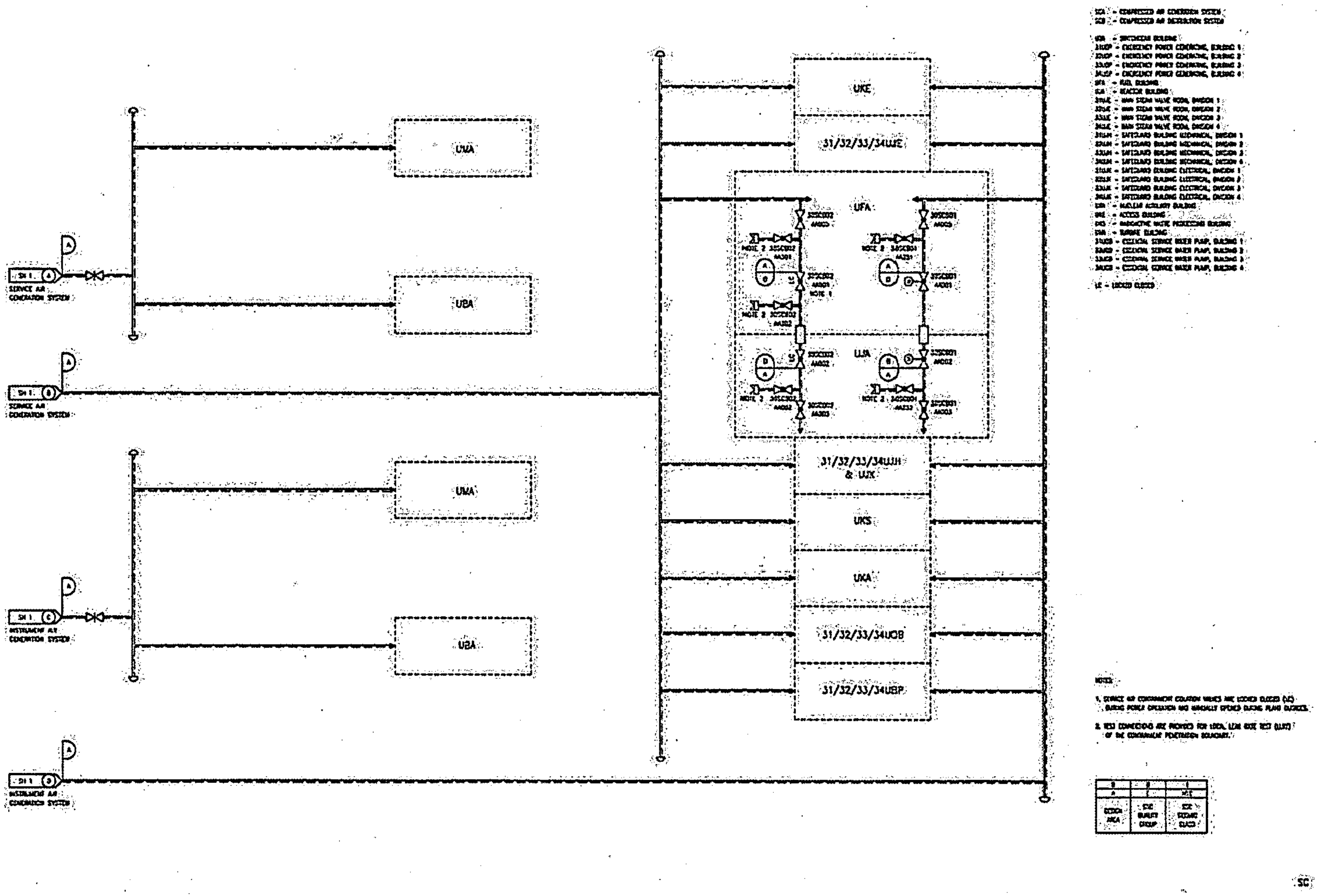


Figure 9.3-2—Compressed Air Distribution System



6.2.2 Containment Heat Removal Systems

Containment heat removal systems reduce containment pressure and temperature following any LOCA and maintain them at acceptably low levels. For the U.S. EPR, the containment heat removal function is accomplished by cooling the IRWST inventory into which the spilled and condensing reactor coolant flows following RCS blowdown, via the LHSI cooling function of the SIS.

Following blowdown, the reactor coolant vapor produced from the RCS leak condenses on the containment heat sinks. The saturated water drains along the intermediate floors, grates, stairwells, and walls to the heavy floor of the containment building. The effects of condensation induce circulation zones that provide a mixing of the containment atmosphere during and after blowdown. The saturated water draining from the heat sinks pools and forms a large condensation surface on the heavy floor. In the case of a LOCA, saturated reactor coolant spills out of the break, splashes on the heavy floor, and induces waves in the pooled water, which provides constant circulation that further promotes condensation on the pool surface. Curbed grates in the heavy floor drains direct the condensed reactor coolant back to the IRWST. The water in the IRWST is recirculated by the LHSI pumps through the heat exchangers, where it is cooled by the component cooling water system (CCWS), and pumped into the RCS to cool the core. The condensation of the reactor coolant vapor by the heat sinks and the subcooled liquid flowing across the heavy floor, and rejection of the heat to the environs via the LHSI heat exchanger cooling chain, results in long-term cooling and depressurization of the containment.

Long-term hydrogen mixing experiments were conducted in the Battelle Model Containment (BMC) facility to verify hydrogen mixing by natural convection after a LBLOCA in the Biblis containment. The geometry of the BMC facility is similar to that of the U.S. EPR; no fan coolers or sprays were available for active cooling during these tests.

The U.S. EPR does not credit active cooling by fan coolers or sprays inside containment during a postulated LBLOCA. The similar geometry and the minimum active containment cooling systems used in the BMC facility tests make the findings directly applicable to the LBLOCA for the U.S. EPR containment analysis.

As described in the June 1999 NEA/CSNI report on containment thermal-hydraulics and hydrogen distribution (Reference 6), these tests provide direct experimental evidence that sump flashing or evaporation resulting from sump liquid superheat generate effective natural convection currents throughout containment. The condition of sump liquid superheat also occurs in the IRWST in the long term as a result of LBLOCA. These tests show that sump evaporation that occurs in the absence of active containment cooling establishes effective natural convection currents throughout the U.S. EPR containment.

The design basis containment analysis for loss of coolant accidents and main steam line breaks, and the containment pressure and temperature responses for these events, is discussed in Section 6.2.1. As shown in Figures 6.2.1-12, 6.2.1-16, and 6.2.1-20, the LHSI heat exchangers are sufficient to reduce the containment pressure to half its peak in less than eight hours after a LOCA.

The SIS provides cooling of the IRWST in the event of a LOCA and provides long-term cooling and pressure suppression of the containment volume. The SIS consists of four independent trains, providing sufficient capacity, diversity, and independence to perform its required safety functions following design basis transients or accidents assuming a single failure in one train while a second train is out-of-service for preventive maintenance. Section 6.3 discusses the SIS, including design bases, instrumentation, and inspection and testing requirements. Section 6.3 includes a discussion of the design features for avoidance of the potential loss of long-term cooling capability due to sump screen blockage in the IRWST and presents the performance evaluation of the design, a summary of component testing, and a comparison to the regulatory positions of RG 1.82.

6.3 Emergency Core Cooling System

The safety injection system (SIS) provides emergency core cooling for the U.S. EPR. Four supply and return trains comprise the system, one for each of the reactor coolant system (RCS) loops. Individually, each of these trains can supply the required core cooling. The four supply trains, which serve the safety injection function, charge through parallel paths from a low head safety injection (LHSI) pump, a medium head safety injection (MHSI) pump, and an accumulator in each train. The injection pumps draw water from the in-containment refueling water storage tank (IRWST) for their emergency function.

The MHSI pumps and the accumulators inject directly into the cold legs. The LHSI pumps inject through the LHSI heat exchangers (HX) to the cold legs. Closed loop cooling via the LHSI pump (in residual heat removal mode) for post-accident heat removal is also available by aligning the suction to the RCS hot legs. The LHSI system may be re-aligned during accident recovery for hot-leg injection to prevent boron precipitation and mitigate steaming from the break.

The residual heat removal (RHR) function of the safety injection system/residual heat removal system (SIS/RHRS) for normal shutdown cooling of the reactor is described in Section 5.4.7.

6.3.1 Design Bases

The SIS limits fuel assembly damage during core flooding and emergency core cooling following a loss of coolant accident (LOCA). The SIS removes post-accident decay heat from the RCS and provides post-accident containment cooling via the LHSI HXs. The system consists of four independent and separated trains, each housed and protected in its own seismically qualified Safeguard Building (SB), as further described in Section 6.3.2. This separation and independence provides protection from physical damage due to natural phenomena and hazards and allows fulfillment of the system safety function in the event of a single failure.

Following postulated LOCAs, the SIS maintains fuel cladding temperature, cladding oxidation, hydrogen generation, core geometry, and long-term core temperature within the limits specified in 10 CFR 50.46. SIS actuation provides protection for the following postulated transients, accidents, and operational events:

- Main steam line break (MSLB) - Following a small or large MSLB, the MHSI trains provide RCS boration and coolant inventory control during cooldown.
- Steam generator tube rupture (SGTR) - Following an SGTR, the MHSI trains inject borated water to provide a sufficient coolant inventory.

- Small-break LOCA (SBLOCA), break size less than or equal to 0.5 ft^2 - The SIS, in conjunction with automatic secondary-side partial cooldown, provides borated coolant injection, which limits RCS draining and keeps the core covered and cooled throughout the event. The system provides this function even if there is a loss of a train due to the most limiting single failure coincident with one train unavailable because of maintenance. Further evaluation of SIS performance for this limiting event is presented in Section 6.3.3.

- Large-break LOCA (LBLOCA), break size greater than 0.5 ft^2 up to a complete rupture of an RCS hot or cold leg - To avoid exceeding the limits of 10 CFR 50.46, the SIS provides sufficient core cooling even if there is a loss of a train, due to the most limiting single failure, coincident with one train being unavailable due to maintenance. Further evaluation of SIS performance for this limiting event is presented in Section 6.3.3.

- Inadvertent opening of a pressurizer safety relief valve (PSRV) - The MHSI pumps provide RCS makeup in the event of inadvertent opening of a PSRV.

- RCS loop level decrease during shutdown or midloop operation - The MHSI pumps provide RCS makeup in the event of spurious draining of the RCS or SBLOCA during shutdown cooling operations. To compensate for the reduced pressure and makeup flow requirement for this operational condition, the large MHSI minimum flow line opens prior to injection to reduce the MHSI injection head. RCS pressure remains below approximately 580 psia during this event.

The SIS and its support and ancillary systems are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Section 3.2 identifies component classifications (GDC 1, 10 CFR 50.55a(a)(1)). Appropriate to its reactor core cooling function, the SIS is:

- Designed to codes consistent with the quality group classification assigned by RG 1.26.

- Protected from the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, and external missiles, and designed to function following such events (GDC 2).

- Designed to the Seismic Category I designation assigned by RG 1.29 so that it remains functional after a safe shutdown earthquake (SSE) (GDC 2).

- Designed to remain functional following the postulated hazards of fire and explosion, internal missiles, pipe whipping, and discharging fluids (GDC 3 and GDC 4).

- Not shared among nuclear power units (GDC 5).

- Provided with both an onsite and an offsite electric power system, each of which can alone power the SIS to its full capacity (GDC 17).

- Capable, in combination with the extra borating system (EBS), of adding sufficient neutron poison to reliably control reactivity changes and maintain core cooling under postulated accident conditions, with an appropriate margin for stuck control rods (GDC 27).
- Designed to remain functional in the event of a single active component failure coincident with the loss of either the onsite or offsite power source (GDC 35).
- Designed to permit appropriate periodic inspection of important components to verify the integrity and capability of the system (GDC 36, GDC 39).
- Designed to permit appropriate periodic pressure and functional testing to confirm:
  - The structural and leak tight integrity of its components.
  - The operability and performance of its active components.
  - The operability of the system as a whole. This testing is performed under conditions as close to design as practical for the full operational sequence of the system, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system (GDC 37, GDC 40).
- Designed, through the features built into the in-containment refueling water storage tank system (IRWSTS), to reduce the containment pressure and temperature following a loss of coolant accident (LOCA) and maintain them at acceptably low levels (GDC 38), and to provide long term post-LOCA core cooling requirements as required in 10 CFR 50.46(b)(5).
- Designed to perform under anticipated normal, testing, and design basis accident environmental conditions in compliance with 10 CFR 50.49.
- Supplied by highly reliable, Class 1E, and diverse power and control systems in conformance with RG 1.32. Class 1E power supply for the U.S. EPR is addressed in Chapter 8.
- Supplied by a highly reliable water source (the IRWST) for long-term recirculation cooling following a LOCA, with adequate protection against loss of net positive suction head (NPSH) due to debris entrainment, in conformance with RG 1.82.
- Designed with the capability for leakage detection and control to minimize the leakage from those portions of the SIS outside of the containment that may contain radioactive material following an accident (10 CFR 50.34(f)(2)(xxvi)).

Positive indication is provided in the control room of flow in the discharge pipe from the RCS safety and relief valves (10 CFR 50.34(f)(2)(xi)) as described in Section 5.2.2. Reactor vessel instrumentation described in Section 7.5.2.1 displays an unambiguous, easy-to-interpret indication of inadequate core cooling (10 CFR 50.34(f)(2)(xviii)).

The SIS design and analysis incorporates resolution of the relevant USIs, and medium- and high-priority GSIs, specified in NUREG-0933 (Reference 1). Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) identifies where each relevant issue is addressed.

The SIS design incorporates operating experience insights from the following generic letters and bulletins:

- GL 80-014 (Reference 2) addresses LWR primary coolant system pressure isolation valves, specifically the mitigation of interfacing systems LOCA. The SIS design features addressing intersystem LOCA are described in Section 5.4.7.
- GL 80-035 (Reference 3) addresses the effect of a DC power supply failure on SIS performance. The four-train SIS design, with independent emergency power supplied to each train, addresses this issue by providing sufficient redundancy to perform its functions even with the unavailability of an entire train as described in Section 6.3.2.5.
- GL 81-021 (Reference 4) addresses natural circulation cooldown. This issue is addressed in Sections 10.4.9.3 and 15.0.4.1.2.
- GL 85-16 (Reference 5) addresses the effects of high boron concentrations. The borated water from the IRWST, where the SIS pumps take suction, is not easily susceptible to precipitation due to its relatively low boron concentration. The extra borating system injects concentrated boric acid solution when required to maintain reactivity margin for plant shutdown. The EBS is designed to prevent boric acid crystallization as described in Section 6.8.
- GL 86-07 (Reference 6) addresses the effects and prevention of water hammer. Refer to Section 5.4.7 for discussion of provisions for the prevention of water hammer in the SIS piping.
- GL 89-10 (Reference 7) addresses safety-related motor-operated valve testing and surveillance. This issue is addressed in Section 3.9.6.
- GL 91-07 (Reference 8) addresses reactor coolant pump (RCP) seal failure and station blackout. Refer to Section 5.4.1 for discussion of provisions for RCP seal failure and station blackout.
- GL 98-04 (Reference 9) addresses the potential for degradation of emergency core cooling and the containment spray systems after a LOCA due to construction and protective coating deficiencies and the entrainment of debris in recirculating reactor coolant. This issue is described in Section 6.3.2.5.
- GL 2004-02 (Reference 18) addresses the potential susceptibility of pressurized water reactor recirculation sump screens to debris blockage during design basis accidents and the potential for additional adverse effects due to debris blockage of flow paths necessary for recirculation and containment drainage. This issue is addressed in Section 6.3.2.5.

- BL 80-18 (Reference 10) addresses the maintenance of adequate minimum flow through centrifugal charging pumps following secondary side high energy line ruptures. The SIS pumps include minimum flow lines that provide adequate recirculation to prevent overheating of the pumps as described in Section 6.3.2.2.

- BL 86-03 (Reference 11) addresses potential failure of multiple ECCS pumps due to single failure of air-operated valves (AOV) in minimum flow recirculation lines. AOVs are not used in the SIS.

- BL 88-04 (Reference 12) addresses the potential for the loss of pump function due to deficiencies in the design of minimum flow lines. The SIS design addresses this issue by incorporating separate minimum flow lines that are not shared among the SIS pumps as described in Section 6.3.2.5.

- BL 93-02 (Reference 13) addresses debris plugging of emergency core cooling suction strainers. This issue is addressed in Section 6.3.2.5.

- BL 01-01 (Reference 14) addresses circumferential cracking of reactor pressure vessel head penetration nozzles. This issue is addressed in Section 5.2.3.

- BL 02-01 (Reference 15) addresses reactor vessel head degradation and reactor coolant pressure boundary integrity. This issue is addressed in Section 5.2.3.

The discharge heads for the SIS accumulators and discharge heads and delivery flowrates for the LHSI system and the MHSI system are listed in Table 6.3-1—Accumulators Design and Operating Parameters, Table 6.3-2—Low Head Safety Injection Pumps Design and Operating Parameters, and Table 6.3-3—Medium Head Safety Injection Pumps Design and Operating Parameters. The SIS provides core cooling capability for a wide spectrum of LOCAs, considering the hydraulic flow resistance of the SIS piping and valves and the available NPSH. The volume of the IRWST, as listed in Table 6.3-4—IRWST Design Parameters, provides sufficient borated water for long-term core cooling. In addition, the boron concentration in the IRWST, in combination with the EBS, provides negative reactivity to keep the core subcritical.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The SIS consists of four independent trains, designated Trains 1, 2, 3, and 4, one supplying each reactor coolant loop. The four trains are separated into four safety divisions and are functionally identical, as shown in Figures 6.3-1—Safety Injection System and 6.3-2—Safety Injection/Residual Heat Removal Train. The IRWST arrangement is shown in Figure 6.3-3—IRWST Layout.

Each SIS train has separate MHSI and LHSI pump trains and an accumulator injection train. The MHSI and LHSI pump trains share an isolable suction line from the IRWST. This three-way valve lines up the IRWST to both the MHSI and LHSI pump suctions when in the open position. The LHSI pump train includes an HX and a suction line from the RCS hot leg for residual heat removal, which may be re-aligned for LHSI hot-leg injection. The discharge lines for all three MHSI, LHSI, and accumulator injection trains branch together to share an injection nozzle on their associated RCS cold leg. Cross-connects between Trains 1 and 2 and between Trains 3 and 4, which are normally isolated by two motor-operated valves in series to maintain train separation, allow individual trains to be removed from service for maintenance. Each cross-connect provides an alternate injection path for the train that remains in service. This configuration mitigates the effect of degraded safety injection due to steam entrainment during a LOCA, when the only available LHSI connection (considering one is unavailable due to single failure, another out for maintenance, and another train feeds the broken loop) is located adjacent to the broken leg. During such maintenance activities, the motor-operated valves for both cross-connects are secured open (breakers racked out) for protection against active single failures, as described in Section 6.3.2.5.

The component cooling water system (CCWS) is the cooling medium for the LHSI HXs (all four trains), the MHSI pump motor coolers (all four trains), and the LHSI pump motor and seal coolers for Trains 2 and 3. The safety chilled water system (SCWS) is the cooling medium for the LHSI pump motor and seal coolers for Trains 1 and 4. The essential service water system (ESWS) serves as the final cooling medium, rejecting the heat transferred from the CCWS to the ultimate heat sink.

The four SIS trains are powered, respectively, by electrical divisions 1 through 4. Each electrical division is a separate and independent power supply housed and protected in its own SB. Each electrical division is also supplied by its assigned emergency diesel generator in the event of a loss of offsite power (LOOP). Chapter 8 provides detailed information on the U.S. EPR electrical system.

6.3.2.2 Equipment and Component Descriptions

6.3.2.2.1 System Overview

Each MHSI train consists of a pump, an isolable supply branch from the shared IRWST suction line, and a discharge line that tees into its respective cold-leg LHSI injection line just upstream of the inboard LHSI-to-RCS isolation check valve. A line tees off of the injection line upstream of the inboard MHSI-to-LHSI injection isolation valve and leads back to the IRWST. This line branches into two flow lines; the smaller one for pump minimum flow protection and the larger one for reducing the MHSI discharge head. A line for filling the accumulator tees off of the smallest of these branch lines upstream of its maintenance isolation valve.

Each accumulator injection train has one accumulator whose isolable injection line tees into its respective cold-leg LHSI injection line just upstream of the inboard LHSI-to-RCS isolation check valve.

The LHSI train consists of an LHSI pump, LHSI HX, LHSI HX bypass line with flow control valve, shared suction line from the IRWST with a motor-operated isolation valve, LHSI HX discharge line with temperature control valve, RCS hot-leg suction line, cross-connects between pairs of trains, and various isolation and realignment valves as required to support operation, maintenance, shutdown, or accident mitigation. A mini-flow and test line tees off of the cold-leg injection line upstream of the outboard LHSI-to-RCS isolation check valve.

The SIS piping is protected from overpressure events by safety relief valves installed at locations most susceptible to such events. The design overpressure transient is the spurious startup of an MHSI pump with the large mini-flow line isolated. The set-points and capacities for these safety relief valves limit the protected system to 110 percent of its design pressure.

Detection and monitoring of SIS leakage within the Reactor Building (RB) is provided by the reactor coolant pressure boundary (RCPB) leakage detection systems described in Section 5.2.5. Leakage from the SIS in the SBs is detected and monitored by operating procedures and programs. Each SB has sump level indication to detect SIS/RHRS leakage.

The postulated accident sequences and analyses, including equipment actuation and response times, and design requirements for SIS delivery lag times, are described in Section 15.6.5.

6.3.2.2.2 System Components

Accumulators

Each accumulator is an austenitic stainless steel tank with a total volume of approximately 1950 ft^3 and is filled with approximately 1250–1400 ft^3 (approximately 10,000 gallons) of borated water and approximately 550–700 ft^3 of pressurized nitrogen. Nominal operating pressure is approximately 665 psig. The accumulators are designed so that the nitrogen pressure after their injection is lower than the LHSI discharge pressure. Thus, they do not inject nitrogen into the RCS prior to commencement of LHSI injection, even in the unlikely event of the loss of MHSI pumps. The relevant accumulator design and performance data are presented in Table 6.3-1.

Pumps

The LHSI and MHSI pumps are horizontally mounted, centrifugal pumps with single mechanical seals. Their motors are water cooled by the CCWS, with the exception of the LHSI pumps for Trains 1 and 4, which are cooled by the SCWS. Nominal flowrate for the LHSI pump is approximately 2200 gpm at 480 ft of total developed head (TDH), and for the MHSI pump it is approximately 600 gpm at 2260 ft of TDH. The relevant LHSI and MHSI pump design and performance data are presented in Tables 6.3-2 and 6.3-3, respectively.

Heat Exchangers

The LHSI HXs are U-tube type, horizontally mounted, with reactor coolant flow through the austenitic stainless steel tubes and CCWS flow through the ferritic shell side. The relevant HX design and performance data are presented in Table 6.3-5—LHSI Heat Exchanger Design and Operating Parameters. Conservative fouling factors are incorporated into the performance evaluation of the LHSI HXs.

Piping, Fittings and Valves

The pipes, valves, and fittings of the SIS are austenitic stainless steel. Their design and performance ratings are commensurate with their expected service conditions. The relevant piping, valves, and fittings design data are presented on Figure 6.3-2—Safety Injection System/Residual Heat Removal Train.

In-Containment Refueling Water Storage Tank

The IRWST is an open pool within a partly immersed building structure. It is located at the bottom of the containment between the reactor pit and the secondary shield wall, below the level of the heavy floor which supports the primary components. It is connected to various safety and non-safety systems and serves as a water source, heat sink, and return reservoir. Select design data for the IRWST are shown in Table 6.3-4.

The IRWST supplies borated water to the SIS, the severe accident heat removal system (SAHRS), and the chemical and volume control system (CVCS). It also supplies the fuel pool cooling system (FPCS) via the CVCS suction line. The IRWST provides the necessary inventory of borated water for design basis events. It contains a minimum 66,886 ft^3 of borated water which is monitored for its level, temperature, and homogeneous boron concentration. The water is used for both refueling and SIS operations and provides:

- Sufficient water during plant shutdown to fill the reactor cavity, the internal storage pool, the RB transfer pool, and the RCS.
- Sufficient water depth (static pressure head) to the suction of the SIS, SAHRS, and CVCS pumps during normal and accident conditions (per RG 1.1).

- A heat sink and water inventory for flooding the core melt in the spreading area during a beyond design basis event (severe accident).

The walls of the IRWST are lined with an austenitic stainless steel liner covering the immersed region of the building structure. The liner prevents leaks and the interaction of the boric acid with the concrete structure. Leaks that occur are collected, monitored, and quantified by the nuclear island drain and vent system (NIDVS).

The IRWST is provided with the following three filtering stages for the borated water return path to its integral sumps as shown in Figure 6.3-4—SIS Sump Debris Entrainment Prevention Features:

- The trash racks and the weirs above the heavy floor openings to the IRWST are considered components of the IRWST. After a LOCA, the flow of coolant out of the RCS back to the IRWST passes through four openings in the heavy floor. The trash racks prevent large debris from entering the IRWST, while the weirs provide a barrier that retains sediment and debris on the heavy floor.
- Retaining baskets in the IRWST below each heavy floor opening trap debris transported by the flow past the trash racks and weirs. Two of the retaining baskets also filter flow from the annular space in containment to the IRWST. The openings in the retaining baskets provide efficient retention of fiber and particulate debris. A gap between the top of the baskets and the heavy floor provides a flow path if the retaining basket is full or clogged.
- The SIS and SAHRS strainers are arranged above each respective SIS and SAHRS sump. These strainers are designed as large cages with inclined sieves to facilitate debris detachment during backflushing. The opening size of the sieves limits the passage of debris during SIS and SAHRS recirculation flow to avoid pump malfunction and clogging of the smallest restriction in the core. The CVCS sump is also provided with a suction strainer.

The large dispersion area within the IRWST results in low flow velocity and promotes settling of fine debris that passes through the retaining baskets. The orientation of the various IRWST sumps is shown on the sump level plan view on Figure 6.3-5—IRWST Sump Level Plan View. The orientation of the trash racks and weirs is shown on the heavy floor plan view on Figure 6.3-6—IRWST Heavy Floor Level Plan View.

The IRWST sump screen flow performance was evaluated to verify that adequate long-term core cooling remains available in spite of impairment by accident-generated debris as well as debris in containment prior to the accident. The conservative estimate of total debris used for the evaluation, and an estimate of total debris in the containment of the U. S. EPR, is presented in Table 6.3-6. The increased use of reflective metal insulation (RMI), which is not subject to transport to the SIS sumps, in the U. S. EPR design in place of most or all of the fibrous or micro-porous insulation assumed in the evaluation further reduces the potential for post-accident blockage of the sumps.

The features of the IRWST screen design conform to RG 1.82 and address the issues of GSI-191, as further described in Section 6.3.2.5. Technical Report ANP-10293, "U.S. EPR Design Features to Address GSI-191" (Reference 19) provides additional description of the U.S. EPR design features that limit the impact of post-accident debris accumulation on SIS performance, summarizes the performance evaluations and component test program, and compares the design to the regulatory positions of RG 1.82 and the information requested in GL 2004-02.

Performance of the strainers is enhanced by cleanliness programs that limit debris in the containment. A COL applicant that references the U.S. EPR design certification will describe the containment cleanliness program which limits debris within containment.

Coolant pH adjustment baskets containing granulated trisodium phosphate dodecahydrate (TSP-C) are strategically placed in the inlet flow path to the IRWST within the boundary perimeter of the weirs at the four heavy floor openings of the RB. Flow through the baskets dissolves the TSP-C into the coolant that returns to the IRWST to passively neutralize entrained acids and maintain the alkalinity of the coolant. The pH of the recirculated coolant is maintained above 7.0. The control of pH in the recirculated coolant reduces the potential for stress-corrosion cracking of the austenitic stainless steel components, limits the generation of hydrogen attributable to corrosion of containment metals, and minimizes the re-evolution of iodine in post-LOCA containment solution, maintaining the radioiodine in solution to reduce radioactive releases to the environment. The minimum amount of granulated TSP-C for this pH control is 12,200 lbm. Section 15.0.3.12 provides an evaluation of post-accident water chemistry control.

The IRWST is connected to the molten core spreading area by pipes that are closed during normal operation and accident conditions. If a severe accident occurs and molten material reaches the spreading area, an actuation device melts, flooding valves open, and IRWST water flows into the spreading area to support the operation of the SAHRS. The IRWST is located at a higher elevation than the core spreading area to provide gravity flooding of the spreading area with the IRWST water inventory. The core spreading area and the SAHRS are described in Section 19.2.3.3.

6.3.2.3 Applicable Codes and Classifications

The SIS design complies with applicable industry codes and standards, and regulatory requirements, commensurate with the appropriate safety function for each of the individual components. Refer to Section 3.2 for seismic and system quality group classifications for the SIS components. Sections 3.9, 3.10, 3.11, 7.3, and 8.1.4 further address these requirements and their implementation for the U.S. EPR.

6.3.2.4 Material Specifications and Compatibility

Material selection for the SIS is based on the expected service conditions for the various components, the design life of the unit, and the materials strength and service requirements as further described in Section 3.9.3. SIS components that transport or come into contact with borated water, which are the majority of the pressure retaining, fluid bearing components, are constructed of austenitic stainless steel. The specific materials of construction for the SIS and their compatibility with system fluids are described in Section 6.1.1.

6.3.2.5 System Reliability

The instrumentation and controls (I&C) that initiate the SIS and are used to manage its operation are separated. They are independently powered from the same normal and emergency sources that power the associated motive equipment of the train. The process variables for the I&C, such as RCS pressure and pressurizer level, derive their input from independent sources. The design of the SIS I&C, including its quality, redundancy, and protection against the effects of single failure, is presented in Section 7.3.

The SIS trains meet Seismic Category I criteria for earthquake protection. Each of the four SIS trains is housed in a separate Seismic Category I structure. The buildings also protect the SIS against damage from other natural phenomena, such as floods, severe weather, and external hazards such as missiles. The design of the SBs is described in Section 3.8.4.

The SIS design allows online testing of the individual trains and components to assess their operational status and availability. The accessibility incorporated into the design allows complete testing and inservice inspection of critical components when plant conditions allow, such as during outages. Preoperational testing of the SIS verifies that the as-designed and as-constructed system fulfills its functional requirements. Periodic inservice testing confirms the continuing capability of the system. Testing and inspection activities for the SIS are addressed in Section 6.3.4.

The SIS is redundant and no single failure compromises the system safety functions. Vital power can be supplied from either the onsite or offsite power systems, as described in Chapter 8. Results of the single failure evaluation are summarized in Table 6.3-7—Safety Injection System Failure Modes and Effects Analysis. The most limiting single active failure for the SIS, assumed to occur at the onset of the design basis LOCA event, is the complete loss of one train. The redundancy incorporated into the system design allows the SIS to fulfill its safety function in spite of such failure, as further addressed in Section 15.6.5. The availability of four separate hot-leg connections, one for each of the SIS trains, preserves the hot-leg injection function to mitigate boron precipitation and steaming from the LOCA break.

As a conservative verification of the adequacy of the SIS design, the effects of a single passive failure during the long-term accident recovery phase are also considered. The most limiting passive failure is the loss of a coolant supply path, which might occur in the unlikely event of debris plugging of one of the sump suction sources or rupture of one of the supply lines. The redundant SIS design allows the unaffected trains to continue to provide long-term cooling in spite of such a passive failure. The addition of guard pipes on piping between the sump connections and the sump three-way isolation valves provides additional protection against flooding due to passive failure of the pipe upstream of the isolation valve.

The redundancy of the design extends to the capability to isolate affected sections of the individual trains as required. Since the critical function of the SIS is RCS injection, automatic containment isolation of the system, which could adversely impact the function of the system, is not provided. Combined manual and passive isolation capability, however, is provided as described in Section 6.2.4.

The SIS valves inside containment are located above the maximum floor flooding level which protects the valve motor operators from submersion following a LOCA. The RB flooding analysis is described in Section 3.4.3.3. The SIS suction piping is continuously vented to maintain it full of coolant whenever the system is required to be operable to prevent loss of pump suction pressure that could result from accumulation of gases in the piping. Components of the SIS, including those for its support and auxiliary equipment, are designed, procured, installed, and maintained to the appropriate quality and reliability standards. These quality standards, coupled with the system redundancy and physical and electrical separation, allow the SIS to fulfill the design objectives presented in Section 6.3.1.

The RB floor drains direct leakage within the containment, up to an accumulation of two inches depth, to the RB sump where it is monitored, quantified, and processed as liquid waste. The RB floor drains are part of the NIDVS described in Section 5.2.5. Accumulation of leakage in containment greater than two inches depth, which is indicative of a LOCA, flows into the IRWST where it is available for accident response. The relatively low volume of the RB drains, in comparison to that of the IRWST, allows mixing of coolant during injection and recirculation so that no areas accumulate very high to low pH solutions.

The IRWSTS design responds to the post-LOCA ECCS sump performance issues of GSI-191 by conforming to the guidance of RG 1.82. The IRWSTS deters post-accident debris accumulation and SIS sump strainer blockage, in accordance with the expectations of RG 1.82, by:

- Minimizing the post-accident debris source term. The RCS piping and components, and other potentially insulated systems or components within containment, are insulated with RMI, and negligible or no fibrous or micro-porous insulation. Due to its high density, RMI is not susceptible to transport and therefore does not contribute to strainer head loss.

- Providing a three-tiered debris retention design. The combination of weirs/trash racks and retaining baskets are effective in retaining most post-accident debris. Furthermore, the sump strainers (the third stage of the three-tiered debris retention design) have a large screen surface area to accommodate the small amount of debris that reaches it. The full coverage screens and retention baskets, which are rigidly mounted to the IRWST floor, prevent bypass of debris into the suction lines.

The design features addressing GSI-191 and the performance evaluations are further described in Section 6.3.2.2.2 and Reference 19. Reference 19 also describes the component test program and compares the design to the regulatory positions of RG 1.82 and the information requested in GL 2004-02.

6.3.2.6 Protection Provisions

The four independent SIS trains are individually housed in four separate, Seismic Category I, reinforced concrete structures as described in Section 3.8.4. Since the SIS itself is Seismic Category I, the system is protected from potential earthquake damage. The rugged structures also protect the system from other natural phenomena and external hazards. The design of the system includes margin to safely accommodate displacement due to thermal stresses and limited movement due to operational anomalies or external stimuli. Physical separation is provided for the SIS/RHR System redundant components, including cross connects, located within the Reactor Building such that local effects of any internal hazard (e.g., pipe whip) are restricted to one train. Specific layout provisions, arrangement of components, or design features prevent any global effects from an internal hazard affecting the operability of system components inside containment. Refer to Section 3.10 for seismic qualification of equipment. Protection against other natural phenomena is addressed in Sections 3.3 and 3.4. Missile protection and protection against dynamic effects are addressed in Sections 3.5 and 3.6, respectively. Section 9.5.1 and Appendix 9A address fire protection, Section 3.11 addresses environmental qualification of equipment, and Section 3.9 reviews the thermal and displacement stresses.

6.3.2.7 Provisions for Performance Testing and Inspection

The general installation and design of the SIS provides ready accessibility for testing and inspection. Process and auxiliary fluid paths are isolable and instrumented to accommodate maintenance and testing of the valves, instrumentation, and other critical SIS components, with multiple minimum flow paths provided for dynamic testing of the SIS pumps. The redundancy provided by the four separate trains of the system allows such activities to be performed online as well as during scheduled maintenance or outages. The arrangement of the piping and components is shown in Figures 6.3-1 through 6.3-3. Performance testing is addressed in Section 6.3.4.

6.3.2.8 Manual Actions

The SIS injects automatically in response to the safety injection signal and requires no operator intervention to accomplish its function. The emergency coolant supply is enclosed within the containment and is constantly replenished by recirculated coolant flow, therefore no operator action is required to provide the continuous supply of coolant or the removal of decay heat during the injection phase.

To prevent boron precipitation and mitigate steaming from the break, manual switchover to hot-leg injection is required approximately one to three hours into the event. This represents the response to the most severe of the postulated events, such as the LBLOCA.

For less severe events such as SBLOCA, automatic action is adequate to manage the event. After completion of the initial automatic response, it may be beneficial to manage the event with deliberate operator action. For instance, while the protection system initiates reactor trip and SIS startup following an SBLOCA, it may be possible, depending on the scale of the event, to identify and isolate the failed component, thereby terminating the event and allowing safe shutdown without further challenges to the safety systems. Such actions are in accordance with approved procedures developed as described in Section 13.5.2.

6.3.3 Performance Evaluation

During normal, at-power operation, the SIS is idle but configured for rapid automatic or on-demand response. Four cold-leg injection and IRWST suction flow paths are open, the hot-leg suction or alternate injection path is isolated, and the CCWS and SCWS cooling function for the SIS pumps and equipment area is in service or available to start on receipt of a demand signal. The SIS is isolated from the RCS cold legs by its boundary check valves which are back-seated by RCS pressure.

During shutdown cooling operations, the MHSI train is maintained in standby for RCS leakage makeup, with CCWS available for pump and area cooling. The large mini flow valve remains open to limit MHSI injection pressure and flowrate to levels appropriate for the shutdown condition.

Section 6.3.1 lists those postulated events for which SIS response is required. The most demanding SIS performance response, which bounds the response required for those events listed in Section 6.3.1, is the response to the range of SBLOCAs and the response to the most limiting LBLOCA. For that reason, SIS performance is evaluated for only these two most limiting events.

This analysis shows that the performance of the SIS during these limiting events limits the accident consequences to accommodate recovery, protect the health and safety of the public, and meet the regulatory requirements specified in Section 6.3.1. The event sequence and analysis, including equipment actuation and response times, and flow delivery curves, are described in Section 15.6.5.

6.3.3.1 Small Break LOCA

The most limiting SBLOCA is a break with a cross-sectional area of up to approximately 0.5 ft^2 in the cold leg between the SIS injection location and the reactor pressure vessel, with coincident LOOP. Such an event may not immediately challenge the SIS if the reactor coolant loss can be made up by the CVCS. The loss of primary coolant eventually results in a decrease in primary system pressure and pressurizer level, sequentially triggering a reactor and turbine trip, and closing the main feedwater full load isolation valves. Upon receipt of an SIS actuation signal, a partial cooldown of the secondary system, and thus the RCS, is initiated. During this sequence, the steam generators are fed by the emergency feedwater system, which is actuated by protection system signals.

The SIS actuates on low pressurizer pressure and automatically starts the MHSI and LHSI pumps. During the partial cooldown, the RCS pressure decreases sufficiently to allow MHSI injection into the cold legs. The partial cooldown is performed by available steam generators via steam dump to the atmosphere. The protection system automatically decreases the main steam relief train setpoints down to a fixed pressure that is low enough to permit MHSI injection, but high enough to prevent core recriticality due to low RCS temperature. For the smallest of these breaks, the RCS leakage, still in liquid form, does not remove sufficient coolant mass to offset injection flow and RCS depressurization stops at the end of the partial cooldown. If the MHSI flowrate is insufficient to compensate for the break flowrate, the RCS inventory continues to decrease. The break flowrate decreases as the void fraction in the cold legs increases. When the break flow changes to single phase steam, the ratio between steam production due to core decay heat and steam break venting changes and the break size is the dominant parameter for the depressurization sequence.

In case of the smallest breaks, condensation in the steam generator tubes, in combination with direct steam venting from the break, eventually reduces production of steam in the core to the point that the RCS saturation pressure plateaus slightly above the steam generator secondary side pressure. In the case of larger small breaks, steam venting is sufficient that the RCS depressurizes, regardless of the steam generator secondary side temperature, down to the point where accumulator injection, and eventually LHSI injection, occurs.

6.3.3.2 Large Break LOCA

The most limiting LBLOCA is a break in the cold-leg piping between the RCP and the reactor vessel for the RCS loop containing the pressurizer. The break is assumed to open instantaneously. For this break, rapid depressurization of the primary system occurs. Automatic partial cooldown (via the secondary side) is unnecessary due to the rapid depressurization caused by the break.

SIS actuates on receipt of a low pressurizer pressure signal. The most limiting single failure for this event is the loss of one SIS train (i.e., loss of one MHSI pump and one LHSI pump). Because one other train is conservatively assumed to be unavailable due to maintenance or other activity, only two pump trains are available for the event. Four accumulators are assumed to be available, as accumulator maintenance is prohibited during power operation and the downstream accumulator isolation valves are secured open (breakers racked out) to protect against active single failure.

When the RCS pressure falls below the accumulator pressure, fluid from the accumulators is injected into the cold legs. SIS flow injects into the RCS when system startup-time delays have elapsed and primary system pressure falls below the respective shutoff heads of the MHSI and LHSI systems. While some of the ECCS flow bypasses the core and goes directly out of the break, the downcomer and lower plenum gradually refill. During this refill phase, heat is primarily transferred from the hotter fuel rods to cooler fuel rods and structures by radiative heat transfer.

When the lower plenum is refilled to the bottom of the fuel rod heated length, the refill phase ends and the reflood phase begins. The ECCS fluid flowing into the downcomer provides the driving head to move coolant through the core. As the mixture level moves up the core, steam is generated and liquid is entrained. As this entrained liquid is carried into the SGs, it vaporizes because of the higher temperature in the SGs. This causes steam binding, which reduces the core reflooding rate. The fuel rods are cooled and quenched by radiation and convective heat transfer as the quench front moves up the core. Long term recirculation cooling is maintained by the LHSI function of the SIS.

6.3.3.3 NPSH Evaluation

An evaluation of the MHSI and LHSI pumps demonstrates sufficient NPSH is available during postulated DBAs. This evaluation includes the effects of IRWST temperature, sump screen resistance with debris, pump performance, and uncertainties in hydraulic resistances.

IRWST temperatures are calculated using RELAP5/B&W (Reference 16) to determine the mass and energy release, and GOTHIC (Reference 17) to determine the containment and IRWST responses. The IRWST temperatures are calculated conservatively by mixing the condensed liquid in the containment with the IRWST water. The limiting case is the double-ended guillotine (DEG) hot-leg break, Figure 6.3-7—IRWST LOCA Temperature Response. The peak IRWST temperature is calculated to be 230°F.

The SIS pump NPSH evaluation for LBLOCA events is performed using the maximum pump flow head-capacity curves, maximum system resistances, debris laden sump screen resistance, and a reduced IRWST level to account for liquid hold up in the containment. The limiting evaluation of NPSH does not credit containment overpressure. It conservatively assumes the IRWST liquid is at the saturation pressure corresponding to the peak calculated IRWST temperature of 230°F. Simultaneous operation of both the MHSI and LHSI pumps is considered. The increase in IRWST temperature is taken into account for the LBLOCA analysis in 15.6.5. The LBLOCA analysis inherently bounds the SBLOCA analysis.

6.3.4 Tests and Inspections

Refer to Section 14.2 (Test abstract #014, #015, #016, #022, #175, and #177) for initial plant testing. Applicable guidance from RG 1.79 is incorporated in the initial plant testing described in Section 14.2.

Surveillance Requirements 3.5.1, 3.5.2, 3.5.3, and 3.5.4 in Chapter 16 describe the SIS surveillance requirements.

The installation and design of the SIS and IRWSTS provides accessibility for periodic testing and in-service inspection. Sections 3.9.6, 5.2.4, and 6.6 address the pre-service and in-service testing and inspection programs for the SIS.

6.3.5 Instrumentation Requirements

The SIS trains and IRWSTS are monitored and controlled from the main control room through the instrumentation and control systems. The instrumentation and control systems process and display information in the main control room, and actuate the safety injection function as required by plant process safety parameters.

Operator intervention to protect the SIS equipment is required in the event of alarms that indicate unacceptable parameters, such as high bearing oil, motor winding, or motor air temperatures, or loss of suction head. Such conditions alarm or indicate in the control room.

The SIS pumps start automatically on receipt of a safety injection signal, with independent power supply for each train provided by the emergency power supply system. When the permissive P12 is not validated (RCS pressure is at or near that for power operation), the SIS pumps start on the receipt of a low pressurizer pressure signal. When the permissive P12 is validated (RCS pressure indicates reactor shutdown and cooldown in progress), the SIS pumps start on the receipt of a low RCS delta-Psat signal (difference between the RCS hot-leg actual pressure and the RCS hot-leg saturation pressure). In the event a LOCA occurs when permissive P15 is validated (LHSI is in RHR mode with no RCPs in operation), the MHSI pumps start automatically on loss of RCS level. Permissive signals are described in Section 7.2.1.3.

On receipt of a safety injection signal, the motor operated valves in the injection paths receive a signal to open and the hot-leg suction or alternate injection line isolation valves receive a signal to close.

The monitored parameters of the IRWST are water level (for leakage detection and inventory monitoring), water temperature, sump screen differential pressure, and the SIS suction line double (guard) pipe pressure.

I&C for the SIS, as well as its respective permissives, are described in Chapter 7. Applicable guidance from RG 1.47 is incorporated in the design of the SIS I&C described in Chapter 7.

6.3.6 References

1. NUREG-0933, "A Prioritization of Generic Safety Issues," ~~NRC~~U.S. Nuclear Regulatory Commission, September 2007.

2. GL 80-014, "LWR Primary Coolant System Pressure Isolation Valves," ~~NRC~~U.S. Nuclear Regulatory Commission, February 1980.

3. GL 80-035, "Effect of a DC Power Supply Failure on ECCS Performances," ~~NRC~~U.S. Nuclear Regulatory Commission, April 1980.

4. GL 81-021, "Natural Circulation Cooldown," ~~NRC~~U.S. Nuclear Regulatory Commission, May 1981.

5. GL 85-16, "High Boron Concentrations," ~~NRC~~U.S. Nuclear Regulatory Commission, August 1985.

6. GL 86-07, "Transmittal of NUREG-1190 Regarding the San Onofre Unit 1 Loss of Power and Water Hammer Event," ~~NRC~~U.S. Nuclear Regulatory Commission, March 1986.

7. GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," ~~NRC~~U.S. Nuclear Regulatory Commission, June 1989.

8. GL 91-07, "GI-23, 'Reactor Coolant Pump Seal Failures' and Its Possible Effect on Station Blackout," ~~NRC~~U.S. Nuclear Regulatory Commission, May 1991.

9. GL 98-04, "~~NRC~~U.S. Nuclear Regulatory Commission Generic Letter 98-04: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," ~~NRC~~U.S. Nuclear Regulatory Commission, July 1998.

10. BL 80-18, "Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture," ~~NRC~~U.S. Nuclear Regulatory Commission, July 1980.

11. BL 86-03, "Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line," ~~NRC~~U.S. Nuclear Regulatory Commission, October 1986.

12. BL 88-04, "Potential Safety-Related Pump Loss," ~~NRC~~U.S. Nuclear Regulatory Commission, May 1988.

13. BL 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," ~~NRC~~U.S. Nuclear Regulatory Commission, May 1993.

14. BL 01-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," ~~NRC~~U.S. Nuclear Regulatory Commission, August 2001.

15. BL 02-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," ~~NRC~~U.S. Nuclear Regulatory Commission, March 2002.

16. BAW-10164P-A, Revision 6, "RELAP5/ MOD2-BAW – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses," AREVA NP Inc., June 2007.

17. BAW-10252(NP)-A, Revision 0, "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC," Framatome ANP, September 2005.

18. GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," U.S. Nuclear Regulatory Commission, September 2004.

19. ANP-10293, Revision 0, "U.S. EPR Design Features to Address GSI-191," AREVA NP Inc., February 2008.

Table 15.0-60—NRC Generic Letters
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GL #	Subject	Disposition for U.S. EPR
GL-80-19	Resolution of Enhanced Fission Gas Release Concern	This GL is satisfied for the U.S. EPR. Fission gas release at extended burnups is calculated by the fuel performance computer codes COPERNIC, RODEX2 and RODEX3 described in References 3 and 4.
GL-80-35	Effect of a DC Power Supply Failure on ECCS Performance	The U.S. EPR design addresses this concern by providing four independent trains of ECCS. The evaluation of LOCA events, Sections 15.6, Decrease in Reactor Coolant Inventory Events, conservatively assumes one train of MHSI, LHSI and EFW is unavailable because of maintenance, a second train is unavailable because of a single failure and a third train is in the broken cold leg.
GL-83-11	Licensee Qualification for Performing Safety Analysis in Support of Licensing Actions	This GL is satisfied for the U.S. EPR. AREVA is qualified to perform safety analysis as demonstrated by NRC's approval of the methodologies developed by AREVA that are used to evaluate the U.S. EPR.
GL-83-22	Safety Evaluation of 'Emergency Response Guidelines'	This item is addressed by the emergency procedure guidelines (EPGs), Section 13.5, Plant Procedures.
GL-83-32	NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS	The U.S. EPR complies to the requirements of 10 CFR 50.62 as described in Section 15.8, Anticipated Transients Without Scram.
GL-85-06	Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related	The quality assurance requirements for ATWS equipment described in Addendum A-19 of AREVA NP Topical Report ANP-10266, "AREVA NP Quality Assurance Plan for Design Certification of the U.S. EPR" apply to the DAS, Section 7.1.1.3.6, Process Automation System.
GL-85-16	High Boron Concentrations	The U.S. EPR design addresses this concern. The MHSI and LHSI pumps take suction from the IRWST, which does not contain boron concentrations high enough to be susceptible to precipitation. An independent, manually initiated, safety-related Extra Borating System, Section 6.8, provides highly borated injection for maintaining reactivity margin during plant cooldown to cold shutdown. It is designed to avoid crystallization issues, Section 6.8.2, EBS System Description.

Table 15.0-60—NRC Generic Letters
Sheet 2 of 3

GL #	Subject	Disposition for U.S. EPR
GL-86-13	Potential Inconsistency between Plant Safety Analyses and Technical Specifications	The potential for inconsistency between the U.S. EPR TSs and Chapter 15 analyses is avoided because safety analysis evaluated the complete operating domain from power operation to cold shutdown and the TS are based on this safety analysis.
GL-86-16	Westinghouse ECCS Evaluation Models	This issue only applies to the Westinghouse evaluation models, and is not applicable to the U.S. EPR.
GL-88-16	Removal of Cycle-Specific Parameter Limits from Technical Specifications	Fuel cycle specific parameter information is provided in the Core Operating Limits Report.
GL-88-17	Loss of Decay Heat Removal	The U.S. EPR design addresses this concern through the automatic actuation of MHSI on a low RCS loop level signal during non-power operation. The actuation of MHSI is adequate to maintain RCS inventory in the event of the loss of the RHR system, Section 7.3.1.2.1, Safety Injection System Actuation.
GL-93-04	Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies	This letter describes a Westinghouse control system issue. The corresponding U.S. EPR rod control system, the Reactor Control, Surveillance and Limitation System, Section 7.1.1.4.5, is designed to prevent a single failure from causing a loss of function. Moreover, reactivity events such as described in the letter are evaluated in Section 15.4, Reactivity and Power Distribution Anomalies.
GL-97-01	Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations	Control rod ejection is evaluated from a reactivity standpoint in Section 15.4.8. A failure in the reactor vessel head penetration that causes a small break LOCA is bounded by the analyses in Section 15.6.5.2.
GL-98-02	Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions while in a Shutdown Condition	The safety injection system (SIS), which provides the emergency core cooling function for the U.S. EPR, comprise of four supply and return trains, one for each of the reactor coolant system (RCS) loops. Since the SIS does not use a common pump suction header for its emergency core cooling function, a common-cause failure is precluded. Also, design features that result in an inadvertent RCS draindown, such as the spurious opening of the LHSI suction isolation valve during residual heat removal, is discussed in the failure modes and effects analysis (FMEA), Section 6.3, Emergency Core Coolant System.

Table 15.0-60—NRC Generic Letters
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GL #	Subject	Disposition for U.S. EPR
GL~~-~~ 200~~0~~4-02	Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors	The U.S. EPR design incorporates mitigative features to address this concern such as reflective metal insulation and filtering devices described in Section 6.3.2.5, ECCS System Reliability.

14.3 Inspection, Test, Analysis, and Acceptance Criteria

Section 14.3 explains the selection criteria and methods used to develop the U.S. EPR Tier 1 certified design material (CDM) and the inspections, tests, analyses, and acceptance criteria (ITAAC). Tier 1 means the portion of the design-related information contained in a generic FSAR that is approved and certified by the design certification rule (10 CFR Part 52). The design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 information includes:

- Definitions and general provisions.
- Design descriptions.
- ITAAC.
- Significant interface requirements.
- Significant site parameters.

The information in the Tier 1 portion of the FSAR is extracted from the detailed information contained in Tier 2. While the Tier 1 information must address the complete scope of the design to be certified, the amount of design information is proportional to the safety-significance of the structures and systems of the design.

There are two material categories in Tier 1: CDM and ITAAC.

- CDM is the design commitment. CDM is in the form of design descriptions, tables, and figures, and is binding for the lifetime of a facility.
- ITAAC will be used to verify the U.S. EPR as-built features. ITAAC material is in tabular format only and expires at initial fuel loading.

Tier 1 consists of five chapters:

- Chapter 1 (Introduction) provides definitions of terms, a figure legend, a list of acronyms and abbreviations, and general provisions applicable to design descriptions, figures, and ITAAC.
- Chapter 2 (System Based Design Descriptions and ITAAC) provides descriptions of safety-significant design features and the ITAAC verifying those features. Chapter 2 is organized by systems, and those systems are grouped into sections for convenience. Every system included in Tier 2 that is within the scope of CDM is listed in Chapter 2. The applicable portions of systems that are partially within the scope of CDM are also included in Chapter 2. Safety-significant systems outside the scope of CDM are addressed as interface requirements in Chapter 4. Interface requirements for systems that are partially in scope are included in Chapter 2 so the CDM for those systems are in one location.

- Chapter 3 (Non-System Based Design Descriptions and ITAAC) provides CDM not suited to the system design description format of Chapter 2. Material in Chapter 3 addresses security, reliability assurance program (RAP), initial test program (ITP), human factors engineering (HFE), and containment isolation.

- Chapter 4 (Interface Requirements) provides information on safety-significant interface requirements that must be met by site-specific portions of a facility that are not within the scope of CDM. Interface requirements define design features and characteristics so that the site-specific portion of the design conforms to the CDM.

- Chapter 5 (Site Parameters) provides bounding values for safety-significant site parameters that a combined license (COL) applicant referencing the U.S. EPR design will use for site selection. Compliance with these site parameters is verified during the COL application process.

Information presented in Tier 1 contains the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the U.S. EPR design certification has been constructed and will be operated in accordance with the design certification, the provisions of the Atomic Energy Act, and NRC regulations (10 CFR 52.47(b)(1)).

A COL applicant that references the U.S. EPR design certification will provide ITAAC for emergency planning, physical security, and site-specific portions of the facility that are not included in the Tier 1 ITAAC associated with the certified design (10 CFR 52.80(a)). Additionally, a COL applicant that references the U.S. EPR design certification will describe the selection methodology for site-specific SSC to be included in ITAAC, if the selection methodology is different from the methodology described within the FSAR, and will also provide the selection methodology associated with emergency planning and physical security hardware.

14.3.1 Tier 1, Chapter 1, Introduction

Tier 1, Chapter 1 presents definitions, general provisions, a figure legend, and a list of acronyms and abbreviations. The definitions help minimize interpretation issues over the words and phrases used in Tier 1. The general provisions help with ITAAC verification of the configuration of SSC by providing more details on inspections, tests, and analyses that are common to multiple systems. The ITAAC include inspection of the functional arrangement of the system as described in the design description and as shown in the figures. A figure legend and a list of acronyms facilitate the use and interpretation of U.S. EPR design information. The technical terminology used in Tier 1 is consistent with Tier 2 terminology, industry standards, and regulatory documents.

The criteria for selecting definitions include those in Standard Review Plan (SRP) 14.3 (Reference 1) and any other terms in the FSAR that could be subject to interpretation. The selection process for determining which terms are to be defined begins with a review of the terms and definitions in Tier 2 and the guidance in SRP 14.3. Those terms that are important to Tier 1, potentially ambiguous, or unique to Tier 1 are selected.

The criteria for inclusion in the general provisions section includes those items needed to clarify the technical requirements that apply to multiple systems, provide guidance on ITAAC implementation, provide guidance on the interpretation of figures, provide guidance on operational considerations, and specify the U.S. EPR core thermal power level. Selecting the general provisions to be included in Tier 1 involves following the SRP 14.3 (Reference 1) guidance and reviewing Tier 2 against the specific criteria previously listed.

14.3.2 Tier 1, Chapter 2, System Based Design Descriptions and ITAAC

Tier 1, Chapter 2 contains CDM system design descriptions (SDD) and associated ITAAC. This chapter is the result of the process to determine which U.S. EPR design features addressed in Tier 2 should be addressed in the Tier 1 CDM SDDs, interface requirements, and site parameters. The selection process considers the U.S. EPR design philosophy of simple, redundant, and active systems coupled with advanced control technology, which reduces the frequency of transients and improves the reliability of the response to those transients. Given this design philosophy, the process of determining the safety-significant features uses the availability of probabilistic risk assessment (PRA) information to determine the significant design features and performance criteria that lead to safe operation. Using this process allows the top level Tier 1 information to be extracted from the more detailed Tier 2 design information. Tier 1, Chapter 2 provides no technical information not already presented in Tier 2.

The Tier 1 information selection process uses two distinct, parallel approaches: those based on equipment classification and those based on features credited in various analyses. The first approach uses specific equipment classification criteria derived from SRP 14.3, including the system checklists in Appendix C of SRP 14.3 (Reference 1). Examples of equipment selection criteria include ASME BPV Code, Section III (Reference 2), Seismic Category I, and IEEE Class 1E. This selection process provides those safety significant features credited to comply with 10 CFR Parts 20, 50, 52, 73, or 100. In keeping with the SRP guidance, features provided solely for equipment protection are not included in Tier 1 material.

Tier 1 SDDs developed during the first approach address each system identified in Tier 2. The amount of detail included in a Tier 1 SDD for a specific system is a function of the number and safety significance of the system design features. Systems addressed in Tier 2 that have no safety-significant features are listed in Tier 1 as 'No entry for this system.'

The second approach to develop Tier 1 material uses assumptions and insights from key safety and integrated plant safety analyses to identify Tier 1 material. Addressing these assumptions and insights in Tier 1 means the integrity of the fundamental analyses is preserved in the as-built facility referencing the U.S. EPR design. The various review teams for this approach were led by a subject matter expert and included, at a minimum, representatives from engineering integration, PRA, and licensing. The following areas were reviewed for safety-significant design features:

- Design Basis Accidents (DBA) — Analytical input summaries and key assumptions for the safety analyses were reviewed. Also, system engineers performing containment analyses and overpressure protection analyses identified items to be included as DBA safety-significant design features. The results are in Table 14.3-1—DBA Analysis.

- Radiological Protection — The radiological engineering information record that summarizes the design input for radiological analyses was reviewed for safety-significant items. The results are in Table 14.3-2—Radiological Analysis.
- Fire Protection — Fire hazards analyses were reviewed for safety-significant design features. The results are in Table 14.3-3—Fire Protection.
- Flooding Protection — Flooding evaluations were reviewed for safety-significant design features. The results are in Table 14.3-4—Flooding Analysis.
- Anticipated Transient Without Scram (ATWS) —10 CFR 50.62 (the ATWS rule) and the engineering evaluation addressing ATWS were reviewed for safety-significant design features. The results are in Table 14.3-5—ATWS.
- PRA and Severe Accident — The PRA insights report and severe accident analyses were reviewed for safety-significant design features. Using the PRA insights report provided a process to identify non-safety-related features that are safety-significant and otherwise may not have been identified. The results are in Table 14.3-6—PRA and Severe Accident Analysis.
- Licensing — Three Mile Island (TMI) items from 10 CFR 50.34(f) and high-priority generic safety issues (GSI) items from NUREG-0933, Appendix B were reviewed for safety-significant design features relevant to the U.S. EPR design. The items were then compared to the other Section 14.3 tables for redundancy. Items not already addressed by another Section 14.3 table or not already addressed by other Tier 1 criteria are listed in Table 14.3-7—Licensing.

In addition to identifying the safety-significant features, the tables developed during the second approach (team reviews of analyses) list the Tier 2 section that describes the identified design feature. As part of the Tier 1 development process, roadmaps were also created to maintain consistency between Tier 1 and Tier 2 material. Additionally, the information contained in the Tier 2, Section 14.3 tables was verified to be included in Tier 1, and Tier 1 material related to testing was verified to be consistent with the initial test program in Tier 2, Section 14.2.

The U.S. EPR systems are listed in Table 14.3-8—ITAAC Screening Summary. Systems within the scope of Tier 1 or that contain ITAAC are identified in the table. Conceptual systems only consisting of interface requirements are not considered within the scope of Tier 1. The Kraftwerks Kennzeichen System (KKS) codes are also listed in Table 14.3-8 because Tier 1 equipment tags use the KKS identification system.

The commitments listed in the Tier 1 ITAAC tables will be verified to satisfy the acceptance criteria using the inspection, test, or analysis listed. If the as-built item satisfies the acceptance, then the ITAAC is complete. For items not satisfying the acceptance criteria, corrective actions will be taken to resolve the issue.

14.3.2.1 Content of Tier 1 System Design Descriptions

The content of the Tier 1 SDDs for systems and structures reflects the graded approach previously approved for other certified designs, as described in SRP 14.3. This graded approach results in only the top level design features that are safety significant being included in the Tier 1 SDDs. The level of detail provided similarly reflects a graded approach, with the detail provided commensurate with the safety significance of the system. The SDDs constitute the CDM and consist of descriptive material, tables, and figures.

The checklists provided in Appendix C of SRP 14.3 were used to guide the content of the U.S. EPR Tier 1 SDDs. Generally, the following information is included:

- A brief statement of the purpose of the system or structure.
- A listing of the safety-significant functions.
- System location.
- Key design features.
- Classifications (e.g., ASME Code, seismic category, IEEE Class 1E, environmental qualification).
- Minimum controls and displays.
- 1E power requirements.
- Interface requirements.

The SDDs generally contain no numerical values. Numerical values listed in the tables within Chapter 2 are provided to be used as the basis for ITAAC acceptance criteria and appear in the associated ITAAC acceptance criteria that verify the as-built facility. To the extent practical, standardized wording is used in the SDDs to avoid confusion.

The following types of information presented in Tier 2 are not addressed in Tier 1 for the indicated reasons:

- Proprietary and safeguards information because the Office of the Federal Register requires that information incorporated into the design certification rule is publicly available.
- Portable equipment and replaceable items because the certified design descriptions focus on the permanent physical characteristics of the as-built facility and portable equipment, and replaceable items are controlled through other operational programs.
- Programmatic requirements related to operations, maintenance, and other programs are not detailed in the Tier 1 design descriptions.

- Programmatic aspects of the design and construction processes, such as worker selection, qualification, and training, are not covered in the CDM.

- Operational issues, such as procedures and training, are not design features and therefore are not presented in Tier 1.

- Integrated test requirements are presented in Tier 1, but specific details of the initial test program are not presented in Tier 1. Details of the initial testing program are presented in Tier 2, Section 14.2.

- The use of codes and standards (with the exception of the ASME Code) are minimized in Tier 1 design descriptions because the Tier 1 SDDs are intended to stand alone. Specific information needed from external documents is included in the applicable Tier 1 chapter when necessary.

14.3.2.2 Selection Criteria for ITAAC

An ITAAC table is provided for each Tier 1 system that has a design description. The ITAAC table defines the activities to be performed to verify that the as-built system conforms to the design features contained within the design description, as well as the acceptance criteria for those activities.

The following items are considered when developing the ITAAC entries:

- Section 1 of the SDDs provides a brief summary of the Tier 1 functions. Commitments of plant features begin in Section 2 and are in each subsequent section of the SDD.

- ITAAC are only intended to verify the as-built configuration of important design features and performance characteristics described in the design descriptions. Therefore, there are no ITAAC for features not addressed in the design description.

- Each U.S. EPR system that has a design description also has associated ITAAC. The scope of the ITAAC corresponds to the scope of the design descriptions.

- A single inspection, test, or analysis may verify multiple provisions in the certified design description.

- The inspections, tests, and analyses must be completed and the acceptance criteria verified prior to the initial loading of fuel (10 CFR 52.103).

14.3.2.3 Content of ITAAC

ITAAC tables for the U.S. EPR use the standard format in Appendix D of SRP 14.3. The ITAAC tables have columns for design commitments; inspections, tests, and analyses; and acceptance criteria. Each design commitment in the left-hand column has an associated inspection, test, or analysis requirement in the middle column with the applicable acceptance criteria listed in the right-hand column.

Column 1 (Design Commitment) defines the specific commitment extracted from the SDD features.

Column 2 (Inspections, Tests, and Analyses) defines the specific method the licensee will use to demonstrate that the specific design commitment in Column 1 has been met. The methods used are inspection, test, analysis, or a combination of the three:

- Inspections are used when verification can be done by visual observations, physical examinations, walkdowns, or by reviewing records that are based on observations or examinations. The inspections required for basic configuration walkdown follow the general provisions in Tier 1, Section 1.2.
- Tests mean that either operating or establishing specified conditions to evaluate the performance of the as-built structures, systems, or components. In addition to testing final and installed equipment, examples of alternative testing methods include factory testing, test facility testing, and laboratory testing. Testing can also include type testing such as might be performed to demonstrate qualification to meet environmental requirements.
- Analysis is used when verification can be done by calculation or engineering evaluation of the as-built SSC.

For the methods used to demonstrate commitment satisfaction, supporting details are provided in Tier 2. The initial test program is described in Section 14.2 of Tier 2 and covers both visual inspections and tests. The details in Tier 2 are not referenced in Tier 1 CDM and are not part of the certified design.

Column 3 (Acceptance Criteria) depends upon the design feature to be verified and the method used for the verification. Acceptance criteria are objective and clear to avoid confusion over whether or not acceptance criteria have been satisfied. Some acceptance criteria contain numerical values that are not specifically identified in the Tier 1 design description or the ITAAC table design commitments column. This is acceptable because the design description defines the important design feature that needs to be included in the CDM, whereas the numerical value is a measurement standard that determines if the feature has been provided.

14.3.3 Tier 1, Chapter 3, Non-System Based Design Descriptions and ITAAC

The format and selection process for Tier 1, Chapter 3 is similar to Tier 1, Chapter 2 in that it includes CDM and ITAAC tables. Tier 1, Chapter 3 addresses the following non-system based topics:

- Section 3.1 – Security.
- Section 3.2 - Reliability assurance program (RAP).
- Section 3.3 - Initial testing program (ITP).
- Section 3.4 - Human factors engineering (HFE).
- Section 3.5 - Containment isolation.

14.3.4 Tier 1, Chapter 4, Interface Requirements

Interface requirements are items to be met by the site-specific portions of a facility that are not within the scope of the certified design. The site-specific portions of the design are those that depend on site characteristics. Interface requirements define the design features and characteristics that demonstrate that the site-specific portion of the design conforms to the certified design. Interface requirements comply with 10 CFR 52.47(a)(26) requirements.

14.3.5 Tier 1, Chapter 5, Site Parameters

Tier 1, Chapter 5 defines safety-significant site parameters that are the basis for the standard plant design presented in the U.S. EPR design certification application. The list of site parameters follows the suggested list contained in SRP 14.3 and corresponds with the requirements for site parameter information contained in 10 CFR 52.47(a)(1). Compliance with these site parameters is verified during the COL application process, so no ITAAC are necessary for site parameters.

14.3.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Nuclear Regulatory Commission, March 2007.

2. ASME Boiler and Pressure Vessel Code, Section III, "Rules of Construction of Nuclear Facility Components, "Class 1, 2, and 3 Components, The American Society of Mechanical Engineers, 2004, (No Addenda).