GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

January 22, 1993

Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule

Dear Chet:

Enclosed are ABWR SSAR markups addressing: open items 6.2.1.6-1, 6.2.1.6-2, 6.2.4.1-1, 6.2.4.1-2, 6.2.4.1-3 and 6.5.1-2; and confirmatory items 6.5.1-2. Also included is an additional change in Subsection 6.2.1.2-2 requested by the staff.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Jack Duncan (GE)
Norman Fletcher (DOE)

See attached distribution

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consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Main Steam Safety/Relief Valves

SRV lift in a main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS and SRV discharge piping system for the period from opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the main steam and discharge piping following the relatively rapid opening of the SRV cause this piping to vibrate.

The analysis of the MS and discharge piping transient due to SRV discharge consists of a stepwise time-history solution of the fluid flow equation to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of ASME Code flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves in a MS line is assumed in the analysis because simultaneous discharge is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum-change, and fluid-friction terms.

The method of analysis applied to determine response of the MS piping system including the SRV discharge line, to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction thus causing momentary reactions. The resulting loads on the SRV, the main steamline, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. In accordance with Tables 3.9-1 and 3.9-2, the Code stress limits for service levels corre- sponding to load combination classification as normal, upset, emergency, and faulted are applied to the main steam and discharge pipe.

3.9.3.3.2 Other Safety/Relief Valves

An SRV is identified as a pressure relief valve or vacuum breaker. SRVs in the reactor components and subsystems are described and identified in Subsection 5.4.13.

The operability assurance program discussed in Subsection 3.9.3.2.5 applies to safety/relief valves. The qualification of active relief valves is specifically outlined in Subsection 3.9.3.2.5.1.2.2.

ABWR safety/relief valves (safety valves with auxiliary actuating devices and pilot operated valves) are designed and manufactured in accordance with the ASME Code, Section III, Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000, and NB-3500 (pilot operated and power actuated pressure relief valves).

The design of ABWR SRVs incorporates SRV opening and pipe reaction load considerations required by ASME III, Appendix O, and including the additional criteria of SRP, Section 3.9.3, Paragraph II.2 and those identified under Subsection NB-3658 for pressure and structural integrity. Safety/relief valve operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both in compliance with the requirements of SRP Subsection 3.9.3.

3.9.3.3.3 Rupture Disks

There are no rupture disks in the ABWR plant design, that must function during and after a dynamic event (SSE including other RBV loads).

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The effect of consecutive valve actuation is included by assuming SRV pipe temperature to be 300 degree F before valve actuation. This assumption induces higher loads on the SRV pipes because the steam does not condense imediately after the valve is opened.

(4) upper drywell volume = 194,000 ft^3;

(5) the air volume ratio (Wetwell/Drywell) = 0.81.

The vacuum breaker size is characterized by the ratio $A/\sqrt{k}$, where A is the actual flow area of the vacuum breaker and k its pressure loss coefficient. When $A/\sqrt{k} \geq 8.3\ ft^2$, the calculated negative pressure differential is 1.46 psid between the wetwell and drywell. The pressure-time histories are shown in Figure 6.2-17. Thus a WDVBS effective area of 8.3 ft^2 is adequate to satisfy the drywell-to-wetwell negative design pressure requirements of 2.0 psid.

With the WDVBS size determined above, the PCV negative design pressure on the drywell side is checked. This analysis utilizes the wetwell spray in order to minimize the wetwell/drywell pressure. Figure 6.2-18 shows the pressure time histories for wetwell and the drywell. It should be noted that no drastic depressurization occurs because the WDVBS has sufficient size to prevent the initial rapid depressurization in the drywell. In addition, the wetwell airspace contains a large amount of air and the wetwell spray capacity is less than 15% of the drywell break flow capacity. The lowest containment pressure, thus the maximum PCV-to-reactor building negative pressure, occurs during the steady-state end of the transient. The final pressure becomes lower than the initial containment pressure because the drywell/wetwell sprays decrease the vapor partial pressure and cool the air in the PCV as the WDVBS equalizes the pressure in the drywell to that in the wetwell.

The maximum negative pressure is 0.8 psig for the drywell and the wetwell, which satisfies the PCV negative design pressure requirement of 2.0 psig.

With a typical vacuum breaker diameter of 20 inches and a loss coefficient, k, of 3, The required number of wetwell-to-drywell vacuum breakers is 8 which considers one single failure in the WDVBS.

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6.2.1.1.4.2 Wetwell-to-Reactor Building Negative Differential Pressure

Since the WDVBS meets the PCV negative design pressure requirement on the drywell, the PCVBS must satisfy the PCV negative design pressure requirement on the wetwell. The specific requirements are:

The wetwell-to-reactor building negative pressure shall be less than 2.0 psig to protect the PCV liner in the wetwell.

The following wetwell depressurization events which may result in the were considered:

(1) drywell and wetwell spray actuation during normal operation;

(2) wetwell spray actuation subsequent to stuck open relief valve; and

(3) drywell and wetwell spray actuation following a LOCA.

The limiting transient corresponds to the wetwell spray actuation subsequent to stuck open relief valve. Figure 6.2-19 illustrates the variations in containment temperature and pressure during the event.

The effect of SRV discharge to the suppression pool is to heat the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure, the WDVBS allows the flow of air from the wetwell to the drywell, thereby pressurizing both volumes. The wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal from the continued pool cooling and wetwell spray. The wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace slowing down wetwell depressurization rate. The pressure differential between the drywell and the wetwell is maintained constant at the hydrostatic head

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Vacuum breakers are intended to be swing check type valves which open passively due to negative differential pressure (WW airspace pressure greater than the DW pressure) across the valve disk, and require no external power to actuate them. These valves are installed horizontally locating in WW airspace, one valve per penetration (through pedestal wall) opening into lower drywell. Position location of these valves, both axially and azimuthally, are shown in Figures 1.2-3c and 1.2-13k.

In view that these vacuum breaker valves are located in the wetwell airspace, appropriate design features will be provided protecting these valves from loads due to pool swell during a loss-of-coolant accident. These features would include providing solid catwalk area (below the valves) of sufficient measure assuring complete structural shielding from possible direct pool swell loads, as well as protection from water fallback. The solid catwalk area structure will be designed for pool swell impact loads determined based on the methodology approved for Mark II/III designs.

3.8.2.5 and 3.8.3.5. Since the internal structures are not subject to external design or tornado winds, they are not designed for these loads.

Localized pipe forces, pool swell and SRV actuation are asymmetric pressure loads which act on the containment and internal structures. For magnitudes of pool swell and SRV loads, see Subsection 6.2.1.1.5.

The loads associated with embedded plates are concentrated forces and moments which differ according to the type of structure or equipment being supported. Earthquake loads (OBE and SSE) are inertial loads caused by seismic accelerations. The magnitude of these loads is discussed in Section 3.7.

6.2.1.1.7 Containment Environment Control

The drywell ventilation system maintains temperature, pressure and humidity in the containment and its subcompartments at the normal design conditions. The safety-related containment heat removal systems described in Subsection 6.2.2 maintain required containment atmosphere conditions during accidents. Since the loss of the drywell ventilation system does not result in exceeding the design environmental conditions for the safety-related equipment inside containment, the drywell system is not classified as safety-related.

6.2.1.1.8 Post-Accident Monitoring

Refer to Subsections 6.2.1.7, 7.2, 7.3, 7.5, 7.6.1.2, and 7.6.1.11 for discussion of instrumentation inside the containment which may be used for monitoring various containment parameters under post-accident conditions.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

The design of the containment subcompartments is based upon the postulated DBA occurring in each subcompartment.

For each containment subcompartment in which high energy lines are routed, mass and energy release data corresponding to a postulated double ended line break are calculated. The mass and energy release data, subcompartment free volumes, vent path geometry and vent loss coefficients are used as input into an analysis to obtain the pressure/temperature transient response for each subcompartment.

6.2.1.2.2 Design Features

The upper drywell, lower drywell and wetwell subcompartment volumes are covered in depth in Subsection 6.2.1.1. The remaining containment subcompartment volumes are:

(1) Drywell Head Region

The drywell head region is covered with removable steel head which forms part of the containment boundary. The drywell bulkhead connects the RPV flange to the containment and represents the interface between the drywell head region and the drywell.

The DBA for the drywell head region is the double ended circumferential break of the 6- inch RPV head spray line of the RWCU system at the connection to the RPV head nozzle. The other high energy line in the drywell head region is the 2-in. main steam vent line. The RPV head spray line is chosen as the DBA for this subcompartment due to the higher mass and energy release rates from a postulated break of this line.

(2) Reactor Shield Annulus

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The reactor shield annulus exists between the reactor shield wall (RSW) and the RPV. The reactor shield wall is a concrete cylinder surrounding the RPV and is supported by the reactor pedestal.

The annulus surrounding the RPV is sealed at the top of the RSW by a blowout panel in the insulation that is assumed to open instantaneously following a postulated break of a high energy line inside the annulus.

Several high energy lines extend from the

RPV through the reactor shield wall. There are penetrations in the shield wall for other piping, vents, and instrumentation lines. The 12-in. feedwater pipe is the DBA for the reactor shield annulus subcompartment.

6.2.1.2.3 Design Evaluation

The peak differential pressures do not exceed the design differential pressures for either case described above.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

The environmental condition created by any high energy line break (HELB) are analyzed according to Regulatory Guide 1.89. The first step in such analysis is to calculate the mass and energy release rate from the HELB.

Figure 6.2-22 shows the break flow rate and specific enthalpy for the feedwater line break flow coming from the feedwater system side. Figure 6.2-23 shows the same information for the feedwater line break flow coming from the RPV. Figures 6.2-24 and 6.2-25 show the same information for the main steam line break flow with two-phase blowdown starting when the collapsed water level reaches the main steam line nozzle and when t = 1.0 second.

When the break size is small and the reactor pressure stays at approximately 1060 psia, the critical flow table (such as in Reference 2) may be used. If the long term performance is required or the break size is in the intermediate break range, then the reactor pressure does not stay constant. In this case, the transient is analyzed by using GE-developed computer codes to determine the mass and energy release rate based on References 1, 2 and 3.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (PWR)

(Not Applicable)

6.2.1.5 Maximum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR)

(Not Applicable)

6.2.1.6 Testing and Inspection

6.2.1.6.1 Preoperational Testing

Preoperational testing and inspection programs for the containment and associated structures, systems and components are described in Subsections 3.8.1.7, 3.8.2.7, 3.8.3.7, 6.2.6 and Chapter 14. These programs demonstrate the structural integrity and desired leak tightness of the containment and associated structures, systems, and components.

6.2.1.6.2 Post-Operational Leakage Rate Test

For descriptions of the containment integrated leak rate test (ILRT) and other post-operational leakage rate tests (10CFR50, Appendix J, Tests Type A and B) see Subsection 6.2.6.

Accessible portions of the WDVBS will be visually inspected at each refueling outage to determine that they are free of foreign debris. The WDVBS valves which are remote manually testable from the control room will be tested manually. The maximum allowable leakage for each valve shall be per ASME Section XI, Subsection IWV-3420.

6.2.1.6.3 Design Provisions for Periodic Pressurization

In order to assure the capability of the containment to withstand the application of peak accident pressure at any time during plant life, for the purpose of performing integrated leakage rate tests, close attention has been given to certain design and maintenance provisions. Specifically, the effects of corrosion on the structural integrity of the containment have been minimized by the use of stainless steel liner in the suppression pool

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(2) Reactor Shield Annulus

The reactor shield annulus exists between the reactor shield wall (RSW) and the RPV The reactor shield wall is a concrete cylinder surrounding the RPV. The reactor shield wall is supported by the reactor pedestal and extends to a height 0.1 m below the containment top slab.

Several high energy lines extend from the RPV through the reactor shield wall. There are penetrations for other piping, vents, and instrumentation lines and personnel access holes in the shield wall.

A double ended high energy line pipe break inside the annulus region was postulated and analyzed to determine transient pressure loading inside the annulus region. Flow area between annulus and the drywell region comprised of clearance area between top of the shield wall and containment top slab, and the area of penetration door openings. Break at the RHR shutdown cooling suction nozzle was the DBA for the reactor shield annulus subcompartment pressurization.

6.2.1.2.3 **Design Evaluation**

The reactor shield wall structure, and the reactor pressure vessel and its internals design considered and accounted for the transient pressure loads due to the DBA pipe break inside the reactor shield annulus.

SECTION 6.2

CONTENTS (Continued)

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6.2.4.3.4 Evaluation of Containment Purge and Vent Valves Isolation Barrier Design

6.2.4.3.5 Evaluation of Simultaneous Venting of Drywell and Wetwell

6.2.4.3.6 Evaluation of Containment Purge Supply Against Criterion 54

Influent and effluent lines of this group are isolated by automatic or remote-manual isolation valves located as close as possible to the containment boundary.

6.2.4.3.2.4 Evaluation Against Regulatory Guide 1.11

Instrument lines that connect to the RCPB and penetrated the containment have 1/4-inch orifices and manual isolation valves, in compliance with Regulatory Guide 1.11 requirements.

6.2.4.3.3 Evaluation of Single Failure

A single failure can be defined as a failure of a component (e.g., a pump, valve, or a utility such as offsite power) to perform its intended safety functions as a part of a safety system. The purpose of the evaluation is to demonstrate that the safety function of the system will be completed even with that single failure. Appendix A to 10CFR50 requires that electrical systems be designed specifically against a single passive or active failure. Section 3.1 describes the implementation of these standards as well as General Design Criteria 17, 21, 35, 38, 41, 44, 54, 55 and 56.

Electrical as well as mechanical systems are designed to meet the single-failure criterion, regardless of whether the component is required to perform a safety action. Even though a component, such as an electrically-operated valve, is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure if the system component changes state or fails. Electrically-operated valves include valves that are electrically piloted but air operated, as well as valves that are directly operated by an electrical device. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed, regardless of whether the loss of a safety function is caused by a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

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6.2.4.4 Test and Inspections

The containment isolation system is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested remote-manually from the control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

Air-testable check valves are provided on influent emergency core cooling lines of the HPCF and RHR systems whose operability is relied upon to perform a safety function.

A discussion of testing and inspection of isolation valves is provided in Subsection 6.2.1.6. Instruments are periodically tested and inspected. Test and/or calibration points are supplied with each instrument. Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals at least once every 3 months.

6.2.5 Combustible Gas Control in Containment

The atmospheric control system (ACS-T31) is provided to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. The flammability control system (FCS-T49) is provided to control the potential buildup of oxygen from design-basis radiolysis of water. The objective of these systems is to preclude combustion of hydrogen and damage to essential equipment and structures.

6.2.5.1 Design Bases

Following are criteria that serve as the bases for design:

(1) Since there is no design requirement for the ACS or FCS in the absence of a LOCA and there is no design-basis accident in the ABWR that results in core uncovery or fuel failures, the following requirements mechanistically assume that a LOCA

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6.2.4.3.4 Evaluation of Containment Purge and Vent Valves Isolation Barrier Design

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Protection of the containment purge system CIV's from the effects of flood and dynamic effects of pipe breaks will be provided in accordance with Sections 3.4 and 3.6. The CIV's are air-operated with pilot DC solenoid valve. The power to the DC solenoid valve is supplied from the DC distribution system to the demultiplexer from the valve. Both the supply and return lines for the DC are fused at the multiplexer so that faults are isolated and do not propagate back up into the portions of the DC system common with other systems. This is also discussed in the Fire Hazard Analysis in Section 9A.5.

6.2.4.3.5 Evaluation of Simultaneous Venting of Drywell and Wetwell

OI 6.2.4.1-2

The large (550 mm) purge and vent lines for the ACS, shown in Figure 6.2-39 are not used for purge or venting during normal reactor operation. The isolation valves in these lines are normally closed, they fail in the closed position, they receive an automatic closure signal in the event of a LOCA and they are not needed for pressure control of the containment during normal operation. Administrative controls are used to prevent opening of these valves except at the beginning and end of an operating cycle.

Pressure control of the containment during operation is maintained by a single, small (50 mm) nitrogen supply line, and a single, small (50 mm) vent line. The supply line is divided and provides makeup nitrogen to both drywell and wetwell. The small vent line is attached to the 550 mm drywell purge exhaust line and bypasses the closed 550 mm valve (F004). There is no equivalent vent line from the wetwell. Therefore, the drywell and wetwell are not vented simultaneously during operation and the system has only one supply and one exhaust line as required by BTP CSB 6-4.

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6.2.4.3.6 Evaluation of Containment Purge System Against Criterion 54.

The containment purge system has redundant CIV's each powered from independent electrical division. The CIV's are arranged such that any single failure will not compromise the integrity of the containment. These valves are designed to fail in closed position upon loss of air or loss of electric power to the pilot solenoid valve. With the exception of the makeup valves (50 mm), all containment purge system CIV's are in closed position during normal reactor operation. The purge and vent valves are open only during the inerting and de-inerting modes. All containment purge system CIV's automatically close upon receipt of containment isolation signal. Also, these valves are outside containment and are accesssible should manual actuation be required. Since this arrangement has adequate redundancy, and independence and is not unduely vulnerable to common mode failures, it is not necessary to have redundant and independent CIV's as would be required by Criterion 54.

SECTION 6.5

CONTENTS (CONTINUED)

TABLES

ILLUSTRATIONS

6.5.1.7 Operability and Effectiveness

Efficiency in the usual sense, can not be measured for adsorption systems. Adsorption is time dependent and therefore instantaneous contaminant-removal efficiency is meaningless. True efficiency tests are run on small, representative samples (test cannisters) of the adsorbent using a radioactively tagged tracer gas having similar properties and composition of those of the contaminant of interest (e.g., radioactive elemental iodine or methyl iodide).Because of the difficulty in handling radioactive materials, this type of test is generally not made in the field. The in-place field tests of installed systems are leak tests only. The iodine removal efficiency tests are carried out in a laboratory duplicating the field conditions as closely as possible.

The double filter train design for the SGTS depends on STATIONARY components for normal (Routine) and accident operation. The pre-filter assembly is filled with glass fibers as are the pre and after HEPA filters. The charcoal iodine adsorber bed is located between the HEPA filters. All are located in a welded housing making up the filter train. The redundant active space heaters and fans operate only in the standby mode of the SGTS to dry the charcoal and maintain low relative humidity in the sealed train. READINESS FOR DESIGN OPERATION IS ASSURED BY EFFECTIVE SURVEILLANCE TESTS.

The filter train availability depends on the stationary components replacement. The filter fiber glass sections are modularised for ease in handling. The charcoal is replaced by dumping old charcoal from below the bed and refilling with new charcoal from above. The integrity of the charcoal bed structure is maintained by limiting the moisture content of the charcoal in standby. The charcoal bed is oversized to reduce heating and weathering or aging effects. The bed has 795 Kgs. of charcoal and is 150% thick over the calculated 335 Kgs.required for adequate adsorber saturation and combustion protection.

Per R.G. 1.52 section 4d, each filter train should be operated at least 10 hours per month, with the heaters on, in order to reduce the buildup of moisture on the adsorbers and HEPA filters. The flow element in the filter train flow path and related recorders supply the operating and standby time measurement to assure timely surveillance testing. Charcoal penetration tests are conducted after 720 hours of system operation. Penetration and bypass leakage tests are run every 18 months for the systems maintained in a standby status and following painting, fire or chemical release in the service area and after adsorbent replacement. Surveillance includes functional operation and pressure drop measurements. Technical specifications are satisfied and a single failure for any stationary component is very remote.

The SGTS may be used during de-inerting of the primary containment prior to plant shutdown. Of all the routine operational use of the SGTS, the more likely, though still infrequent, potential use of SGTS is during de-inerting. Because of the high availability of the ABWR, de-inerting and the potential use of SGTS during de-inerting will occur primarily at the end of the fuel cycle. In this way, HEPA filter and charcoal adsorber effectiveness will be tested, and the filter and/or charcoal replaced, if necessary, before the plant returns to full power.

General Electric reviewed the data obtained from operating power plants. It is GE's opinion that an effective surveillance testing and prompt stationary parts replacement is an effective way to assure the availability of the SGTS for the designed operation.

The data for Perry Nuclear power plant which has five filter trains with activated charcoal - two in M15 Annulus Exhaust Gas Treatment System (AEGTS) which operate continuously and three in the M40 Fuel Handling Building (FHB) ventilation system was requested. The surveillance testing results of two systems one each from AEGTS-M15 and FHB-M40 were provided. The M15 data shows that the charcoal bed replacement was necessitated after nearly four years of continuous operation and bypass failure of the HEPA filters for train B occured only once in six years of operation. The M26 data shows that in five years time only once charcoal bed had to be replaced for deluge. Current ABWR SGTS does not have an automatic deluge system and an inadvertent operation of the deluge system is unlikely.

The SGTS data from 1971 to 1991 for Quad cities Nuclear power plant was reviewed. For train B, the charcoal bed replacement was needed in 1979, 1983 and 1987. The bypass leakage occurred rarely and HEPA filter replacement was needed. The train A needed the the charcoal bed replacement in 1984 and 1990.

The availability and reliability of the SGTS to perform the designed function depend on an effective surveillance testing and the prompt replacements of the inefficient parts. Should the LOCA occur when the SGTS is in between surveillance testing period with the probability that the next surveillance test may indicate the need of the charcoal bed replacement, the process radiation monitoring system will alarm if there is any radiation leaking out and the operator can switch to second (redundant) SGTS filter train. Probability of both the SGTS filter trains failing at the same time is very remote.