3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The MSSVs also provide protection against overpressurization of the Reactor Coolant Pressure Boundary by providing a heat sink for the removal of energy from the Reactor Coolant System if the preferred heat sink is not available. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 16.66×10^6 lbs/hr which is approximately 110% of the maximum calculated steam flow of 15.12×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with inoperable safety valves within the limitations of the ACTION requirements on the basis of the reduction in secondary steam flow associated with the required reduction of RATED THERMAL POWER. The acceptable power level (in percent RATED THERMAL POWER) for operation with inoperable safety valves was determined by performing explicit transient analysis.

The events that challenge the relief capacity of the safety valves are those resulting in decreased heat removal capability. In this category of events, a loss of external electrical load and/or turbine trip is the limiting anticipated operational occurrence. A series of cases was analyzed for this transient covering up to two inoperable safety valves on each steam generator. The results of these cases were used to determine a maximum thermal power level from which the event could be initiated without exceeding the primary and secondary side design pressure limits. Thus, the maximum allowed power level as a function of the number of inoperable MSSVs on any steam generator is presented in Table 3.7-1. Note that the power level values presented on this table are the direct inputs into the transient analysis cases and do not include any allowance for calorimetric error. Actual power level reductions must include calorimetric uncertainty and other allowances for operating margin as deemed necessary.

Specific accident analyses for RCCA Bank Withdrawal at Power scenarios demonstrate that adequate safety valve relief capacity exist with up to two inoperable safety relief valves on each steam generator. These cases demonstrate that the reactor trip on OTDT along with the relief from the available main steam safety valves is sufficient to meet secondary side pressurization limits.

For three inoperable main steam safety valves in one or more steam generators, thermal reactor power must be reduced in conjunction with a reduction in the Power Range Neutron Flux High trip setpoint to prevent overpressurization of the main steam system.

The transient analysis assumes that the MSSVs will start to open at the lift setpoint with 3% allowance for setpoint tolerance. In addition, the analysis accounts for accumulation by including a 5 psi ramp for the valve to reach its fully open position. Inoperable MSSVs are assumed to be those with the lowest lift setting. Surveillance testing as covered in Table 3.7-4 allows a ± 3% lift setpoint tolerance. However, to allow for drift during subsequent operation, the valves must be reset to within ± 1% of the lift setpoint following testing.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

Verifying that each Auxiliary Feedwater (AFW) pump's developed head at the flow test point is greater than or equal to the required minimum developed head ensures that the AFW pump performance has not degraded during the cycle, and that the assumption made in the accident analysis remain valid. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code. Because it is undesirable to introduce cold AFW into the steam generators while operating, the test is performed on recirculation flow. This test confirms one point on the pump design curve (head vs flow curve), and is indicative of pump performance. Inservice testing confirms pump operability, trends performance and detects incipient failures by indication of pump performance.

The flow path to each steam generator is ensured by maintaining all manual maintenance valves locked open. A spool piece consisting of a length of pipe may be used as an equivalent to a locked open manual valve. The manual valves in the flow path are: 2AF1, 21AF3, 22AF3, 23AF3, 21AF10, 22AF10, 23AF10, 24AF10, 21AF20, 22AF20, 23AF20, 24AF20, 21AF22, 22AF22, 23AF22, 24AF22, 21AF86, 22AF86, 23AF86, and 24AF86.

LCO 3.0.4.b is not applicable to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

3/4.7.1.3 AUXILIARY FEED STORAGE TANK

The OPERABILITY of the auxiliary feed storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the main steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

If the closure time of the main steam isolation valve (MSIV) during technical specification surveillance testing (performed at a Steam Generator pressure between 800 psig and 1015 psig) is 5.0 seconds or less and the engineered safety feature response time (including valve closure time) for the steam line isolation (MSI) signal (Table 3.3-5) is 5.5 seconds or less, then assurance is provided that MSI occurs within 12 seconds under accident conditions, where Steam Generator pressure may be lower. This method of testing assures that for main steam line ruptures that are initiated from Modes 1-3 conditions that generate a MSI signal via automatic or manual initiation and have adequate steam line pressure to close, the main steam lines isolate within the time required by the accident analysis. Fast closure of the MSIVs is assured at a minimum steam pressure of 170 psia. However, the MSIV will still close via the steam assist function between 118 – 170 psia with slightly greater closure times. For main steam line ruptures that receive an automatic or manual signal for MSI and do not have adequate steam pressure to close the MSIVs (less than 118 psia), the event does not require MSIV closure to provide protection to satisfy design basis requirements (e.g., minimum DNBR remains above the minimum DNBR limit value and peak containment pressure remains below 47 psig).

Testing for SR 4.7.1.5 is performed prior to opening the MSIVs for power operation. During testing, only one valve is opened at a time, with the other three valves remaining closed in the safe position, ensuring isolation capability is maintained. In the event of a steam line rupture, a postulated failure of the tested valve in the open position would result in the blowdown of a single steam generator since the remaining three MSIVs are closed. Failure of a single MSIV to close is consistent with the accident analysis assumptions for a major secondary system pipe rupture (UFSAR Section 15.4.2).

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The component cooling water (CCW) loops are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. The CCW system consists of two safeguards mechanical trains supplied by three pumps powered from separate vital buses. This complement of equipment assures adequate redundancy in the event of a single active component failure during the injection phase and either a single active failure or passive failure during the recirculation phase. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses (i.e., separation of loops as required by the single failure analysis). OPERABILITY of the CCW system exists when both loops are OPERABLE. An OPERABLE CCW loop consists of one mechanical train and one CCW pump.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. The limit of elevation 10.5' Mean Sea Level is based on the elevation above which facility flood control measures are required to provide protection to safety-related equipment.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

BACKGROUND:

The control room emergency air conditioning system (CREACS) provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The OPERABILITY of the CREACS ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The CREACS consists of two independent, redundant trains, one from each unit that re-circulate and filter the air in the Control Room Envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREACS train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and fans. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system. The CREACS is a shared system between Unit 1 and 2 supplying a common CRE. During emergency operation following receipt of a Safety Injection or High Radiation actuation signal, for areas inside the CRE, one 100% capacity fan in each Unit's CREACS will operate in a pressurization mode with a constant amount of outside air supplied for continued CRE pressurization. One fan from each train will automatically start upon receipt of an initiation signal, with one fan in each train in standby. A failure of one fan will result in the standby fan automatically starting.

Each CREACS train has two 100% capacity fans, such that any one of the four fans is sized to provide the required flow for CRE pressurization within the common CRE during an emergency.

A failure of one CREACS filtration train requires manual actions to properly reposition dampers in support of single filtration train operation.

To minimize control room radiological doses, the CREACS outside air is supplied from the non-accident unit's emergency air intake through the cross-connected supply duct (as determined by which unit received an accident signal). Outside air is mixed with recirculated air, passed through each CREACS filter bank (pre-filter, HEPA filter, and charcoal adsorber) and cooling coil, and distributed to the common CRE. The CREACS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding 5 Rem total effective dose equivalent (TEDE).

The CREACS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. CREACS will be manually initiated in the recirculation mode only in the event of a fire outside the CRE, a toxic chemical release, or testing.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room and other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

APPLICABLE SAFETY ANALYSES

The CREACS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREACS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the UFSAR, Chapter 15.

The CREACS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release, as described in UFSAR, Section 6.4. The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels, as described in UFSAR, Section 9.5.

LCO

Two independent and redundant CREACS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of all ventilation trains or from an inoperable CRE boundary could result in exceeding a dose of 5 rem TEDE to the CRE occupants in the event of a large radioactive release.

In order for the CREACS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls are proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition, when a need for CRE isolation is indicated.

A significant contributor to this system's OPERABILITY are the dampers, which are required to actuate to their correct positions. The following dampers are associated with the respective LCO*:

a.1 Fan outlet dampers: 1(2)CAA15 and 1(2)CAA16

These dampers ensure that the flow path for CREACS is operable and are required to open upon CREACS initiation. The associated fan outlet damper will open on fan operation.

a.4 Return air isolation damper: 1(2)CAA17

When aligned for single train operation, the associated air return isolation damper will be administratively controlled in the open position.

b. Other dampers required for automatic operation in the pressurization or recirculation modes:

Control Area Air Conditioning System (CAACS) outside air intake isolation dampers: 1(2)CAA40, 1(2)CAA41, 1(2)CAA43 and 1(2)CAA45

The normally open outside air intake dampers 1(2)CAA40 and inlet plenum isolation dampers 1(2)CAA43 will be closed under emergency conditions. The normally closed outside air intake dampers 1(2)CAA41 and inlet plenum isolation dampers 1(2)CAA45 are normally closed and remain closed under emergency conditions.

* Operability of the CREACS requires that each of the Unit 1 dampers are also operable

Control Area Air Conditioning System (CAACS) exhaust isolation dampers: 1(2)CAA18 and 1(2)CAA19.

These dampers are normally closed and are required to remain closed to prevent inleakage from the outside environment in the event of a toxic release.

Control Room Emergency Air Conditioning System (CREACS) air intake dampers: 1(2)CAA48, 1(2)CAA49, 1(2)CAA50 and 1(2)CAA51

CREACS outside air intake dampers are maintained closed during normal and recirculation operation and are opened automatically upon initiation of CREACS pressurization. The control logic will automatically open the CREACS air intake dampers farthest from the radiation source based upon which Unit's Solid State Protection System (SSPS) or Radiation Monitoring System (RMS) signal is received.

CAACS and CREACS interface isolation dampers: 1(2)CAA14 and 1(2)CAA20

These two dampers are normally open and do not have associated redundant dampers. These dampers serve a boundary function by isolating the CREACS from the CAACS during emergency operation of the CREACS.

Note: Dampers 1(2)CAA5, CAACS recirculation damper will receive an accident alignment signal to ensure proper accident configuration of CAACS. This damper, however, is not required for the OPERABILITY of CREACS as defined in the LCO.

APPLICABILITY

In all MODES and during movement of irradiated fuel assemblies, the CREACS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies, the CREACS must be OPERABLE to cope with the release from a fuel handling accident, involving handling irradiated fuel.

ACTIONS

When one CREACS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to align CREACS for single filtration train operation within 4 hours, and restore the inoperable filtration train to OPERABLE status within 30 days. Single filtration train alignment is only permitted if the Unit with the operable CREACS train is also in Chilled Water LCO 3.7.10.a configuration. Single filtration train alignment is not permitted if in the LCO 3.7.10.c configuration. This ensures required cooling coil heat removal capacity is available. In this Condition, the remaining OPERABLE CREACS train is adequate to perform the CRE occupant protection function. With CREACS aligned for single filtration train operation and with one of the two remaining fans or associated outlet damper inoperable, restore the inoperable fan or damper to OPERABLE status within 72 hours. However, the overall reliability is reduced because a failure in the OPERABLE CREACS train could result in loss of CREACS function. The 72 hours completion time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train components to provide the required capability.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24-hour completion time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day completion time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day completion time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

In MODE 1, 2, 3, or 4, if the inoperable CREACS train or the CRE boundary cannot be restored to OPERABLE status within the required completion time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within the following 30 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREACS train cannot be restored to OPERABLE status, align CREACS for single filtration train operation within 4 hours, or suspend movement of irradiated fuel assemblies. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours. The 72 hours completion time is based on the ability of the remaining train components to provide the required capability.

In MODE 5 or 6, or during the movement of irradiated fuel assemblies, with two CREACS trains inoperable or with one or more CREACS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

SURVEILLANCE REQUIREMENTS

Standby systems should be checked periodically to ensure that they function properly. TS Surveillance Requirement verifies that each fan is capable of operating for at least 15 minutes by initiating flow through the HEPA filter and charcoal adsorbers train(s) to ensure that the system is available in a standby mode. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Filter testing verifies that the required CREACS testing is performed in accordance with the surveillance requirements. The surveillance requirements include testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the surveillance requirements. Filter testing will be in accordance with the applicable sections of ANSI N510 (1975) with the exception that laboratory testing of activated carbon will be in accordance with ASTM D3803 (1989). The acceptance criteria for the laboratory testing of the carbon adsorber is determined by applying a minimum safety factor of 2 to the charcoal adsorber removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.

Actuation testing verifies that each CREACS train starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The control room envelope is considered intact and able to support operation of the CREACS when the emergency air conditioning system is capable of maintaining positive pressure with the control room boundary door(s) closed. Unfiltered air inleakage testing verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

Each CAACS normal air intake ductwork has two radiation detector channels. The two detector channels from Unit 1 and Unit 2 CAACS air intake provide input to common radiation monitor processors. Each radiation monitor processor (one for 1R1B-1/1R1B-2 and one for 2R1B-1/2R1B-2) provides a signal to initiate CREACS in the pressurization mode should high radiation be detected. A minimum of one out of two detectors in either intake will initiate the pressurization mode. With two detector channels inoperable on a Unit, operation may continue as long as CREACS is placed in-service in the pressurization or recirculation mode. Pressurization mode will be initiated after 7 days with one inoperable detector. Radiological releases during a fuel handling accident while operating in the recirculation mode could result in unacceptable radiation levels in the CRE since the automatic initiation capability has been defeated for high radiation due to isolation of the detectors. Therefore, movement of irradiated fuel assemblies or Core Alterations at either Unit will not be permitted when in the recirculation mode.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. The testing verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, CRE boundary is inoperable. Required action allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F. These compensatory measures may also be used as required mitigating actions. Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM

The Auxiliary Building Ventilation System (ABVS) consists of two major subsystems. They are designed to control Auxiliary Building temperature during normal and emergency modes of operation, and to contain Auxiliary Building airborne contamination (by maintaining slightly negative pressure) during Loss of Coolant Accidents (LOCA). The two subsystems are:

1. A once through filtration exhaust system, designed to contain particulate and gaseous contamination and prevent it from being released from the building in accordance with 10CFR20, and

2. A once through air supply system, designed to deliver outside air into the building to maintain building temperatures within acceptable limits. For the purposes of satisfying the Technical Specification LCO, one supply fan must be administratively removed from service such that the fan will not auto-start on an actuation signal; however, the supply fan must be OPERABLE with the exception of this administrative control.

These systems operate during normal and emergency plant modes. Additionally, the system provides a flow path for containment purge supply and exhaust during Modes 5 and 6. Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished or close the equipment hatch and personnel airlocks.

Appropriate filtration surveillances are contained in the Updated Final Safety Analysis Report (UFSAR) Section 9.4.2.4, Test and Inspections. Auxiliary Building exhaust air filtration system functionality is not required to meet LCO 3.7.7.

The ventilation exhaust consists of three 50% capacity fans that are powered from vital buses. The fans are designed for continuous operation, to control the Auxiliary Building pressure at –0.10" Water Gauge with respect to atmosphere.

The ventilation supply consists of two 100% capacity fans that are powered from vital buses, and distribute outdoor air to the general areas and corridors of the building through associated ductwork.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM (cont'd)

AUXILIARY BUILDING VENTILATION ALIGNMENT MATRIX
NORMAL VENTILATION (Normal plant operations)*

Any two of the three exhaust fans and either of the two supply fans:

- The normal alignment is two exhaust fans and one supply fan. During cooler seasons, and with the absence of the system heating coils, it may be required to limit the amount of colder outside air entering the building. In this case, it is acceptable to secure both supply fans from operation and reduce the number of operating exhaust fans to one. There is sufficient capacity with the single exhaust fan to maintain the negative pressure within the auxiliary building boundary.

EMERGENCY VENTILATION (Emergency plant operation)

At least two of the three exhaust fans and either one of the two supply fans.

Note: During a Safety Injection (SI) all three exhaust fans and one of the supply fans will start. This is acceptable and will maintain the boundary pressure while supplying the required cooling to the building. Should access/egress become difficult with the three exhaust fans running, one of the exhaust fans should be secured.

OPERABILITY of the Auxiliary Building Ventilation System ensures that air,
which may contain radioactive materials leaked from ECCS equipment following a LOCA, is monitored prior to release from the plant via the plant vent.
Operation of this system and the resultant effect on offsite and control room dose calculations was assumed in the accident analyses. ABVS is discussed in UFSAR Section 9.4.2.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from the program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they were installed, would have no adverse effect on any safety-related system.

The program for examination, testing and service life monitoring for snubbers is required to be performed in accordance with ASME BPV Code, Section XI or the OM Code and the applicable addenda as required by 10 CFR 50.55a(g) or 10 CFR 50.55a(b)(3)(v), except where the NRC has granted specific written relief, pursuant to 10 CFR 50.55a(g)(6)(i), or authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

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3/4.7.10 CHILLED WATER SYSTEM - AUXILIARY BUILDING SUBSYSTEM

The OPERABILITY of the chilled water system ensures that the chilled water system will continue to provide the required normal and accident heat removal capability for the control room area, relay rooms, equipment rooms, and other safety related areas. Verification of the actuation of each automatic valve on a Safeguards Initiation signal includes actuation following receipt of a Safety Injection signal.

The Auxiliary Building Chilled Water (AB CH) systems can be operated in three possible LCO configurations:

1. Three Chillers Required (LCO 3.7.10.a)
2. Two Chillers Required (LCO 3.7.10.b)
3. Units Cross-Tied (LCO 3.7.10.c)

Three Chillers Required Configuration:

Removal of non-essential heat loads from the chilled water system in the event one chiller is inoperable ensures the remaining heat loads are within the heat removal capacity of the two operable chillers.

Removal of non-essential heat loads from the chilled water system in the event two chillers are inoperable and aligning the CREACs to the maintenance mode ensures the remaining heat loads are within the heat removal capacity of the operable chiller.

During chiller testing, operator actions can take the place of automatic actions

During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components do not have to be considered inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable. This is consistent with Technical Specification 3.8.1.2 which only requires two operable diesel generators.

Two Chillers Required Configuration:

In Two Chiller configuration the analyses demonstrate the system will continue to provide required cooling capability to the control room and safety related areas during normal operation and in the event of an accident in conjunction with a single failure. The analyses for Two Chiller configuration were performed with both trains of Control Room Emergency Air Conditioning (CREACS) operable and one chiller operating in each unit. This configuration accounts for one of the two required chillers in a unit being out of service and an accident and single failure (loss of chiller) in the opposite unit. The restrictions for entering Two Chiller configuration ensure that the heat loads are within the heat removal capacity of the remaining operable chiller. The heat removal capacity of the chiller is based on the service water and outside air temperatures present during the period of November 1st through April 30th. Removal of the Emergency Control Air Compressor (ECAC) from the CH system ensures that the heat load is within the capacity of the remaining chiller.

If one unit is in the Two-Chiller configuration (LCO 3.7.10.b) and the other unit is in the Three Chiller configuration (LCO 3.7.10.a), CREACS single filtration train alignment is allowed with the unit that is in Three Chiller configuration supplying the CREACS train. Additionally, nonessential heat loads must be isolated from the chilled water system on BOTH Units. Alignment of the single CREACS train to the unit in the Two-Chiller configuration is not permitted.

When entering LCO 3.7.10.b, the third chiller must have CH flow isolated to prevent recirculation of cooling water flow through the non-operating chiller. When restoring from LCO 3.7.10.b for transitioning to the Three Chiller configuration, the third chiller may be un-isolated under administrative controls. The administrative controls will require that an operator be dedicated during restoration activities to re-isolate the chiller, if necessary, in the event an accident occurs during the restoration activities.

The loss of the 2 required chillers requires the unit that has the lost the chillers to commence a controlled shutdown (or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies if in MODES 5 or 6 or during the movement of irradiated fuel) and transition the CREACS to single filtration operation with the opposite unit supplying the CREACS train unless both units transition to the Cross-Tied configuration. In the event that the Cross-tied configuration cannot be implemented or the transition to CREACS single filtration train alignment cannot be implemented, both units will commence a controlled shutdown (or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies if in MODES 5 or 6 or during the movement of irradiated fuel).

Required operating conditions will be verified every 24-hours (SR 4.7.10.d) when in the Two-Chiller configuration.

Cross-Tied Configuration:

In Cross-tie configuration the analyses demonstrate the system will continue to provide required cooling capability to the control room and safety related areas during normal operation and in the event of an accident in either unit. The supporting calculations were performed assuming that one of the required chillers is unavailable due to either a single failure or being out of service (two chillers remaining).

The analyses for Cross-Tied configuration determined that both train of CREACS must be operable. With only a single train of CREACS operable, the remaining CREACS cooling coil cannot maintain the control room envelope temperatures within acceptable limits. Therefore, entry into CH Cross-Tied configuration is only allowed when both trains of CREACS are operable. A note is added to TS 3.7.6 Action a to alert operators that CREACS single filtration operation is not permitted if the units are in the CH Cross-tied configuration.

The restrictions for entering the Cross-Tied configuration ensure that the heat loads are within the heat removal capacity of the remaining two operable chillers. The heat removal capacity of the chillers is based on the service water and outside air temperatures present during the period of November 1st through April 30th. Removal of both units' ECACs and both units' non-essential heat loads from the CH system ensures that the heat load is within the capacity of the remaining chillers.

When restoring from LCO 3.7.10.c, the cross-tie valve can be closed under administrative controls. The administrative controls will require that an operator be dedicated during restoration activities to re-open the cross-tie valve, if necessary, in the event an accident occurs during the restoration activities.

If two Chillers become inoperable in Cross-Tie configuration then both units must commence a controlled shutdown (or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies if in MODES 5 or 6 or during the movement of irradiated fuel).

Required operating conditions will be verified every 24-hours (SR 4.7.10.e) when in the Cross-Tied configuration.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. Region 1, with 300 storage positions, is designed to accommodate new fuel with a maximum enrichment of 4.25 wt% U-235. Unirradiated and irradiated fuel with initial enrichments up to 5.0 wt% U-235 can also be stored in Region 1 with some restrictions. These restrictions are stated in TS 3/4.7.12. Region 2, with 1332 storage positions, is designed to accommodate unirradiated and irradiated fuel with stricter controls as compared to Region 1. These controls are also stated in TS 3/4.7.12.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the USNRC letter of April 14, 1978, to all Power Reactor Licensees – OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (Accession # 7910310568) allows credit for soluble boron under other abnormal or accident conditions, consistent with postulated accident scenarios. For example, the most severe accident scenario is associated with the abnormal location of a fresh fuel assembly of 5.0 wt% enrichment which could, in the absence of soluble poison, result in exceeding the design reactivity limitation (k_{eff} of 0.95). This could occur if a fresh fuel assembly of 5.0 wt% enrichment were to be inadvertently loaded into a Region 1 or Region 2 storage cell otherwise filled to capacity. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Calculations for the worst case configuration confirmed that 800 ppm soluble boron (includes an appropriate allowance for boron concentration measurement uncertainty)is adequate to compensate for a mis-located fuel assembly. Subcriticality of the MDR with no movement of assemblies is achieved without credit for soluble boron and by controlling the location of each assembly in accordance with TS 3/4.7.12. Prior to movement of an assembly, it is necessary to verify the fuel storage pool boron concentration is within limit in accordance with TS 3/4.7.11.

Most postulated abnormal conditions or accidents in the spent fuel pool do not result in an increase in the reactivity of either MDR region. For example, an event that results in an increase in spent fuel pool temperature or a decrease in water density will not result in a reactivity increase. An event that results in the spent fuel pool cooling down below normal conditions does not impact the criticality analysis since the analysis assumes a water temperature of 4°C. This assures that the reactivity will always be lower over the expected range of water temperatures.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION (continued)

However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality exceeding limits in both regions. The postulated accidents are basically of three types. The first type of postulated accident is an abnormal location of a fuel assembly, the second type of postulated accident is associated with lateral rack movement, and the third type of postulated accident is a dropped fuel assembly on the top of the rack. The dropped fuel assembly and the lateral rack movement have been previously shown to have negligible reactivity effects (<0.0001 δk). The misplacement of a fuel assembly could result in Keff exceeding the 0.95 limit. However, the negative reactivity effect of a minimum soluble boron concentration of 600 ppm compensates for the increased reactivity caused by any of the postulated accident scenarios. The accident analyses are summarized in the FSAR Section 9.1.2.

The determination of 600 ppm has included the necessary tolerances and uncertainties associated with fuel storage rack criticality analyses. To ensure that soluble boron concentration measurement uncertainty is appropriately considered, additional margin is incorporated into the limiting condition for operation. As such, increasing the minimum required boron concentration in the fuel storage pool to 800 ppm conservatively covers the expected range of boron reactivity worth along with allowances associated with boron measurements.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). The fuel storage pool boron concentration is required to be greater than or equal to 800 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool, until a complete spent fuel storage pool verification has been performed following the last movement of fuel assemblies in the spent fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION (continued)

The Required Actions are modified indicating that LCO 3.0.3 does not apply. Storage of fuel assemblies and the boron concentration in the spent fuel storage pool are independent of reactor operation. Therefore TS 3/4 3.7.11 and TS3/ 4 3.7.12 include the exception to LCO 3.0.3 to preclude an inappropriate reactor shutdown. When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position. If the LCO is not met while moving fuel assemblies in the spent fuel pool while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving fuel assemblies in the spent fuel pool while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

3/4.7.12 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. Region 1, with 300 storage positions, is designed to accommodate new fuel with a maximum enrichment of 4.25 wt% U-235. Unirradiated and irradiated fuel with initial enrichments up to 5.0 wt% U-235 can also be stored in Region 1 with some restrictions. These restrictions are stated in TS 3/4.7.12. Region 2, with 1332 storage positions, is designed to accommodate unirradiated and irradiated fuel with stricter controls as compared to Region 1. These controls are also stated in TS 3/4.7.12.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the USNRC letter of April 14, 1978, to all Power Reactor Licensees – OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (Accession # 7910310568) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the abnormal location of a fresh fuel assembly of 5.0 wt% enrichment which could, in the absence of soluble poison, result in exceeding the design reactivity limitation (k_{eff} of 0.95). This could occur if a fresh fuel assembly of 5.0 wt% enrichment were to be inadvertently loaded into a Region 1 or Region 2 storage cell otherwise filled to capacity, for any of the configurations. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Calculations for the worst case configuration confirmed that 800 ppm soluble boron (includes an appropriate allowance for boron concentration measurement uncertainty) is adequate to compensate for a mis-located fuel assembly. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with TS 3/4.7.12. Prior to movement of an assembly into a fuel assembly storage location in Region 1 or Region 2, it is necessary to perform SR 4.7.11 and either SR 4.7.12.1 or SR 4.7.12.2. In summary, before moving an assembly into the storage racks it is necessary to:

- validate that its final location meets the criticality requirements;
- and since there is a potential to misload the assembly, we need to ensure that the Fuel Storage Pool boron concentration is greater than the minimum required to preclude exceeding criticality limits prior to moving.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.7.12 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL (CONTINUED)

The restrictions on the placement of fuel assemblies within the spent fuel pool in accordance with TS 3/4.7.12, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain < 0.95, assuming the pool to be flooded with unborated water.

This LCO applies whenever any fuel assembly is stored in Region 1 or Region 2 of the fuel storage pool.

The Required Actions are modified indicating that LCO3.0.3 does not apply. Storage of fuel assemblies and the boron concentration in the spent fuel storage pool are independent of reactor operation. Therefore TS 3/4.3.7.11 and TS 3/4.3.7.12 include the exception to LCO 3.0.3 to preclude an inappropriate reactor shutdown. When the configuration of fuel assemblies stored in Region 1 or Region 2 of the spent fuel storage pool is not in accordance with TS 3/4.7.12, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with TS 3/4.7.12. If unable to move fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

The SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with TS 3/4.7.12 in the accompanying LCO.

3/4.7.13 Main Feedwater Isolation Valves (FIVs), Main Feedwater Regulating Valves (FRVs), FRV Bypass Valves (FRVBVs), and Steam Generator Feedwater Pump (SGFP) Turbine Steam Stop Valves

The OPERABILITY of the FIVs (BF13s), FRVs (BF19s), FRVBVs (BF40s) and SGFP turbine steam stop valves (MS43s and RS15s) ensures that the valves will be capable of performing their intended safety function. The safety function of these valves is to rapidly close following: (1) a steam line or feedwater line rupture, thereby limiting the Reactor Coolant System cooldown and limiting the total energy release to the containment; or (2) a feedwater system malfunction, thereby limiting Reactor Coolant System cooldown.

The analysis of excessive RCS heat removal due to a feedwater system malfunction (UFSAR Section 15.2.1 0) assumes full opening of one or more FRVs, due to a control system malfunction or operator error, resulting in a step increase in feedwater flow to one or more steam generators. The analysis assumes a feedwater isolation signal is generated by a high-high steam generator level or a safety injection (SI) signal. Feedwater isolation is assumed to occur as a result of the FIV(s) closing in 32 seconds and failure of the FRV(s) and associated FRVBV(s) to close as a result of the feedwater isolation signal. The trip of the SGFPs is not credited in the feedwater system malfunction analysis.

Rupture of a steam line (UFSAR Section 15.4.2) is analyzed to determine the response of the reactor core and to determine the resulting mass and energy releases. Two separate analyses are performed since conservative assumptions for the core response analysis are different than the conservative assumptions for the mass and energy release analysis.

The core response analysis credits feedwater isolation as a result of the safety injection signal which results in a feedwater isolation signal. For the steam generators with the non-faulted steam line, feedwater isolation is assumed to occur as a result of closure of all FRVs and FRVBVs in 10 seconds following receipt of the SI signal. For the steam generator with the faulted steam line, the FRV is assumed to fail with feedwater isolation achieved by closure of the FIV in 32 seconds following receipt of the SI signal.

The mass and energy release analysis assumes feedwater isolation occurs as a result of the SI signal which generates the feedwater isolation signal. The most limiting case for mass addition to the containment assumes failure of the FRV to close in the loop with the faulted steam line. Feedwater isolation occurs as a result of closure of the FIVs in 32 seconds and tripping of the SGFPs in 7 seconds. Reduction of feedwater flow due to SGFP coast down and closure of the FIV is credited in the containment analysis.

Rupture of a feedwater line between the feedwater stop-check valve and the steam generator (UFSAR Section 15.4.3) is analyzed to determine the response of the reactor core. The feedwater line break could cause either a reactor coolant system (RCS) cooldown or a heat up depending on the size of the rupture.

The RCS cooldown for a feedwater line rupture is bounded by the analysis for a steam line rupture. For the RCS heatup analysis, main feedwater to all steam generators is assumed to stop at the time of the feedwater line rupture due to the feedwater spilling out the break. A feedwater isolation signal is generated as a result of the safety injection signal and is accomplished by closure of the FRVs and FRVBVs in 10 seconds following receipt of the SI signal.

The mass and energy release that would result from a rupture of a main feedwater line inside containment is bounded by the analysis of the rupture of a main steam line.

The APPLICABILITY of this specification is MODES 1, 2, and 3, except when:

- a FIV or FRV and FRVBV valve are closed and deactivated or the main feedwater line is isolated by a closed manual valve; or
- the SGFP turbine steam stop valve is closed and deactivated or the steam supply to the SGFP turbine is isolated, or the SGFP discharge to the steam generators is isolated.

The basis for the mode applicability is that in MODES 1 and 2 there is significant mass and energy in the RCS and steam generators and in MODE 3 there may be significant mass and energy in the RCS and steam generators. With significant mass and energy in the RCS and steam generators, the valves are needed for isolation of the steam generators in the event of a secondary system pipe rupture. The mode applicability is modified by exception based upon the impacted valve or SGFP being placed in its required accident-analysis assumed position or the flow path being isolated such that the credited accident analysis function has already been completed.

The ACTION statements for an inoperable FIV, FRV, or FRVBV require that action must be taken to restore the affected valves to OPERABLE status, or to close or isolate the inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function. The 72 hour action time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the feedwater flow paths. The 72 hour action time is reasonable, based on operating experience.

The ACTION statement for an inoperable SGFP turbine steam stop valve requires that action must be taken to restore the affected valves to OPERABLE status, or isolate the associated steam supply to the SGFP, or isolate the associated SGFP feedwater flow path within 72 hours. When the SGFP steam stop valves are closed or isolated, or the SGFP feedwater flow path is isolated, the required safety function has been completed. The 72 hour action time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour action time is reasonable, based on operating experience.

Inoperable valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day action time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

Separate ACTION entry is allowed for each inoperable valve unless there is a loss of feedwater isolation capability for a flow path. Redundant components in the flow path would perform the feedwater isolation function.

With either (1) a FRV or FRVBV and FIV inoperable, or (2) SGFP turbine steam stop valve (resulting in a loss of SGFP trip function) and FRV or FRVBV inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the FIV, FRV, FRVBV, or SGFP turbine steam stop valve, or otherwise isolate the affected flow path. With both a SGFP turbine steam stop valve and FIV inoperable, the FRV and FRVBV will operate automatically to provide feedwater isolation for the flow path.

If the FIV(s), FRV(s), FRVBV(s) and SGFP turbine steam stop valves cannot be restored to OPERABLE status, or closed, or the flow path isolated within the associated allowed outage time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least HOT STANDBY (MODE 3) within 6 hours and in HOT SHUTDOWN (MODE 4) within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SR 4.7.13.1 verifies that the closure time of each FIV, FRV, FRV bypass and SGFP turbine steam stop valve is within the limit in the Technical Requirements Manual and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the Inservice Testing Program. The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 4.7.13.2 verifies that each FIV, FRV, FRV bypass and SGFP turbine steam stop valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.