2.2.1 REACTOR TRIP SET POINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.6% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 106.6% of RATED THERMAL POWER and a minimum setpoint of 14.6% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 111.6% of RATED THERMAL POWER, which is the value used in the accident analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow.

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Reactor Coolant Flow-Low (Continued)

The low-flow trip setpoint and Allowable Value have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.17 under normal operation and expected transients.

Pressurizer Pressure-High

The pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is approximately 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpont for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The trip setting is sufficiently below the full-load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow.

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Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits. A safety margin is provided which includes allowances for equipment response times, core power, RCS temperature, and pressurizer pressure measurement uncertainties, processing errors, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuring transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

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3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS (Continued)

pressure measurement instrument inaccuracies are already reflected in the Technical Specification acceptance criteria.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The Technical Requirements Manual contains the list of containment isolation valves (except the containment air lock and equipment hatch). Any changes to this list will be reviewed under 10CFR50.59 and approved by the committee(s) as described in the NUQAP Topical Report.

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment isolation valves are used to close all fluid (liquid and gas) penetrations not required for operation of the engineered safety feature systems, to prevent the leakage of radioactive materials to the environment. The fluid penetrations which may require isolation after an accident are categorized as Type P, O, or N. The penetration types are listed with the containment isolation valves in the Technical Requirements Manual.

Type P penetrations are lines that connect to the reactor coolant pressure boundary (Criterion 55 of 10CFR50, Appendix A). These lines are provided with two containment isolation valves, one inside containment, and one outside containment.

Type O penetrations are lines that are open to the containment internal atmosphere (Criterion 56 of 10CFR50, Appendix A). These lines are provided with two containment isolation valves, one inside containment, and one outside containment.

Type N penetrations are lines that neither connect to the reactor coolant pressure boundary nor are open to the containment internal atmosphere, but do form a closed system within the containment structure (Criterion 57 of 10CFR50, Appendix A). These lines are provided with single containment isolation valves outside containment. These valves are either remotely operated or locked closed manual valves.

Locked or sealed closed containment isolation valves may be opened on an intermittent basis provided appropriate administrative controls are established. The position of the NRC concerning acceptable administrative controls is contained in Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," and includes the following considerations:

(1) stationing an operator, who is in constant communication with the control room, at the valve controls,

(2) instructing this operator to close these valves in an accident situation, and

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

The administrative controls for valves 2-SI-651 and 2-SI-709 apply only during preparations for initiation of SDC, and during SDC operations. They are acceptable because RCS pressure and temperature are significantly below normal operating pressure and temperature when 2-SI-651 and 2-SI-709 are opened, and these valves are not opened until shortly before SDC flow is initiated. The penetration flowpath can be isolated from the control room by closing either 2-SI-652 or 2-SI-651, and the manipulation of these valves, during this evolution, is controlled by plant procedures.

The pressurizer auxiliary spray valve, 2-CH-517, can be used as an alternate method to decrease pressurizer pressure, or for boron precipitation control following a loss of coolant accident. When this valve is opened from the control room, either one of the two required licensed (Reactor Operator) control room operators can be credited as the dedicated operator required for administrative control. It is not necessary to use a separate dedicated operator.

The exception for 2-CH-517 is acceptable because the fluid that passes through this valve will be collected in the Pressurizer (reverse flow from the Pressurizer to the charging system is prevented by check valve 2-CH-431), and the penetration associated with 2-CH-517 is open during accident conditions to allow flow from the charging pumps. Also, this valve is normally operated from the control room, under the supervision of the licensed control room operators, in accordance with plant procedures.

A dedicated operator is not required when opening remotely operated valves associated with Type N fluid penetrations (Criterion 57 of 10CFR50, Appendix A). Operating these valves from the control room is sufficient. The main steam isolation valves (2-MS-64A and 64B), atmospheric steam dump valves (2-MS-190A and 190B), and the containment air recirculation cooler RBCCW discharge valves (2-RB-28.2A-D) are examples of remotely operated containment isolation valves associated with Type N fluid penetrations.

MSIV bypass valves 2-MS-65A and 65B are remotely operated MOVs, but while in MODE 1, they are closed with their opening coils removed and stored to satisfy Appendix "R" requirements.

Local operation of the atmospheric steam dump valves (2-MS-190A and 190B), or other remotely operated valves associated with Type N fluid penetrations, will require a dedicated operator in constant communication with the control room, except when operating in accordance with AOPs or EOPs. Even though these valves can not be classified as locked or sealed closed, the use of a dedicated operator will satisfy administrative control requirements. Local operation of these valves with a dedicated operator is equivalent to the operation of other manual (locked or sealed closed) containment isolation valves with a dedicated operator.

The main steam supplies to the turbine driven auxiliary feedwater pump (2-MS-201 and 2-MS-202) are remotely operated valves associated with Type N fluid penetrations. These valves are maintained open during power operation. 2-MS-201 is maintained energized, so it can be closed from the control room, if necessary, for containment isolation. However, 2-MS-202 is deenergized

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3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

The upstream vent valves for the steam generator atmospheric dump valves, 2-MS-369 and 2-MS-371, are opened during steam generator safety valve set point testing to allow steam header pressure instrumentation to be placed in service. When either 2-MS-369 or 2-MS-371 is opened, a dedicated operator in continuous communication with the control room is required.

The determination of the appropriate administrative controls for these containment isolation valves included an evaluation of the expected environmental conditions. This evaluation has concluded environmental conditions will not preclude access to close the valve, and this action will prevent the release of radioactivity outside of containment through the respective penetration.

The containment purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Such a demonstration would require justification of the mechanical operability of the purge valves and consideration of the appropriateness of the electrical override circuits. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. The containment purge supply and exhaust isolation valves are sealed closed by removing power from the valves. This is accomplished by pulling the control power fuses for each of the valves. The associated fuse blocks are then locked. This is consistent with the guidance contained in NUREG-0737 Item II.E.4.2 and Standard Review Plan 6.2.4, "Containment Isolation System," Item II.f.

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3/4.6.5 SECONDARY CONTAINMENT

3/4.6.5.1 ENCLOSURE BUILDING FILTRATION SYSTEM

The OPERABILITY of the Enclosure Building Filtration System ensures that containment leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the accident analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

The laboratory testing requirement for the charcoal sample to have a removal efficiency of ≥ 95% is more conservative than the elemental and organic iodine removal efficiencies of 90% and 70%, respectively, assumed in the DBA analyses for the EBFS charcoal adsorbers in the Millstone Unit 2 Final Safety Analysis Report. A removal efficiency acceptance criteria of ≥ 95% will ensure the charcoal has the capability to perform its intended safety function throughout the length of an operating cycle.

3/4.6.5.2 ENCLOSURE BUILDING

The OPERABILITY of the Enclosure Building ensures that the releases of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with operation of the Enclosure Building Filtration System, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

One Enclosure Building Filtration System train is required to establish a negative pressure of 0.25 inches W.G. in the Enclosure Building Filtration Region within one minute after an Enclosure Building Filtration Actuation Signal is generated. The one minute time requirement does not include the time necessary for the associated emergency diesel generator to start and power Enclosure Building Filtration System equipment.

To enable the Enclosure Building Filtration System to establish the required negative pressure in the Enclosure Building, it is necessary to ensure that all Enclosure Building access openings are closed. For double door access openings, both doors are required to be closed and latched, except for normal passage. For single door access openings, that door is required to be closed and latched, except for normal passage.

A door is OPERABLE when it is capable of automatically closing and latching. If the required door is not capable of automatically closing and latching, the door must be maintained closed and latched or personnel may be stationed at the door to ensure that the door is closed and latched after each transit through the door. Otherwise, the access opening (door) should be declared inoperable and the appropriate technical specification action statement entered.

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The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The required circuits between the offsite transmission network and the onsite Class 1E distribution system (Station Busses 24C, 24D, and 24E) that satisfy Technical Specification 3.8.1.1.a (MODES 1, 2, 3, and 4) consist of the following:

1. At least two of the four 345 kV transmission lines (310 Line, 348 Line, 371 Line, and 383 Line) that tie the switchyard to the grid are in service with their switchyard breakers closed.

AND

2. The following circuits from the switchyard to the onsite electrical distribution system:

   a. Station safeguards busses 24C and 24D via the Unit 2 Reserve Station Service Transformer and bus 24G; and

   b. Station bus 24E via the Unit 3 Reserve Station Service Transformer or Unit 3 Normal Station Service Transformer (energized with breaker 13T and associated disconnect switches open) and bus 34A or 34B.

The required circuit between the offsite transmission network and the onsite Class 1E distribution system (Station Busses 24C, 24D, and 24E) that satisfies Technical Specification 3.8.1.2.a (MODES 5 and 6) consists of the following:

1. At least one of the four 345 kV transmission lines (310 Line, 348 Line, 371 Line, and 383 Line) that tie the switchyard to the grid are in service with their switchyard breakers closed.

AND

2. The following circuit from the switchyard to the onsite electrical distribution system:

   a. Station safeguards bus 24C or 24D via the Unit 2 Reserve Station Service Transformer and bus 24G; or

   b. Station bus 24E via the Unit 3 Reserve Station Service Transformer or Unit 3 Normal Station Service Transformer (energized with breaker 13T and associated disconnect switches open) and bus 34A or 34B.

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3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for these isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. FSAR Table 6.2-65 lists all containment isolation valves. The addition of deletion of any containment isolation valve shall be made in accordance with Section 50.59 of 10CFR50 and approved by the committee(s) as described in the NUQAP Topical Report.

3/4.6.4 COMBUSTIBLE GAS CONTROL

Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. Containment hydrogen concentration is also important in verifying the adequacy of mitigating actions. The requirement to perform a hydrogen sensor calibration at least every 92 days is based upon vendor recommendations to maintain sensor calibration. This calibration consists of a two point calibration, utilizing gas containing approximately one percent hydrogen gas for one of the calibration points, and gas containing approximately four percent hydrogen gas for the other calibration point.

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the Mechanical Vacuum Pumps are capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Post-LOCA performance of the hydrogen recombiner blowers is based on a series of equations supplied by the blower manufacturer. These equations are also the basis of the acceptance criteria used in the surveillance procedure. The required performance was based on starting containment conditions before the LOCA of 10.59 psia (total pressure), 120°F and 100% relative humidiy.

The surveillance procedure shall use the following methods to verify acceptable blower flow rate:

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generator, the licensee will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in ACTION Statement 'b'.

According to Generic Letter 84-15, 24 hours is reasonable to confirm that the OPERABLE diesel generator is not affected by the same problem as the inoperable diesel generator.

Action Statement 'c'

Required ACTION Statement 'c' provides an allowance to avoid unnecessary testing of the other OPERABLE diesel generator. If it can be determined that the cause of the inoperable diesel generator does not exist on the operable diesel generator, Surveillance Requirement 4.8.1.1.2.a.5 does not have to be performed. If the cause of inoperability exists on the other OPERABLE diesel generator, the other OPERABLE diesel generator would be declared inoperable upon discovery and ACTION Statement 'f' would be entered and appropriate actions will be completed per ACTION Statement 'f'. Once the failure is repaired, the common cause failure no longer exists, and the required ACTION 'c' will be satisfied. If the cause of the initial inoperable diesel generator can not be confirmed not to exist on the remaining diesel generator, performance of Surveillance Requirement 4.8.1.1.2.a.5 (within 8 hours of entering ACTION Statement 'c') suffices to provide assurance of continued OPERABILITY of the other diesel generator.

In the event, the inoperable diesel generator is restored to operable status prior to determination of the cause of the inoperability of the diesel generator, the licensee will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under eight hours constraint imposed while in ACTION Statement 'c'.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

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