August 8, 1990

NOTE FOR: Steven A. Varga, Director
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SUBJECT: RESOLUTION OF THE GENERIC DPO ISSUES CONCERNING THE MCGUIRE TECHNICAL SPECIFICATIONS

Dr. Thomas Murley's memorandum dated December 29, 1989, identified the scope of work required to resolve the Differing Professional Opinion (DPO) issues concerning the McGuire Technical Specifications. The Reactor Systems Branch (RSB) was assigned the responsibility to resolve all the plant-specific issues. The memorandum dated May 14, 1990 from A. Thadani to S. Varga contains the RSB resolutions to the plant-specific issues. The Technical Specifications Branch (OTSB) was assigned the responsibility to resolve all the DPO generic issues in the May 14, 1990 memorandum. The enclosure to this note contains the OTSB resolutions to the generic issues. The resolutions of these 20 generic issues are directed toward the new Westinghouse Standard Technical Specifications.

ORIGINAL SIGNED BY JOSE A. CALVO

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AVAILABILITY OF RCPs DEPARTURE FROM NUCLEATE BOILING (DNB) TS

Concern 9A
Question 8e
3/4.2.5

Issue

The assertion involving Concern 9A consists of the following three parts:

I. The DPO asserts that the value for the reactor coolant system average temperature (Tavg) given in the TS Table 3.2-1 is not consistent with the value given for Tavg in FSAR Figure 5.3.3-1 for the rated power conditions. Furthermore, the DPO asserts that the following should be provided in the TS:

a.) The setpoint and allowable values of Tavg;
b.) The related power level ascribed to Tavg; and
c.) The reactor coolant system Tavg for the zero power condition.

II. The DPO asserts that the values for pressurizer pressure in TS Table 3.2-1 are not consistent with the information given in FSAR Table 15.1.2-2 and Table 4.1-1 of DPO reference 20. Also, the DPO asserts that the setpoint and allowable values of the pressurizer pressure should be provided in the TS.

III. The DPO asserts that the pressurizer pressure should be provided in TS 2.1-1 and 3/4.4.3.

Resolution - I.

The values of Tavg listed in Table 3.2-1 of the TS are Limiting Conditions of Operation (LCOs) and are derived from plant safety analyses. These limiting values are established in conjunction with limiting values for other principal thermal-hydraulic parameters to ensure sufficient DNB margin. These limits ensure that the DNB safety limit will not be violated in the event of a plant transient.

FSAR Figure 5.3.3-1 is a plot of the expected Tavg versus power level. The values of Tavg in the plot are not derived from the plant safety analyses. They are estimates of the actual values of Tavg that will exist when the plant is operated the way the licensee intends. All the plotted Tavg values are within the limits in TS Table 3.2-1.

a.) There is no instrumentation which monitors Tavg and generates a reactor trip signal based on the values in Table 3.2-1. Therefore, setpoints and allowable values corresponding to the limits in Table 3.2-1 do not need to be specified in the TS.

b.) and c.) The Tavg limits in TS Table 3.2-1 were derived by considering plant transients initiated from all power levels. Therefore, they are bounding values which are applicable at any power level. No related power level needs to be ascribed to the Tavg values in TS Table 3.2-1, and no separate Tavg limit needs to be specified in TS Table 3.2-1 for zero power operations.

Resolution - II.

The values of pressurizer pressure listed in Table 3.2-1 of the TS are LCOs and are derived from plant safety analyses. Pressurizer pressure is another principal thermal-hydraulic parameter in the calculation of DNB. These limits ensure that the DNB safety limit will not be violated in the event of a plant transient.

Since there are no automatic reactor trips actuated based on the values in TS 3.2-1, there is no need to specify setpoints or allowable values. The instrumentation that would initiate a reactor trip based on these parameters is addressed in TS 3.3.1.

The pressurizer pressure value in Table 4.1-1 of reference 20 of the DPO is the nominal design pressure for the reactor coolant system and reactor internals and is an expected value for plant operation. It is an estimate of the actual value of pressurizer pressure that will exist when the plant is operated the way the licensee intends. The nominal value is within the limits of TS Table 3.2-1.

FSAR Table 15.1.2-2 is part of the description of the plant safety analyses. These analyses include adjustments to account for steady state fluctuations and measurement error. The DPO suggests that the limits in TS Table 3.2-1 should equal the reference 20 nominal value minus the adjustment specified in the safety analyses. This suggestion is not correct. The limits in the TS are derived by making adjustments on safety analysis limiting values of the pressure - not nominal values.

Resolution - III.

In the new STS, pressurizer pressure is included in the curves in Section 2.1.1 (it is also included in the Section 2.1.1 curves of the current STS). Specification 3/4.4.3 specifies the operability of the pressurizer. The operability of the pressurizer is determined based on water volume and heater capacity; therefore, pressurizer pressure does not need to be included in TS 3/4.4.3.

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

Concern 15B
Questions 8a, 8b, 8c, 8d, and 8e
TS 3/4.4.1

Issue

The DPO asserts that four Reactor Coolant System (RCS) loops should be required to be OPERABLE in MODE 3 (Hot Standby) to meet the assumptions of the safety analysis for a number of accident scenarios. Each of Questions 8a - 8d below discusses this concern for a different type of accident.

Question 8a: OCCURRENCES WITH RAPID REACTIVITY INCREASE
Pertaining to "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Sub-Critical Condition."

This Technical Specification (TS) at the time of the DPO submittal required that two RCS loops be OPERABLE and one RCS loop be in operation in MODE 3. The FSAR for McGuire (and other Westinghouse plants) assumes four Reactor Coolant Pumps (RCPs) are running for this event. The DPO asserts that any Technical Specification allowing operability of less than four RCS loops in MODE 3 would not be in conformance with the FSAR and is non-conservative.

Question 8b: STEAM LINE BREAKS: OCCURRENCES
Pertaining to "Major Rupture of a Main Steamline."

The McGuire FSAR states that the resulting impact on shutdown margin for this event during MODES 3, 4, and 5 is improved over that of the design basis for zero power, just critical and Tavg = 557°F. The DPO asserts, however, that the design basis case may not be the most limiting case. It states that it is conceivable that two loop operation may be less conservative than either four RCPs continuing to operate or four RCPs tripped on Safety Injection. The conclusion of the DPO is that any Technical Specification allowing operability of less than four RCS loops in MODE 3 would not be in conformance with the FSAR and is non-conservative.

Question 8c: LOSS OF PRIMARY COOLANT: OCCURRENCES
Pertaining to "Small Break LOCA (SBLOCA)."

The McGuire FSAR and WCAP 8356 describe the SBLOCA as a design basis event when it occurs from the Rated Power (MODE 1) and Hot Standby (MODE 3) conditions. The assertion contained in the DPO is that "until further evaluations are made, we must conclude that the current Safety Analysis Limits of the SBLOCA event is four RCS pumps OPERABLE in MODE 3 down to 425 psig/350°F" and that the operability of less than four RCS loops in MODE 3 would not be in conformance with the current safety analysis limits and is not conservative. The DPO also contains a similar assertion for the large break LOCA scenario.

Question 8d: OCCURRENCES CAUSING AN INITIAL INCREASE OF RCS TEMPERATURE

The assertion contained in the DPO states that the increase of RCS temperature events are of concern because of the potential influence of the positive moderator temperature coefficient. It discusses several events and states that all but one are licensing basis events from rated power. The conclusion of the DPO is that these events are important in MODE 3 due to the positive moderator temperature coefficient and states that operability of less than four RCS loops in MODE 3 would not be in conformance with the safety analyses limits and is not conservative.

Question 8e: AVAILABILITY OF RCPs

The DPO states that four RCS loops would be required in MODE 3 to meet the requirements of the licensing basis events from zero power. In addition it suggests that, in MODE 4, a minimum set of RCS pumps and loops be used to cool and depressurize the plant down to effectively zero pressure in the steam generators before transferring the heat sink to the RHR system. This is to ensure control of Steam Line Break and LOCA events down to RCS conditions where RCS flows are not necessary. The part of this question addressing MODE 4 is addressed in Concern 18A.

Resolution

In the new STS the LCO for RCS loops in MODE 3 states:

[Two] RCS loops shall be OPERABLE, and

a. [Two] RCS loops shall be in operation when the reactor trip breakers are closed, or

b. One RCS loop shall be in operation when the reactor trip breakers are open.

The numbers in brackets indicate that each plant must supply the number of pumps which is required to meet their safety analysis. For four loop Westinghouse plants, analysis indicates that two is the appropriate number of RCS loops.

At the time of the DPO submittal, this TS required that two RCS loops be OPERABLE and only one RCS loop be in operation in MODE 3. The FSAR for McGuire (and other Westinghouse plants) assumed from two to four RCPs operating for many of the accidents discussed above. Westinghouse acknowledged the discrepancy in a letter dated July 9, 1984, from E. P. Rahe to D. Eisenhut. At that time, Westinghouse reviewed the safety analyses for the accidents which are the most limiting at zero power for the reduced flow conditions of one RCP. These accidents are the steamline break, rod ejection, and control rod bank withdrawal from subcritical conditions. For the rod ejection and steamline break events, Westinghouse determined that the inconsistency between the safety analysis and the Technical Specification would not impact the conclusions presented in the FSAR. The analyses showed that the applicable accident criteria were met with only one RCS pump operating.

For the bank withdrawal from subcritical event, Westinghouse performed calculations which showed that the DNBR design basis may not be met when only one RCP is in operation. Consequently, the Westinghouse STS were changed to require at least two RCS loops in operation with the reactor trip breakers closed to meet the safety analysis limits for an inadvertent bank withdrawal from subcritical.

For the SBLOCA an analysis was conducted by Westinghouse assuming that all pumps were initially operating followed by either all the pumps tripping or all the pumps continuing to operate. The general conclusion was that there was a smaller peak clad temperature for the case of all the pumps operating when compared to the case of all the pumps tripped. This case forms the bounding analysis since the reactor coolant pumps are not automatically tripped during the SBLOCA and continue to operate after the SBLOCA. For ECCS analysis for Westinghouse four LOOP plants the most conservative results are obtained when the RCPs are assumed to be tripped at the initiation of a postulated LOCA. The DPO's assertion is unsubstantiated since the ECCS analysis demonstrated that acceptable fuel cladding temperatures resulted for the more conservative scenario which resulted when the RCPs are assumed to be shut down.

Therefore, for the limiting design basis events at zero power, the proposed new STS will ensure the safety analysis limits are met. The other events described in the DPO are not limiting design basis events at zero power and are thereby bounded by the limiting events.

AVAILABILITY OF REACTOR COOLANT SYSTEM LOOPS IN MODE 4
(HOT SHUTDOWN)

Concern 18A
Question 10
TS Page 3/4 4-3

Issue

The DPO proposed two additional requirements for this specification. The first is that two RCS loops be Operable whenever RHR loops are in operation, in order to provide for the failure of a single motorized valve in the RHR/RCS suction line. The second is that surveillance requirements be added to require a determination of the operability of the associated Auxiliary Feedwater System and the Atmospheric Dump Valves.

Specifically, the DPO concerns the McGuire FSAR, which describes a scenario comprised of the failure of a single motorized valve in the RHR/RCS suction line concurrent with the loss of offsite power. For this scenario the DPO asserts that two RCS loops should be operable whenever a plant has RHR operating in Mode 4. Furthermore, the DPO asserts that the current specifications are not conservative because they lack operability requirements for the Auxiliary Feedwater Systems or Atmospheric Valves in Mode 4.

Resolution

The new Westinghouse STS require two loops consisting of any combination of RCS loop and RHR loops be Operable and at least one loop be in operation in Mode 4. The Bases for this LCO states, "Any one loop in operation provides enough flow capacity to remove the decay heat from the core with forced circulation. The second loop, which is required to be OPERABLE, meets single failure criteria." Therefore, in order for a licensee to take credit for each loop, there cannot be a single failure which could render both loops inoperable. The McGuire design which is typical of Westinghouse plants includes a single RHR suction line which connects the reactor coolant loop to the RHR pumps. This RHR suction line contains two motorized valves in series. The DPO asserts that a single failure concurrent with the loss of offsite power could cause one of these valves to fail close during Mode 4; thereby, eliminating the core cooling capability of the RHR system. These valves are opened and left open when core cooling via the RHR is initiated in Mode 4. Since motorized valves fail in the "as is" position, these suction line valves remain open after a single active failure concurrent with a loss of offsite power resulting in the RHR system maintaining its full functional capability in Mode 4. Therefore, to require in the TS that 2 RCS loops be operable whenever the RHR loop(s) are in operation is not necessary.

As discussed in the resolution of concern 29A, the new Westinghouse STS require the operability of one motor driven AFS pump in Mode 4 when a steam generator is relied on for heat removal. The new STS do not require operability of the Atmospheric Dump Valves (ADVs) in Mode 4. The preferred method of removing heat from the steam generators in Mode 4 is through the turbine bypass valves to the condenser. If this path becomes unavailable, the heat load is low enough in Mode 4 that SG secondary side steaming would take time to reach a high enough pressure to necessitate venting. Several options (including the opening of ADVs) would be available to the operators during that time to achieve venting or eliminate the need to vent. Ultimately, the safety valves would vent the pressure. The safety valve LCO does not require the safety valves to be operable in Mode 4: however, the TS definition of Operability and the ASME code require operability of the safety valves when the steam generator is operable.

The DPC also discussed concerns about the depth of the Surveillance Requirements (SR) and suggested that additional SRs be added on the systems in this LCO. The existing SRs are not intended to be complete tests of the system performance; they are quick, simple, frequent checks (every 12 hours) to ensure that the equipment is operating properly. The more detailed testing is done in chapter 5 of the STS and the inservice test program. Therefore, there is no need to supplement the existing SRs.

AVAILABILITY OF REACTOR COOLANT SYSTEM LOOPS IN MODE 5
(COLD SHUTDOWN)

Concern 19A
Question 8e
TS 3.4.1

Issue

The DPO made the following assertions for the Cold Shutdown Mode of operation:

(1) if the steam generators are used for cooling the Auxiliary Feedwater System and Atmospheric Dump Valves should be required to be operable.

(2) There is no basis for allowing the operating RHR pumps to be de-energized for 1 hour.

(3) The surveillance requirements do not fully test all aspects of operability of the RHR and RCS loops.

Resolution

The new Westinghouse STS require one RC loop or one RHR loop to be operating and either one additional RHR loop to be OPERABLE or the secondary-side water level of at least two steam generators to be [17]% or greater of the Low-Low Trip Setpoint. A note in the Limiting Condition for Operation (LCO) allows the RHR pump or the RHR loop in operation to be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause reduction of the RCS boron concentration; and (2) core outlet temperature is maintained at least 10°F below saturation temperature. The Surveillance Requirements verify that at least one RHR or RCS loop is operating and that there is adequate water level in the SG.

In MODE 5 with the RCS loops filled, the objectives of this LCO are: (1) to remove decay heat generated in the fuel; and (2) to prevent stratification of the soluble boric acid. In MODE 5, an operating RHR or RCS loop accomplishes these functions. The other operable RHR loop or the two steam generators with adequate secondary side water level provide single failure protection. Under these conditions of low heat load, the heat sinks in the two steam generators provide adequate back-up cooling until a RCS or RHR loop can be put into operation. Also, under these low heat load conditions, operability of neither the Auxiliary Feedwater System nor the Atmospheric Dump Valves is necessary.

In the new STS, the note allowing the operating RHR or RCS pump to be de-energized for up to 1 hour is limited; it may only be exercised once in an 8 hour period. This time period is needed to perform surveillance testing. As explained above, compensatory measures including close monitoring of coolant temperatures are required to exercise the 1 hour allowance. The RHR or RCS loop would still be available to be restarted if coolant temperatures exceeded the surveillance limit in the note. Experience in the use of this note has shown that plants do not experience heating or boric acid stratification problems.

The surveillances in this LCO do not include testing of alarms and design basis flow rates. The purpose of these surveillances is to provide quick, simple, frequent checks (every 12 hours) to ensure that the equipment is operating properly. The more detailed testing is done in chapter 5 of the STS and the inservice testing program. Therefore, there is no need to augment the existing surveillance requirements.

STANDBY NUCLEAR SERVICE WATER POND
(ULTIMATE HEAT SINK)

Concern 20B
TS Section 3/4.7.4

Issue

The DPO asserts that the applicability section of the Standby Nuclear Service Water Pond (SNSWP) TS which includes Modes 1, 2, 3, and 4 should also include Modes 5 and 6.

Resolution

The need for operability of the Ultimate Heat Sink (UHS) in Modes 5 and 6 is addressed in the new STS through the definition of Operability. UHS is required as a support system for other systems such as RHR which are required by STS to be operable in Modes 5 and 6. In Modes 5 and 6 the heat load is low; therefore, the demands on the UHS as a support system would be well below the temperature and volume requirements of the UHS LCO.

REFUELING OPERATIONS - LOW WATER LEVEL

Concern 21B
TS Page 3/4.9-11

Issue

The DPO asserts that both RHR loops should be in operation in Mode 6 with less than 23 feet of water above the top of the reactor vessel flange. In support of this statement the DPO postulates the loss of the operable RHR loop without operator action; the DPO asserts that this scenario would result in boiling in 5 minutes and core uncovery in 100 minutes.

Resolution

The new STS require that one RHR loop be operating and the other RHR loop be operable under the low water level conditions. The new STS also require action to restore RHR cooling if it is lost. The DPO seems to express concern over a scenario where the operating RHR loop fails and the reactor coolant heats up and uncovers the core before the operators become aware of the inoperable RHR loop and take action to operate the other RHR loop. The operating RHR loop has an alarm for low RHR flow and other instruments provide multiple, diverse indications of loss of RHR cooling to the operators. In addition, several operations personnel would be present in the area of the reactor cavity. For these reasons it is highly unlikely that a loss of RHR flow would go unnoticed and uncorrected long enough to allow the core to become uncovered. The other RHR loop is required by STS to be operable. Through the definition of Operability the support systems necessary for operation of the other RHR loop must also be operable. Finally, both offsite and emergency diesel generator power are required to be operable in Mode 6 by STS 3.8.1.2. Therefore, an additional STS requirement to have both RHR loops operating is not necessary.

AUXILIARY FEEDWATER SYSTEM

Concern 29A
TS Page 3/4.7-4

Issue

The DPO states that the TS should require operability of the steam driven auxiliary feedwater pump in Mode 4. The DPO also questions the derivation of the Steam Line - Pressure Low signal.

Resolution

The new STS require operability of the one motor driven Auxiliary Feedwater System (AFP) pump in Mode 4 whenever a steam generator is relied on for heat removal. Once the plant is switched to RHR cooling, operability of the Auxiliary Feedwater System (AFS) is no longer required. This is a change over the current STS which do not require AFS operability in Mode 4. The current STS assume that the plant switches from SG cooling to RHR cooling when a change from Mode 3 to Mode 4 occurs. During the review of the new STS, it was found that some plants maintain cooling via the steam generators into the upper temperature range of Mode 4. These plants maintain operability of the Auxiliary Feedwater System via administrative controls until cooling is switched to RHR.

The Steam Line Pressure Low Signal used in the main steam line break accident analysis is derived from steam line sensors downstream of the steam generator flow restriction orifices. This results in a conservative measure of steam generator pressure since the steam flow restrictors do not cause a significant pressure drop except during a doubled ended steam line break. The blowdown phase of the double ended steam line break lasts only a few seconds. The accurate pressure sensing in the steam lines (the generation of the steam line pressure low signal) requires less than 2 seconds and steam line isolation requires less than 7 seconds. Deriving this low pressure signal from sensors downstream of the steam generator flow restriction orifices is conservative.

MAIN STEAM ISOLATION VALVES

Concern 30A
TS Page 3/4.7-8

Issue

The DPO contains an assertion that there is a conflict between TS Sections 3.7.1.4, 3.6.3 and TS Table 3.3-4 dealing with the applicability modes for operability of the Main Steam Isolation Valves.

Resolution

The Main Steam Isolation Valves (MSIV) have two accident mitigation functions. First, during a steam line break the MSIVs close to prevent blowdown from more than one steam generator. This function is necessary in Modes 1, 2, and 3. In Mode 4, the lower reactor coolant temperature reduces the consequences of the steamline blowdown such that MSIV closure is not necessary. In the new STS, the LCOs which address this function in plant systems and instrumentation chapters requirs MSIV operability in Modes 1, 2, and 3.

The second accident mitigation function for the MSIVs is containment isolation. This function is necessary in Modes 1, 2, 3, and 4. In the new STS, the LCOs which address this function in the containment and instrumentation chapters require MSIV operability in Modes 1, 2, 3, and 4.

STEAM GENERATOR POWER OPERATED RELIEF VALVES (SGPORV)

Concern 31A
TS Page 3/4.7.8a

Issue

The DPO states that the TS should include the SGPORVs since under the loss of offsite power condition these valves are necessary for cooling down the plant by natural circulation. Furthermore, the DPO states that additional relieving capacity should be covered by TS since the reactor will operate at power levels as high as 20% during the loss of offsite power condition.

Resolution

The loss of offsite power will cause the Reactor Coolant Pumps (RCP) to trip since the only power source for these pumps is the offsite grid. At reactor power levels greater than or equal to 10% the tripping of the RCP will initiate a reactor scram. At reactor power levels less than 10% the reactor would be manually scrammed by the operator. The power level for either scram is equivalent to the initial decay heat power level after a scram. The required heat removal capacity is within the design limits of natural circulation.

The bases for the new STS state that the Atmospheric Dump Valves (ADVs) will be used to cool down the plant for accidents which are accompanied by a loss of offsite power. Therefore, the ADVs are part of the primary success path for such accidents and are required by the new STS in Modes 1, 2 and 3. PORVs are used to minimize the opening of the Main Steam Safety Valves (MSSVs); the MSSVs are part of the primary success path for events such as full power turbine trip without steam dump. Since the SGPORVs are not part of the primary success path, they do not meet the criteria for inclusion in TS pursuant to the Commission's Policy Statement. Therefore, operability of the SGPORVs is not required by the new STS.

SERVICE WATER SYSTEM

Concern 33A
TS Section 3/4.7.4

Issue
The DPO states that the applicability section of the Service Water System (SWS) TS which includes Modes 1, 2, 3, and 4 should also include Modes 5 and 6.

Resolution

The need for operability of the Service Water System (SWS) in Modes 5 and 6 is addressed in the new STS through the definition of Operability. SWS is required as a support system for other systems such as RHR which are required by STS to be operable in Modes 5 and 6. Since the trains of the SWS are typically cross connected as in the McGuire Plant, one train of SWS is adequate to meet the support function for both RHR trains in Modes 5 and 6. Both trains of the SWS are not required to be operable to provide single failure protection in Modes 5 and 6 since the heat load is low, and there are other methods which can be instituted by the operators to handle the low heat load if the SWS fails. Also, this allows licensees to perform necessary maintenance and system modifications.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION - HIGH WATER LEVEL

Concern 35A
TS 3/4.9.8

Issue

The DPO states that the action statement should require containment isolation within 2.5 hours when no RHR loops are operable. Also, the DPO states that the TS should require operability of the containment sump and alternate cooling methods in this Mode.

Resolution

In current ST', the action statement allows 4 hours to isolate containment when no RHR loops are operable. In the new STS, the instruction to isolate containment has been removed from the action statement. The new STS require the establishment of alternate cooling methods whenever the RHR is unavailable. The action statement concentrates on the most important task of supplying core cooling and leaves the provision of containment isolation to the licensee's contingency procedures.

The alternate cooling methods do not need to be required to be operable in Mode 6 with the cavity flooded. The Commission's Interim Policy Statement on Technical Specification Improvement states criteria for deciding which equipment and conditions should be included in TS. Under those criteria the primary success path system, RHR, is required by TS. The provision of alternate cooling methods referred to in the first paragraph is left as the responsibility of the licensee's contingency procedures. In Mode 6 with the cavity flooded, there is a large volume of water over the core and a low decay heat load. Under these conditions the operator has (1) alternative cooling methods which can handle the low decay heat load and (2) time to implement those alternatives.

REFUELING OPERATIONS - LOW WATER LEVEL

Concern 36A
TS Page 3/4.9-11

Issue

The DPO states that the action statement should require containment isolation immediately when no RHR loops are operable.

Resolution

In the current STS, the action statement allows 4 hours to isolate containment when no RHR loops are operable. In the new STS, the instruction to isolate containment has been removed from the action statement. As discussed in the resolution to concern 35A, the action statement concentrates on the most important task of supplying core cooling and leaves the provision of the containment isolation to the licensee's contingency procedures.

REACTOR TRIP INSTRUMENTATION SETPOINT

Concern 38A
Table 2.2.-1

The assertion in the DPO states that the Technical Specification nomenclature "Low Power Reactor Trips Block, P-7" is incorrect and should be labeled "High Power Reactor Trips Block".

Resolution

The nomenclature is an acceptable description for this function without change; however, information describing the P-7 permissive and the P-10 and P-13 trips is discussed in detail in the new STS Bases under the title "Low Power Reactor Trips Block, P-7."

REACTOR TRIP INSTRUMENTATION SETPOINTS

Concern 3B
Table 2.2-1

Issue

The assertion in the DPO is that the absence of the permissive P-7 [on P-10 and P-13] introduces new events to evaluate for safety. The DPO further asserts that the impact of blocking the Pressurizer Water Level-High trip below P-7 should be evaluated.

Resolution

The new STS include in Table 3.3.1-1 the P-7 [on P-10 and P-13] interlock.

Several reactor trips (including Pressurizer Water Level-High) are only required when operating above 10% power, the P-7 setpoint. The P-7 interlock enables and disables trips as reactor power passes through the 10% power setpoint. Below 10% power, the RCS is capable of sufficient natural circulation without any RCP running to prevent DNB.

The Pressurizer Water Level-High trip is a back-up signal for Pressurizer Pressure High trip and provides protection against passing water through the pressurizer safety valves. A reactor trip is actuated before the pressurizer is water solid. These level channels provide input to the pressurizer level control system and do not actuate the safety valves.

This trip must be operable in Mode 1 when there is a potential for overfilling the pressurizer. This trip is automatically enabled on increasing power by the P-7 interlock. On decreasing power the absence of P-7, automatically blocks this trip. Below the P-7 setpoint, transients which could raise pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

ENGINEERED SAFETY FEATURES ACTUATION
SYSTEM (ESFAS) INSTRUMENTATION

Question 5B
Concern 12B
Table 3.3-3

Issue

The assertion in the DPO recommended that the staff consider the consequences of not requiring automatic switchover to recirculation on RWST level for Mode 4 in addition to Modes 1, 2, and 3.

Resolution

The new STS and the current STS require the operability of the switchover to containment sump on RWST level low for Modes 1, 2, 3, and 4.

SOURCE RANGE NEUTRON FLUX

Concern 10A*
TS Page 3/4.3 - Item 6c

Issue

The assertions in the DPO are as follows:

1) Power range neutron flux trip (low and high) setpoints and intermediate range neutron flux are used for events being initiated in a "Subcritical" condition as described in FSAR (table 7.2.1-4); however, the TS does not require their operability in Modes 3, 4, and 5.

2) Furthermore, the source range trip is required to be operable in Modes 3, 4, and 5, yet there is no technical specification for it.

Resolution

1) The Power Range, Neutron Flux-High Setpoint and Low Setpoint do not have to be operable in Modes 3, 4, and 5, because the reactor is shutdown and the Nuclear Instrumentation System (NIS) power range detectors cannot detect neutron levels in the shutdown range. Other RTS functions and administrative controls provide protection against reactivity additions when in Modes 3, 4, and 5.

   The Intermediate Range Neutron Flux trip does not have to be operable in Modes 3, 4, or 5 because the controls rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required Shutdown Margin to mitigate the consequences of a positive reactivity addition accident and this margin is required to be monitored frequently. In Mode 6, all rods are fully inserted and the core has an increased Shutdown Margin. Also, the NIS intermediate range detectors cannot detect neutron levels in this range.

2) The new STS require the source range neutron flux trip function to be operable in Modes 2, 3, 4, and 5 with the reactor trip breakers closed and the rod control system capable of rod withdrawal. It is also required to be operable in Mode 3, 4, and 5 with trip breakers open when the only function of the source range monitor is indication.

P-11 INTERLOCK

Concern 10B
TS Page 3/4.3-2

Issue

The assertion in the DPO is that the licensee needed to evaluate the consequences of an event involving a Main Steam Line Break below the P-11 interlock reactor trip such that the trip will not be initiated by the Negative Steam Line Pressure Rate - High signal. This concern acknowledges the source range and intermediate range nuclear flux trips under these (small and intermediate size breaks) circumstances, on any return to power, as not being necessary because they are not required in the safety analysis. Their current proposed status precludes crediting their function capability and would leave only the power range low setpoint trip to trip the reactor. Furthermore, the resulting power levels of 35% as a safety analysis limit would be unacceptable without a substantial analysis of the event.

Resolution

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of safety injection (SI) or main steam line isolation. With 2/3 pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low Steam Line Isolation signal. When the Steam Line Pressure - Low Steam Line Isolation signal is manually blocked, the main steam isolation signal on Steam Line Pressure - Negative Rate - High is enabled. This provides protection for a steam line break by closure of the main steam isolation valves and initiation of a reactor trip.

ESFAS INSTRUMENTATION

Concern 14A
Table 3.3-3

Issue

The DPO asserts that the operability of the containment Phase B isolation on a Containment Pressure High-High signal should be required in Mode 4. The DPO also asserts that a containment Phase B isolation is necessary to establish containment integrity.

Resolution

The Containment Pressure High-High signal is initiated due to a large break LOCA or steam line break and it actuates containment spray and Phase B containment isolation. Containment Pressure High-High must be operable in Modes 1, 2 and 3 when there is sufficient energy in the primary and secondary sides to challenge the containment pressure High-High setpoint. In Mode 4, there is insufficient energy in the primary and secondary sides to challenge the Containment Pressure High-High set point. Therefore, operability of the Containment Pressure High-High signal is not necessary.

Containment Pressure High actuates SI and SI actuates containment Phase A isolation. Containment Phase A isolation isolates all lines into containment except those associated with the Engineered Safety Features. The CCW System, which is typically an Engineered Safety Features System as in the McGuire Plant, is not isolated by the Phase A isolation. Containment Phase A isolation establishes containment integrity and allows the continued use of the Reactor Coolant Pumps (RCPs) which rely on the CCW. The containment Phase B isolation is actuated by Containment Pressure High-High and isolates the CCW. The high pressure which causes the Containment Pressure High-High signal indicates accident conditions for which RCP operation is not necessary.

ESFAS INSTRUMENTATION TRIP SETPOINTS

Concern 15A
Table 3.3-4

Issue

The DPO asserts that a new Functional Unit which is part of ESFAS should be included in the TS. This new Functional Unit is "Closure of the Feedwater Isolation, Main Feedwater, and Bypass Modulation Valves."

Resolution

The new STS and the current STS include these valve closure functions under other functions in the ESFAS tables. The DPO acknowledges this fact, but asserts that the function needs to be included as a separate function in the ESFAS tables. The DPO gives no justification for including this separate function; therefore, no additional functional unit needs to be included in the STS.