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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2003-0033 10 CFR 50.36

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATION BASES REVISIONS

Nuclear Management Company, LLC (NMC), licensee for the Point Beach Nuclear Plant (PBNP) Units 1 and 2, hereby submits a revision to the Technical Specifications (TS) Bases for the following TSs: LCO 3.4.10, "Pressurizer Safety Valves," LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," LCO 3.5.2, "ECCS-Operating," LCO 3.6.4, "Containment Pressure," LCO 3.6.6, "Containment Spray and Cooling Systems," LCO 3.7.1, "Main Steam Safety Valves (MSSVs)," LCO 3.7.2, "Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves," and LCO 3.7.5, "Auxiliary Feedwater System." A description of the changes is provided in Attachment 1.

These changes have been screened for evaluation pursuant to the requirements of 10 CFR 50.59 in accordance with approved PBNP procedures and were determined to be acceptable.

Attachment 2 provides clean copies of the affected TS Bases pages indicating the changes.

If there are questions on this matter, please contact Roger Scott, of my staff, at (920) 755-7255.

Sincerely,

A. J. Cayia
Site Vice President

RDS/kmd

Attachments: 1 - Description of Changes
2 - Revised Technical Specification Bases Pages

cc: NRC Regional Administrator
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PSCW

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ATTACHMENT 1

To

NRC 2003-0033

Technical Specification Bases Revisions

Description of Changes

1.0 INTRODUCTION

Nuclear Management Company, LLC (NMC), licensee for the Point Beach Nuclear Plant (PBNP) Units 1 and 2, hereby submits a revision to the following Bases for Technical Specifications (TS):

- LCO 3.4.10, "Pressurizer Safety Valves"
- LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)"
- LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System"
- LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage"
- LCO 3.5.2, "ECCS-Operating "
- LCO 3.6.4, "Containment Pressure"
- LCO 3.6.6, "Containment Spray and Cooling Systems"
- LCO 3.7.1, "Main Steam Safety Valves (MSSVs)"
- LCO 3.7.2, "Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves"
- LCO 3.7.5, "Auxiliary Feedwater System"

2.0 DESCRIPTION OF CHANGES

B 3.4.10, "Pressurizer Safety Valves"

The Bases for LCO 3.4.10 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

B 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)"

The Bases for LCO 3.4.11 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

B 3.4.12, "Low Temperature Overpressure Protection (LTOP) System"

The Bases for LCO 3.4.12 was revised to replace various inconsistent statements regarding the status of accumulator discharge MOVs and supply breakers (when the accumulators are required to be isolated), with a single statement in the surveillance section. The statement clarifies that an accumulator is considered isolated when the discharge isolation valve is closed and power has been removed from the valve operator under administrative controls.

Additionally, a clarification was made such that the reactor head is considered unbolted when the bolting hardware does not impede the head from lifting to relieve pressure. Furthermore, the use of "reactor vessel head removed" as an example of "an opening in the RCS boundary to atmosphere with an equivalent system pressure relieving capability as a PORV" has been deleted.

Also, although the Bases for LCO 3.4.12 references ASME Section XI, it does not discuss this code, therefore this reference has been deleted.

B 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage"

The Bases for LCO 3.4.14 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

B 3.5.2, "ECCS-Operating "

The Bases for LCO 3.5.2 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

B 3.6.4, "Containment Pressure"

The Bases for LCO 3.6.4 was revised consistent with the measurement uncertainty recapture power uprate to clarify the power level and uncertainty used in the specific analyses, evaluations, or design basis.

B 3.6.6, "Containment Spray and Cooling Systems"

The Bases for LCO 3.6.6 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

Additionally, the Bases for LCO 3.6.6 was revised consistent with the measurement uncertainty recapture power uprate to clarify the power level and uncertainty used in the specific analyses, evaluations, or design basis.

B 3.7.1, "Main Steam Safety Valves (MSSVs)"

The Bases for LCO 3.7.2 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

Additionally, the Bases for LCO 3.7.1 was revised consistent with the measurement uncertainty recapture power uprate to clarify the power level and uncertainty used in the specific analyses, evaluations, or design basis.

B 3.7.2, "Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves"

The Bases for LCO 3.7.2 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

B 3.7.5, "Auxiliary Feedwater System"

The Bases for LCO 3.7.5 was revised to reflect the adoption of the ASME OM (Operation and Maintenance) Code, replacing ASME Section XI during the 4th IST interval.

ATTACHMENT 2

To

NRC 2003-0033

Technical Specification Bases Revisions

Affected TS Bases Pages:

B 3.4.10-4
B 3.4.11-6 and B 3.4.11-7
B 3.4.12-1 through B 3.4.12-7 and B 3.4.12-9 through B 3.4.12-11
B 3.4.14-4 and B 3.4.14-5
B 3.5.2-6 and B 3.5.2-7
B 3.6.4-1
B 3.6.6-4, B 3.6.6-8 and B 3.6.6-10
B 3.7.1-3, B 3.7.1-5, and 3.7.1-6
B 3.7.2-5 and B 3.7.2-6
B 3.7.5-8 and 3.7.5-10

BASES

ACTIONS (continued)

below the LTOP enabling temperature specified in the PTLR, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to +2.67%/-1.78% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Chapter 14.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS (continued)

removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours with T_{avg} reduced to < 500°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

BASES

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. WCAP-14602, Section 4.2.
3. ASME Code, Code for Operation and Maintenance of Nuclear Power Plants.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one Safety Injection (SI) pump incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event. Two PORVs are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capacity.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup

BASES

BACKGROUND (continued)

control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one SI pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

PORV Requirements

The Low Temperature Overpressure Protection System consists of two control trains. The trains incorporate two key-operated enabling switches and two valve control switches in the control room. Signals from pressurizer pressure instrumentation and reactor coolant Loop A hot leg pressure instrumentation are used to control the PORVS. The pressurizer pressure instrumentation controls one PORV, while the reactor coolant pressure instrumentation controls the other PORV.

The protection circuits are enabled by turning the key switches to the enabled position. When the circuit is enabled and the PORV block valves are fully open, a red light above the respective key switch illuminates, signifying the circuits are armed. With both circuits properly armed, each PORV with its valve control switch in the Auto position will open, if system pressure increases to the lift setpoint.

The PTLR presents the PORV setpoints for LTOP. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits.

BASES

BACKGROUND (continued)

The RCS is defined as vented if there is an opening in the reactor coolant system pressure boundary to atmosphere or the pressurizer relief tank that has an equivalent system pressure relieving capability as a PORV. Some examples of such openings include an open or removed PORV, open steam generator or pressurizer manways, a removed pressurizer safety valve, and the top of the reactor vessel when the reactor vessel head has been unbolted (i.e., bolting hardware does not impede head from lifting to relieve pressure, Reference 8). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

The required vent capacity may be provided by one or more vent paths.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, 3 and in MODE 4 with RCS cold leg temperature exceeding the LTOP enabling temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At the LTOP arming temperature specified in the PTLR and below, overpressure prevention falls to two OPERABLE PORVs or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the PORV method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but one SI pump incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops — MODE 4," and LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one SI pump is actuated. Thus, the LCO allows only one SI pump OPERABLE during the LTOP MODES. Since neither one PORV nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (approximately 265 °F and below) than that of the LCO (270 °F and below).

Fracture mechanics analyses established the temperature of LTOP Applicability at 270°F.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of one SI pump OPERABLE and SI actuation enabled.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one SI pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent path with venting capability equivalent to or greater than a PORV is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one SI pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of the NRC Policy Statement.

BASES

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires a maximum of one SI pump capable of injecting into the RCS and each accumulator isolated from the RCS when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. (See SR 3.4.12.1 and SR 3.4.12.2 discussion regarding equipment isolation and status control.)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. One of the following pressure relief capabilities:

   1. Two OPERABLE PORVs; or

      A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

   2. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with a venting capability equivalent to or greater than a PORV.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ the LTOP enabling temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the LTOP enabling temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3 and MODE 4 above the LTOP enabling

BASES

APPLICABILITY (continued)

temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

The ACTIONS are modified by a Note stating that while the LCO is not met, entry into MODE 6, with the reactor vessel head on, from MODE 6, with the reactor vessel head removed, is not permitted. This Note prevents entry into the MODES of applicability for LTOP without the requirements of LCO 3.4.12 being met. This Note is necessary, because LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.

A.1

With two SI pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1, C.1 and C.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > LTOP enabling temperature specified in the PTLR, an accumulator pressure of 800 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

BASES

ACTIONS (continued)

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of one SI pump is verified capable of injecting into the RCS and the accumulators are verified to be isolated from the RCS when accumulator pressure is ≥ the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves provided in the PTLR.

The SI pump is rendered incapable of injecting into the RCS through removing the power from the pump by racking the breaker out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull out and at least one valve in the discharge flow path being closed.

The accumulators are isolated from the RCS by closing the discharge isolation valves and removing power from the valve operators under administrative controls. SR 3.4.12.2 is modified by a Note specifying that this verification is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. If accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS beyond the LTOP limits.

The Frequency of 12 hours is sufficient, considering the administrative controls and indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.3

The RCS vent path with a venting capability equivalent or greater than a PORV is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context).

BASES

SURVEILLANCE REQUIREMENTS (continued)

b. Once every 31 days for other vent path(s) (e.g., a vent or a valve that is locked, sealed, or secured in position). A removed pressurizer safety valve or open manway also fits this category.

The passive vent path arrangement must only be open when required to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.c.2.

SR 3.4.12.4

The required trains of LTOP must be verified enabled every 72 hours to provide the flow path for each required PORV to perform its function when actuated. A LTOP train is verified enabled by ensuring its enabling switch is in the correct position and that the associated PORV Block Valve is open.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.5

Performance of a COT is required every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.6

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

SR 3.4.12.7 and SR 3.4.12.8

Operating the PORVs, the solenoid air control valves and the check valves on the nitrogen gas bottles ensures the PORVs and PORV control system will actuate properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the frequency of other surveillances used to demonstrate PORV OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR, Chapter 14
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. Engineering Evaluation 2001-0037, Rev 0, 12/13/01, Evaluation of Unbolted Head as an RCS vent path.

BASES

ACTIONS (continued) B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the limit contained in the PIV Leakage Program and to identify each leaking valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. Event V Order, April 20, 1981.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. Technical Requirements Manual.
7. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.
8. 10 CFR 50.55a(g).

BASES

ACTIONS (continued)

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

With more than one component inoperable such that both ECCS trains are not available, the facility is in a condition outside design and licensing basis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance

BASES

SURVEILLANCE REQUIREMENTS (continued)

of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which implements the requirements of the ASME OM Code, providing the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.3 and SR 3.5.2.4

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.5

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and on the need to have access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. FSAR, Section 6.1.1.
2. 10 CFR 50.46.
3. FSAR, Section 6.2.1.
4. FSAR, Chapter 14, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside the upper containment pressure limit coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB. The LOCA and SLB containment integrity evaluations are accomplished by use of the digital computer code, COCO.

The initial pressure condition used in the containment LOCA analysis was 14.7 psia (0.0 psig). This resulted in a maximum peak pressure from a LOCA of between 52 and 53 psig. The initial pressure condition used in the SLB containment analysis was 16.7 psia (2.0 psig). This resulted in a maximum peak pressure from the limiting SLB inside containment of 59.8 psig. The limiting SLB case assumed the failure of a feedwater regulating valve at 100% rated thermal power plus measurement uncertainty. The SLB containment analysis shows that the maximum peak calculated containment pressure results from this limiting SLB case. The limiting SLB case does not exceed the containment design pressure, of 60 psig.

The containment was also designed for an external pressure load equivalent to -2.0 psig. This limit is sufficient to accommodate increases in atmospheric pressure and decreases in containment temperature after the establishment of containment integrity without the use of the containment purge valves.

BASES

APPLICABLE SAFETY ANALYSES (continued)

or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and two containment accident fan cooler units being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the peak containment pressure and temperature are approximately 52-53 psig and 291°F respectively (experienced during a LOCA.) Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion). The analyses and evaluations assume a unit specific power level of 100% RTP plus power measurement uncertainty, one containment spray train and two containment accident fan cooler units with their accident fans in operation, and initial (pre-accident) containment conditions of 120°F and 0.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment Hi-Hi pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of 63 seconds includes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 3).

Containment accident fan cooler unit performance for post accident conditions is given in Reference 3. The results of the analysis show that one train of containment spray and two containment accident fan cooler units will provide 100% of the required cooling capacity during the post accident condition.

The modeled containment accident fan cooler unit actuation from the containment analysis is based upon a response time associated with exceeding the containment Hi pressure setpoint to achieving full Containment Cooling System air and service water flow. The

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Operating each containment cooling unit's accident fan ensures that all accident fans are OPERABLE and that all associated indications are functioning properly. It also ensures that blockage, fan or motor failure, can be detected for corrective action. Acceptable performance is verified through verification of main control panel accident fan run indication, motor running amps, and clearing of low flow alarms. The 31 day Frequency was developed considering the known reliability of the accident fans and indications, the redundancy available, and the low probability of significant degradation of the accident fans occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that each containment accident fan cooler unit can achieve its assumed post accident flow rate with at least one containment accident fan cooler service water outlet valve open provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on a test line. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray and containment accident fan cooler service water outlet valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment Hi-Hi

BASES

REFERENCES

1. FSAR, Section 1.3.
2. 10 CFR 50, Appendix K.
3. FSAR, Section 14.
4. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to re-close once opened. The passive failure mode is failure to open upon demand. The MSSVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	The accident analysis requires that four MSSVs per steam generator be operable to provide overpressure protection for design basis transients occurring at 100% RTP plus power measurement uncertainty. The LCO requires that four MSSVs be OPERABLE in compliance with Reference 2 and the DBA analysis. The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the required tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program. This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.
APPLICABILITY	In MODES 1, 2, and 3, four MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization. In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.
ACTIONS	The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV. <u>A.1</u> With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

BASES

ACTIONS (continued)

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the Attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation" provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the required actions are not completed within the associated Completion Time, or if one or more steam generators have three or more inoperable MSSVs, 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 MODE 3 within 6 hours, and in MODE 4 within 12 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME OM Code (Ref. 4), requires that safety and relief valve tests be performed in accordance with Appendix I (Ref. 5). According to Reference 5, in addition to routine lift setpoint verifications, the following tests are required following equipment refurbishment:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

BASES

SURVEILLANCE REQUIREMENTS (continued)

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ±3% setpoint tolerance for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.1.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR, Section 14.1.9.
4. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.
5. ASME OM Code, Appendix I, Inservice Testing of Pressure Relief Devices in Light-Water Reactor Power Plants.
6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

BASES

ACTIONS (continued)

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable MSIVs or non-return check valves that cannot be restored to OPERABLE status within the specified Completion Time, but are isolated, the flowpath must be verified on a periodic basis to be closed and the MSIV de-activated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of flowpath indications (MSIV position) available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs or non-return check valves cannot be restored to OPERABLE status or are not closed within the associated Completion 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 at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds, as measured from the time of signal initiation until the valves indicate closed. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 4), requirements during operation in MODE 1.

The Frequency is in accordance with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at the Frequency required by the Inservice Testing Program. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 2 under low steam flow conditions (≤ 5% steam flow) at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODES 2 and 3 prior to performing the SR. This allows a delay of testing to

BASES

SURVEILLANCE REQUIREMENTS (continued)

establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV will actuate to its isolation position on a actuation isolation signal. The 18 month Frequency is based on a refueling cycle interval and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components normally pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that allows entry into and operation in MODES 2 and 3 prior to performing the SR. This allows delaying testing until conditions where the testing can be performed are established.

SR 3.7.2.3

This SR verifies that each main steam non-return check valve can close. As the non-return check valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 4), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at the Frequency required by the Inservice Testing Program. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.1.
2. FSAR, Section 14.2.5.
3. 10 CFR 100.11.
4. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.
5. NUREG-0800, Standard Review Plan 15.1.5, Appendix A, "Radiological Consequence of Main Steam Line Failures Outside of a PWR", Rev. 2, July 1981.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 3). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 3) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that performance of this SR for the turbine driven AFW pump is required to be completed within 24 hours after the unit exceeds 2% of RTP. This exception is required to prevent excessive RCS cooldowns as a result of steam draw from the steam generators during pump testing. This Note allows suitable test conditions to be established while allowing a reasonable time period to complete the SR during unit startups and low power operation.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each motor driven AFW pump discharge motor operated valve (AF-4020, 4021, 4022, and 4023) actuate to their correct positions on an actual or simulated actuation signal. This

BASES

SURVEILLANCE REQUIREMENTS (continued)

appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator supplied by the respective AFW pump system prior to exceeding 2% of RTP after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned.

REFERENCES

1. FSAR, Section 10.2.
2. FSAR, Section 14.1.10.
3. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.