DISCUSSION OF CHANGES
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.2.E specifies that the limiting conditions for operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-5. ITS LCO 3.4.5 explicitly specifies the RCS leakage detection instrumentation required to be Operable (i.e., Drywell Drain Sump Monitoring System, one channel of the drywell continuous atmospheric particulate system, and one channel of the drywell continuous atmospheric gaseous system). This change deletes a cross reference to a Table which is not included in the ITS and is therefore considered administrative. Similarly, reference to CTS Table 4.2-5 in CTS 4.2.E has been deleted since the CTS surveillances are included in the Surveillance Table of ITS 3.4.5. Any changes to any requirements in CTS Tables 3.2-5 and 4.2-5 are discussed below. This change is consistent with NUREG-1433, Revision 1.

A3 CTS Table 3.2-5 Note 2 is not retained in the ITS. Note 2 refers to another Specification for Action requirements (CTS 3.6.D), and need not be repeated in the ITS since the associated actions of this Specification have been incorporated in ITS 3.4.5. Since no technical requirements are altered, this change is considered administrative.

A4 CTS Table 4.2-5 Note 4, states that instrument checks are not required when these instruments are not required to be operable or are tripped. This Note is deleted in the ITS because the Surveillances to which the Note applies have been deleted and since there is no trip position for this instrumentation. Further, the intent of Note 4 is addressed in SR 3.0.1. Since no technical requirements are altered, this change is administrative and has no adverse impact on safety.

A5 The Instrument Functional Test Frequency of the Floor Drain Sump Flow Integrator identified in Note 1 to Tables 4.2-1 through 4.2-5 has been simplified to once every 31 days. The allowance to be able to change the surveillance frequency by submitting failure rate data to the NRC is always an option. Therefore, the removal of this allowance is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Table 4.6-2 does not require performance of an Instrument Functional Test, and requires that a Sensor Check be performed once per day for the Containment Atmosphere Monitoring System channels. ITS SR 3.4.5.1 and SR 3.4.5.2 require that a CHANNEL CHECK be performed at a Frequency of 12 hours, and a CHANNEL FUNCTIONAL TEST be performed at a Frequency of 31 days, respectively. This change imposes more frequent performance of the CHANNEL CHECK and adds the new requirement to perform a CHANNEL FUNCTIONAL TEST, which is more restrictive. These changes are necessary to ensure the equipment remains Operable and has no adverse affect on safety.

M2 CTS 3.6.D.4 requires the operability of the Primary Containment Sump Monitoring System and the Continuous Atmosphere Monitoring System. CTS 3.6.D.5 provides the appropriate actions if the Primary Containment Sump Monitoring System is inoperable and CTS 3.6.D.6 provides the appropriate actions if the Continuous Atmosphere Monitoring System is inoperable. CTS does not provide any restrictions if both the Primary Containment Sump Monitoring System and the Continuous Atmosphere Monitoring System (particulate and gaseous) are inoperable at the same time. CTS 3.6.D.4 and 3.6.D.5 can be entered at the same time. CTS 3.6.D is revised to add ITS 3.4.5 ACTION E, which requires that if all leakage detection systems are inoperable, ITS LCO 3.0.3 be entered immediately. This change is considered more restrictive on plant operation but is necessary since no required automatic means of monitoring LEAKAGE are available. | I | I

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details in CTS Table 3.2-6 that the "floor drain sump flow integrator" must be Operable and the details in CTS Table 4.2.5 that the "floor drain sump flow integrator" must be functionally checked and calibrated are proposed to be relocated to the Bases. The requirement in ITS LCO 3.4.5 that the drywell floor drain sump monitor system must be OPERABLE, the definition of Operability, and the requirements in SR 3.4.5.2 and SR 3.4.5.3 to perform a CHANNEL FUNCTIONAL TEST and CALIBRATION, respectively of the required leakage detection instrumentation suffice. The flow integrator is part of this system. Therefore these details are included in the Bases and are not required to be in the Specification to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 The details in CTS Table 3.2-6 Note 1 that the two flow integrators, one for the equipment drain sump and the other for the floor drain sump, comprise the Basic Instrument System that monitors leakage detection inside the drywell are proposed to be relocated to the UFSAR. The requirements in ITS LCO 3.4.5 that the drywell floor drain sump monitoring system must be Operable, the associated Surveillances, and the definition of Operability will ensure that this portion of the system remains Operable. The requirements of the equipment drain sump flow integrator have been deleted in accordance with L5. However, the requirement to demonstrate Leakage is within limits is still maintained in SR 3.4.4.1. Therefore, the requirement for a means to quantify identified Leakage is adequately addressed by the requirements of ITS 3.4.4 and associated SR 3.4.4.1. As a result, this detail of what comprise the Basic Instrument System that monitors leakage detection inside the drywell is not necessary to be included in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.6.D.6 allows continued operation for 30 days if only one monitor of the Continuous Atmospheric Monitoring System (gaseous or particulate) is inoperable. The CTS also requires a grab sample to be taken every 24 hours during the 30 day period. If both are inoperable, a shutdown per CTS 3.0.C is required. ITS ACTION B will allow one of the two monitors to be inoperable indefinitely, provided SR 3.4.5.1, a CHANNEL CHECK, is performed on the remaining Operable Containment Atmospheric Monitoring System monitor, every 8 hours. ITS ACTION C will allow both Containment Atmospheric Monitoring System monitors to be inoperable for 30 days, provided a grab sample is taken every 12 hours. This change is acceptable since a diverse method to quantify increased leakage is still provided by the drywell floor drain sump monitoring system, and this is the primary method for quantifying leakage. In addition, when only one monitor is inoperable, a second monitor is still available to monitor the drywell atmosphere, and a CHANNEL CHECK is being performed more frequently than normal to ensure it is Operable. Also, to be consistent with the format of the ITS, the term "one channel" is used in the ITS in lieu of defining in the LCO section of the Bases that the System is OPERABLE if only one of the two gaseous monitors (channels) and one of the two particulate monitors (channels) are OPERABLE.

L2 CTS 3.6.D.5 requires that an inoperable sump monitoring system be restored to OPERABLE status within 24 hours. ITS 3.4.5, Required

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

Action A.1 requires an inoperable drywell sump monitoring system be restored to OPERABLE status within 30 days. This is a relaxation of requirements, and therefore less restrictive. This 30 day Completion Time is allowed provided RCS unidentified and total LEAKAGE are determined every 4 hours in accordance with SR 3.4.4.1. This change is acceptable based on operating experience, considering there is another method of leakage detection still available to monitor and assess RCS operational LEAKAGE (drywell continuous monitors) and since the RCS unidentified and total LEAKAGE can be quantified.

L3 A statement that LCO 3.0.4 is not applicable for the condition of the drywell floor drain sump monitoring system inoperable or both drywell atmospheric monitors inoperable has been added as a Note to CTS 3.6.D.5 and 3.6.C.6 (proposed ITS 3.4.5 ACTION A and ACTION C). When this allowance is used, either the drywell floor drain sump flow monitoring system or one or both drywell atmospheric monitors remains available, and the compensatory actions for the inoperable system (or the requirement that unidentified leakage be quantified in accordance with proposed LCO 3.4.5) will provide adequate indication of RCS leakage. Because 1) a 30 day allowed out of service time for one leakage detection system is acceptable based on industry operating experience; 2) a leakage detection system is still Operable; and 3) compensatory measures will still ensure leakage is being quantified, the LCO 3.0.4 exception is considered to not significantly impact safety and is acceptable.

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L4 The CTS Table 4.2-5 requirement that an instrument check be performed on the drywell floor drain sump monitor once per day is not adopted in the ITS. This is a relaxation of requirements, and is less restrictive. This change is acceptable because an instrument check is only a qualitative determination of OPERABILITY by observation of instrument behavior during operation, and simply observing the instrument does not provide sufficient information to determine OPERABILITY because the indication is not consistently the same. This is particularly true when there are no other instruments with which to compare indications. The indicator is a numerical digital readout only, and does not change unless a sump pumpout is in progress. The CHANNEL FUNCTIONAL TEST is the better indicator of OPERABILITY while operating, and this requirement is maintained in the ITS. This change is consistent with NUREG-1433, Revision 1.

L5 The drywell equipment drain sump monitoring system functions to quantify identified leakage. Since the purpose of ITS 3.4.5, RCS Leakage

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 (continued)

Detection Instrumentation, is to provide instrumentation requirements for early identification of unidentified leakage, the drywell equipment drain sump monitoring system requirements of CTS 3.6.D.4, 3.6.D.5, 4.6.D.4, Table 3.2-6, and Table 4.2-5 are proposed to be deleted. The drywell equipment drain sump monitoring system does not necessarily relate directly to the Leakage requirements (other means to quantify identified leakage are available, such as equipment drain sump pump-out times). Control of the availability of, and necessary compensatory activities if not available, for indications and monitoring instruments are addressed by plant operational procedures and policies. The requirement to demonstrate Leakage is within limits is still maintained in SR 3.4.4.1. As a result, the requirement for a means to quantify identified leakage is adequately addressed by the requirements of ITS 3.4.4 and associated SR 3.4.4.1. Therefore, explicit requirements for the drywell equipment drain sump monitoring system instrumentation are not required.

L6 A Note has been added to CTS 4.6.D.4 (Note to ITS 3.4.5 Surveillance Requirements) to allow a channel to be inoperable for up to 6 hours solely for performance of required Surveillances provided the other Leakage Detection System channel is OPERABLE. The 6 hour testing allowance has been granted by the NRC in Technical Specification amendment for Georgia Power Company's Hatch Unit 1 (Amendment 185) and Unit 2 (Amendment 125), in the ITS amendment for Washington Public Power Supply System Unit 2 (amendment 149), Nine Mile Point Unit 2 (Amendment 91, the ITS amendment), and LaSalle Units 1 and 2 (Amendments 147/133, respectively, the ITS amendments). The NRC has also granted this allowance in other topical reports for the Reactor Protection System, Emergency Core Cooling System, and Isolation System Instrumentation. The 6 hour testing allowance does not significantly reduce the probability of properly monitoring leakage since the other channel must be OPERABLE for this allowance to be used.

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TECHNICAL CHANGES - RELOCATIONS

None

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

The Licensee has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below. I

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow continued operation with inoperable leakage detection systems. The leakage detection systems are not considered as initiators of any previously evaluated accident. However, they do provide information to the operator of potential conditions that may be precursors to an accident. In the proposed conditions, sufficient indication will remain Operable to provide the operator with the information necessary to evaluate the potential precursor conditions. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, the leakage detection systems do not provide any accident mitigation functions. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve physical modification to the plant. The leakage detection systems provide information to the operator of potential conditions that may be precursors to an accident. However, under the proposed change, a diverse method to quantify increased leakage is still provided by the remaining Operable leakage detection system. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change would allow continued operation with inoperable leak detection systems. However, under the proposed change a diverse method to quantify increased leakage is still provided by the drywell floor drain sump monitoring system, and this is the primary method for quantifying leakage. In addition, grab samples of the containment atmosphere will be required once per 12 hours when all required drywell continuous atmospheric monitoring systems are inoperable and a CHANNEL CHECK on the remaining drywell continuous atmospheric monitor will be I

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. (continued)

required every 8 hours when one of the two drywell continuous atmospheric monitors is inoperable. Therefore, this change does not involve a significant reduction in a margin of safety since the proposed LCO will maintain adequate indications to the operator, and in addition will continue to provide appropriate compensatory measures.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the time the drywell floor drain sump monitor is permitted to be inoperable from 24 hours to 30 days. The sump monitor is not assumed to be the initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. Another form of leakage detection is still available during the extended interval, and RCS unidentified and total leakage must be determined every 4 hours in accordance with SR 3.4.4.1. This change will not alter assumptions relative to the mitigation of an accident or transient event since the drywell floor drain sump monitoring system is not required to operate during an accident. Therefore, allowing 30 days to comply with the LCO will not significantly affect the consequences of an accident. The 30 days will allow time to restore the drywell floor drain sump monitoring system to OPERABLE status and possibly avoid a shutdown. Shutting down the plant is a transient which puts thermal stress on components. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the time a drywell sump monitor is permitted to be inoperable. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

3. Does this change involve a significant reduction in a margin of safety?

   This change allows the drywell floor drain sump flow monitoring system to be inoperable for up to 30 days. The margin of safety is not significantly reduced because the chance of an event occurring while in this condition is remote. The 30 days allows more time to comply with the LCO instead of having to shutdown. A reduction in power is considered a transient due to the thermal effects it has on plant equipment. In addition, at least one channel of either the drywell continuous atmospheric particulate or atmospheric gaseous monitoring system must be operable. This channel is able to detect increase Leakage rates of 1 gpm within 1 hour. In addition, RCS unidentified and total Leakage must be determined every 4 hours in accordance with SR 3.4.4.1. Therefore, RCS Leakage will be detected and quantified during this 30 day period. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

The Licensee has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below. 1 △I

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change will permit MODE changes when either the drywell floor drain sump monitor or both drywell continuous atmospheric monitors is inoperable. 1 △I The inoperability of RCS leakage detection instrumentation is not considered to be the initiator of any transient or accident. Therefore, the probability of an accident previously evaluated is not significantly increased. However, the RCS leakage detection instrumentation do provide the type of information that could be related to a precursor to an accident. In the proposed change, multiple forms of leakage detection instrumentation will be OPERABLE such that adequate information will be available to evaluate potential precursor conditions. Additionally, leakage detection systems do not function in any accident mitigation capacity. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change will permit MODE changes when leakage detection equipment is inoperable. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will permit MODE changes when one required leakage detection instrument is inoperable. The proposed LCO will maintain adequate indication for the operator, and in addition will continue to

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

3. (continued)

provide appropriate compensatory measures for leakage monitoring. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the requirement to perform instrument checks on the floor drain sump instrumentation. This system consists of monitoring instrumentation only and does not initiate any automatic actuations or isolations during any analyzed accident. The leakage detection systems are not considered as initiators of any previously evaluated accident. However, they do provide information to the operator concerning potential conditions that may be precursors to an accident. The remaining Surveillances will still ensure that the instrumentation remains Operable. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Because the leakage detection systems do not provide any accident mitigation functions, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change deletes the requirement to perform instrument checks on the floor drain sump flow instrumentation. The instrumentation is still tested and maintained operable through Channel Functional Tests and Channel Calibrations. In addition, proposed SR 3.4.4.1 will require the use of the floor drain sump integrators to determine the actual leakage rate every 12 hours. This should minimize the potential for an undetected failure of the integrator. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change does not result in any hardware or operating procedure changes. The drywell equipment drain sump monitoring system is not assumed in the initiation of any analyzed event. The drywell equipment drain sump monitoring system functions to quantify identified leakage. The drywell equipment drain sump monitoring system does not necessarily relate directly to the Leakage requirements (other means to quantify identified leakage are available, such as equipment drain sump pump-out times). The requirement to demonstrate Leakage (including identified leakage) is within limits is still maintained in SR 3.4.4.1. As a result, the requirement for a means to quantify identified leakage is adequately addressed by the requirements of ITS 3.4.4 and associated SR 3.4.4.1. Explicit requirements for the drywell equipment drain sump monitoring system instrumentation are not required. As a result, accident consequences are unaffected b the deletion of the drywell equipment drain sump monitoring system requirements. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed deletion of the drywell equipment drain sump monitoring system requirements does not impact any margin of safety. The drywell equipment drain sump monitoring system functions to quantify identified leakage. The drywell equipment drain sump monitoring system does not necessarily relate directly to the Leakage requirements (other means to quantify identified leakage are available, such as equipment drain sump

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

3. (continued)

pump-out times). The requirement to demonstrate Leakage (including identified leakage) is within limits is still maintained in SR 3.4.4.1. As a result, the requirement for a means to quantify identified leakage is adequately addressed by the requirements of ITS 3.4.4 and associated SR 3.4.4.1. As a result, an explicit requirement to maintain the drywell equipment drain sump monitoring system Operable as a means of quantifying identified leakage is not required. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L6 CHANGE

The Licensee has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change modifies the Surveillance to indicate when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required Leakage Detection System channel is OPERABLE. The Leakage Detection System Instrumentation is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Also, this change does not further degrade the capability of the monitors to perform their required function under these circumstances since one channel is still OPERABLE. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

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2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This does not involve a significant reduction in a margin of safety since the monitors are not required to provide automatic repsonse to any design basis accident. The additional time does not significantly affect the contribution of the monitors to risk reduction since the function is still being monitored by the other OPERABLE channel.

[CTS]

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~6~~ RCS Leakage Detection Instrumentation

5 PA1

[3.6.D.4] LCO 3.4.~~6~~ 5 The following RCS leakage detection instrumentation shall be OPERABLE:

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PA2

a. Drywell floor drain sump monitoring system; ~~and~~

b. One channel of the drywell continuous ~~One channel of either primary containment~~ atmospheric particulate ~~or atmospheric gaseous~~ monitoring system; ~~and~~ and

c. One channel of the drywell continuous atmospheric gaseous ~~Primary containment air cooler condensate flow rate~~ monitoring system~~s~~.

[3.6.D.4] APPLICABILITY: MODES 1, 2, and 3.

ACTIONS TA1 (not shown)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.D5] A. Drywell floor drain sump monitoring system inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ---------------------------- A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	 30 days

(continued)

Revision I

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.D.6] PA2 ~~B~~ C. ~~Required primary containment~~ Both drywell continuous atmospheric monitoring system~~s~~ inoperable.	X1 ---NOTE--- LCO 3.0.4 is not applicable. --- ~~B~~ C.1 Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND LB2 ~~B~~ C.2 Restore ~~required primary containment~~ ONE drywell continuous atmospheric monitoring system to OPERABLE status.	30 days
[L1] ~~C~~ B. ~~Primary containment air cooler condensate flow rate~~ One drywell continuous atmospheric monitoring system inoperable.	~~C~~ B.1 ---NOTE--- Not applicable when ~~required primary containment~~ both drywell continuous atmospheric monitoring system~~s~~ ~~is~~ are inoperable. --- Perform SR 3.4.~~6~~ 5.1. PA1	PA2 Once per 8 hours

(continued)

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Revision I

PA1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D. Required primary containment atmospheric monitoring system inoperable.~~ ~~**AND**~~ ~~Primary containment air cooler condensate flow rate monitoring system inoperable.~~	~~NOTE~~ ~~LCO 3.0.4 is not applicable.~~ ~~D.1 Restore required primary containment atmospheric monitoring system to OPERABLE status.~~ ~~**OR**~~ ~~D.2 Restore primary containment air cooler condensate flow rate monitoring system to OPERABLE status.~~	~~30 days~~ ~~30 days~~
~~E~~ D. Required Action and associated Completion Time of Condition A, B, C, or ~~D~~ not met.	~~E~~ D.1 Be in MODE 3. **AND** ~~E~~ D.2 Be in MODE 4.	12 hours 36 hours
~~F~~ E. All required leakage detection systems inoperable.	~~F~~ E.1 Enter LCO 3.0.3.	Immediately

PA2

[3.6.D.5]
-6.D.6]

PA2

[M2]

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Revision I

PA1

CTS

SURVEILLANCE REQUIREMENTS

drywell continuous atmospheric PA2

	PA1 SURVEILLANCE	FREQUENCY
[T4.6-2]	SR 3.4.⑥.1 ⑤ Perform a CHANNEL CHECK of ~~required~~ primary containment atmospheric monitoring system. s PA2	12 hours
[T4.6-2] [T4.2-5]	SR 3.4.⑥.2 ⑤ Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
[T4.6-2] [T4.2-5]	SR 3.4.⑥.3 ⑤ Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.	~~[18] months~~ CLB3 92 days

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[L6]

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When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required leakage detection instrumentation is OPERABLE.

X2

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Revision I

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 Not used.

CLB2 The brackets have been removed and ITS 3.4.5 Required Action B.2 retained in accordance with CTS 3.6.D.6.

CLB3 The brackets have been removed and the Frequency modified consistent with the requirements in CTS Tables 4.6-2 and 4.2-5.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433, Revision 1 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage", has not been incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Changes have been made to reflect the plant specific nomenclature and number.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 TSTF-60, Revision 0, changes are not incorporated in ITS 3.4.5 (NUREG-1433 Specification 3.4.6) since ITS 3.4.5 Required Action D.1 (NUREG-1433 Specification 3.4.6, Required Action F.1) requires entry into ITS LCO 3.0.3, and a plant shutdown, when all required leakage detection systems are inoperable. As a result, it is inappropriate to allow the MODE change restrictions to not be applicable while in ITS 3.4.5 Condition D (moving the placement of the Note, per TSTF-60, would allow MODE changes while in the ACTIONS of ITS 3.4.5).

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 The brackets have been removed and the exceptions to LCO 3.0.4 included as justified in L3.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

<u>DIFFERENCE FOR OTHER REASONS THAN ABOVE</u> (X)

X2 A Note has been added to allow a channel to be inoperable for up to 6 hours solely for perfomrance of required Surveillances, provided the other Leakage Detection System instrumentation is OPERABLE. This Note is similar to other Notes in the ITS, which allow channels that provide automatic actions to be inoperable for up to 6 hours. This instrumentation only provides indication, and the 6 hour allowance is not allowed unless the other channel is OPERABLE. This change has previously been approved by the NRC in TS amendments for Hatch Units 1 and 2 (amendments 185 and 125, respectively), WNP-2 (amendment 149, the ITS amendment), NMP2 (amendment 91, the ITS amendment), and LaSalle Units 1 and 2 (amendments 147 and 133, respectively, the ITS amendments). The 6 hour testing allowance does not significantly reduce the probability of properly monitoring leakage since the other channel must be OPERABLE for this allowance to be used.

I

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 7)

LCO PA3

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, ~~either~~ the flow monitoring ~~or the sump level monitoring~~ portion of the system must be OPERABLE since this portion is capable of quantifying unidentified LEAKAGE from the RCS. ~~The other monitoring systems~~ (One channel each of the drywell continuous atmospheric particulate and drywell continuous atmospheric gaseous monitoring systems) provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded. CLB1 I

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the ~~primary containment~~ drywell continuous atmospheric ~~activity~~ monitor~~s~~ ~~and the primary containment air cooler condensate flow rate monitor~~ will provide indication of changes in leakage. PA2 DB2 I

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every ~~8~~ 4 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage. PA4

(continued)

Revision I

BASES

ACTIONS (continued)

C D.1 and D.2 PA3

required PA2 drywell continuous

With both gaseous and particulate ~~primary containment~~ atmospheric monitoring channels inoperable, grab samples of the ~~primary containment~~ drywell PA2 atmosphere must be taken and analyzed to provide periodic leakage information. [Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the ~~required~~ two PA2 monitors.] ~~[Provided a sample is obtained and analyzed every 12 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., air cooler condensate flow rate monitor) is available.]~~ DB3

I

I

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

I

PA2

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate ~~primary containment~~ drywell continuous atmospheric monitoring ~~channels~~ systems are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

I

~~[C.1~~ B

one required drywell continuous atmospheric

~~With the required primary containment air cooler condensate flow rate monitoring system inoperable, SR 3.4.6.1 must be~~ channel drywell 5 ~~performed every 8 hours to provide periodic information of activity in the primary containment at a more frequent interval than the routine Frequency of SR 3.4.6.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if the required primary containment atmospheric monitoring systems is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.]~~ PA1 PA3

for the remaining OPERABLE drywell continuous atmospheric monitoring channel

are

both drywell continuous

I

(continued)

Revision I

BASES

ACTIONS (continued)

~~D.1 and D.2~~

~~With both the primary containment gaseous and particulate atmospheric monitor channels and the primary containment air cooler condensate flow rate monitor inoperable, the only means of detecting LEAKAGE is the drywell floor drain sump monitor. This condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.~~

~~The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels and air cooler condensate flow rate are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~

PA3 I

PA3 D ~~E~~.1 and ~~E~~.2

I

and

If any Required Action of Condition A or B ~~, C, or D~~ cannot be met ~~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 at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

PA3 PA3

PA3 E ~~F~~.1

I

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

(continued)

Revision I

BASES (continued)

Insert SR Note
X2

I

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1 5 PA1

This SR is for the performance of a CHANNEL CHECK of ~~the~~ Channels required ~~primary containment~~ drywell continuous PA2 atmospheric monitoring ~~system~~. The check gives reasonable confidence that the channels ~~is~~ are PA3 operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.6.2 5 PA1

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument ~~string~~ channel PA5. INSERT A TA2 The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.6.3 5 PA1

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument ~~string including the instruments located inside containment~~. DB2 The Frequency ~~of [18] months is a typical refueling cycle and considers channel reliability~~ is 92 days and LF3. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 30.~~
2. ~~Regulatory Guide 1.45, May 1973.~~
3. ~~FSAR, Section [5.2.7.2.1].~~
4. ~~GEAP-5620, April 1968.~~
5. ~~NUREG-75/067, October 1975.~~
6. ~~FSAR, Section [5.2.7.5.2].~~

INSERT REF DB4

Revision I

Insert SR NOTE

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required instrumentation (the drywell floor drain sump monitoring system or drywell continuous atmospheric monitoring channel, as applicable) is OPERABLE. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

INSERT A

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 FUNCTIONAL 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.

Insert REF

1. UFSAR, Section 16.6.
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
3. UFSAR, Section 4.10.
4. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flows, General Electric Company, April 1968.
5. NUREG-75/067, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactors, October 1975.
6. UFSAR, Section 16.3.
7. 10 CFR 50.36(c)(2)(ii).

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Bases have been revised to reflect the current requirements in CTS 3.6.D.4. The sump level monitoring portion of the system is not required to be Operable.

CLB2 Not used. | △I

CLB3 The SR 3.4.3.5 Bases have been revised to reflect current Calibration Frequencies in CTS Tables 4.6-2 and 4.2-5.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433, Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," has not been incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Changes have been made to reflect the plant specific nomenclature.

PA3 The Bases have been revised to reflect the Specification. | △I

PA4 The Frequency has been changed to be consistent with the proposed Frequency in SR 3.4.4.1.

PA5 Editorial changes have been made to be consistent with the terminology used in other parts of the Bases.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plants. The JAFNPP Construction Permit was issued on May 20, 1970. The proposed General Design Criteria (GDC) were initially published for comment in the Federal Register on July 11, 1967 (32 FR 10213) and published in final form in the Federal Register on February 20, 1971 (36 FR 3256), and amended on July 7, 1971 (36 FR 12733). UFSAR, Section 16.6, "Conformance to AEC Design Criteria," describes the JAFNPP current licensing basis with regard to the GDC. ISTS statements concerning the GDC are modified in the ITS to reference UFSAR, Section 16.6.

DB2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB3 Not used. | I

DB4 The References have been revised to reflect the plant specific References. The Bases have been revised to reflect any numbering changes.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 TSTF-60 revisions are not incorporated in ITS 3.4.5 (NUREG-1433 Specification 3.4.6) since ITS 3.4.5 Required Action D.1 (NUREG-1433 Specification 3.4.6, Required Action F.1) requires entry into ITS LCO 3.0.3, and a plant shutdown, when all required leakage detection systems are inoperable. As a result, it is inappropriate to allow the MODE change restrictions to not be applicable while in ITS 3.4.5 Condition D (moving the placement of the Note, per TSTF-60, would allow MODE changes while in the ACTIONS of ITS 3.4.5).

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 205 Revision 3 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 NUREG-1433, Revision 1, bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 The Bases have been modified to describe a Note added to the actual Specification. | I

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Leakage Detection Instrumentation

LCO 3.4.5 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain sump monitoring system; I

b. One channel of the drywell continuous atmospheric particulate monitoring system; and I

c. One channel of the drywell continuous atmospheric gaseous monitoring system. I

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ---------------------------- A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	 30 days
B. One drywell continuous atmospheric monitoring system inoperable.	------------NOTE------------ Not applicable when both drywell continuous atmospheric monitoring systems are inoperable. ---------------------------- B.1 Perform SR 3.4.5.1.	 Once per 8 hours

I

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C. Both drywell continuous atmospheric monitoring systems inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ----------------------------		I I
	C.1 Analyze grab samples of drywell atmosphere.	Once per 12 hours	I
	AND		
	C.2 Restore one drywell continuous atmospheric monitoring system to OPERABLE status.	30 days	I
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	12 hours	I
	AND		
	D.2 Be in MODE 4.	36 hours	I
E. All required leakage detection systems inoperable.	E.1 Enter LCO 3.0.3.	Immediately	I

SURVEILLANCE REQUIREMENTS

NOTE

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required leakage detection instrumentation is OPERABLE.

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Perform a CHANNEL CHECK of drywell continuous atmospheric monitoring systems.	12 hours
SR 3.4.5.2	Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
SR 3.4.5.3	Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.	92 days

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the flow monitoring portion of the system must be OPERABLE since this portion is capable of quantifying unidentified LEAKAGE from the RCS. The other monitoring systems (one channel each of the drywell continuous atmospheric particulate and drywell continuous atmospheric gaseous monitoring systems) provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell continuous atmospheric monitors will provide indication of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 4 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is

(continued)

ACTIONS

A.1 (continued)

allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1

With one required drywell continuous atmospheric monitoring channel inoperable, SR 3.4.5.1 must be performed every 8 hours of the remaining OPERABLE drywell continuous atmospheric monitoring channel to provide periodic information of activity in the drywell at a more frequent interval than the routine Frequency of SR 3.4.5.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if both drywell continuous atmospheric monitoring systems are inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.

C.1 and C.2

With both required gaseous and particulate drywell continuous atmospheric monitoring channels inoperable, grab samples of the drywell atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the two monitors.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate drywell continuous atmospheric monitoring systems are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If any Required Action and associated Completion Time of Condition A or B cannot be met, 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 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

E.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required instrumentation (the drywell floor drain sump monitoring system or drywell continuous atmospheric monitoring channel, as applicable) is OPERABLE. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

SR 3.4.5.1

This SR is for the performance of a CHANNEL CHECK of the required drywell continuous atmospheric monitoring channels. The check gives reasonable confidence that the channels are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument channel. 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 FUNCTIONAL 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 Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument channel. The Frequency is 92 days and operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Section 16.6.
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
3. UFSAR, Section 4.10.
4. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flows, General Electric Company, April 1968.
5. NUREG-75/067, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactors, October 1975.

(continued)

BASES

REFERENCES (continued)	6. UFSAR, Section 16.3.
	7. 10 CFR 50.36(c)(2)(ii).

DISCUSSION OF CHANGES
ITS: 3.4.6 - RCS SPECIFIC ACTIVITY

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 The requirement in CTS 4.6.C.1.b to perform an isotopic analysis of a sample of reactor coolant has been reworded to match the current wording in CTS 3.6.C.1 (ITS LCO 3.4.6). SR 3.4.6.1 will require the verification that the reactor coolant DOSE EQUIVALENT I-131 specific activity is ≤ 0.2 μCi/gm. This change is considered administrative since CTS 4.6.1.b is currently interpreted as requiring this evaluation. | △I

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new action has been added to CTS 3.6.C.1 (ITS Required Action A.1 and B.1) which will require the determination of DOSE EQUIVALENT I-131 every 4 hours whenever the DOSE EQUIVALENT I-131 specific activity limit is exceeded. This change, therefore, imposes additional requirements which are more restrictive but necessary to ensure the Reactor Coolant system specific activity is known. This will ensure the appropriate actions are taken if the activity is not reduced and the reactor coolant specific activity exceeds the current activity limit by more than a factor of 10. This change provides additional assurance that the source term assumed in the main steam line break (MSLB) analysis is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits, and that the thyroid dose to the control room operators is within the limits of GDC 19 of 10 CFR 50, Appendix A.

M2 CTS 3.6.C.1 requires that "the reactor shall be placed in the cold shutdown condition within 24 hours" if the iodine concentration exceeds the equilibrium limit by more than a factor of 10. These requirements are proposed to be replaced by ITS 3.4.6 Required Actions B.2.2.1 which requires the plant to be in MODE 3 within 12 hours under the same conditions (see comment L2 for Completion Time to MODE 4). Based on operating experience, this Completion Time limit still allows for an orderly transition to MODE 3 without challenging plant systems. This change is more restrictive because it provides an additional requirement to place the plant in MODE 3 in 12 hours vice 24 hours but is necessary

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~7~~ ⑥ RCS Specific Activity

PA1

[3.6.c.1] LCO 3.4.~~7~~ ⑥ The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ ~~[~~0.2~~]~~ μCi/gm. DB1

I

[Doc L3] APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.c.1] [M1] [L4]	A. Reactor coolant specific activity > ~~[~~0.2~~]~~ μCi/gm and ≤ ~~4~~.0 ② μCi/gm DOSE EQUIVALENT I-131. DB1 DB2	-------------NOTE----------- LCO 3.0.4 is not applicable. ---------------------------	
		A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
		AND	
		A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
[3.6.c.1] [M1] [L3]	B. Required Action and associated Completion Time of Condition A not met. OR Reactor ~~c~~oolant specific activity > ~~[5.0]~~ ② μCi/gm Dose EQUIVALENT I-131. DB1	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
		AND	
		B.2.1 Isolate all main steam lines.	12 hours
		OR	
			(continued)

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I

Typ. All pages

Revision I

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.6 - RCS SPECIFIC ACTIVITY

<u>RETENTION OF EXISTING REQUIREMENT</u> (CLB)

None

<u>PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT</u> (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

1△I

<u>PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS</u> (DB)

DB1 The brackets have been removed and the plant specific JAFNPP value has been included.

DB2 Changes have been made to reflect the appropriate limit applicable to JAFNPP.

<u>DIFFERENCE BASED ON APPROVED TRAVELER</u> (TA)

None

<u>DIFFERENCE BASED ON PENDING TRAVELER</u> (TP)

None

<u>DIFFERENCE FOR OTHER REASONS THAN ABOVE</u> (X)

None

1B1

BASES

APPLICABLE SAFETY ANALYSES (continued)

outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 100.

The limits on the specific activity of the primary coolant also ensure the thyroid dose to the control room operators, resulting from an MSLB outside containment during steady state operation will not exceed the limits specified in GDC 19 of 10 CFR 50, Appendix A. (Ref. 3)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 4). X1

LCO

The specific iodine activity is limited to ≤ [0.2] μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits. DB2 | I

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ [4.0] μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion ② DB1 | I

(continued)

Revision I

BASES

ACTIONS

A.1 and A.2 (continued)

Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2 (2) DB1 I

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > ~~4~~.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100.11 and GDC 19 of 10 CFR 50 Appendix A (Ref. 3) during a postulated MSLB accident. DB1

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the ~~unit~~ plant in MODES 3 and 4 are reasonable, based on operating

PA2

(continued)

Rawson I

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.6 - RCS SPECIFIC ACTIVITY

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Changes have been made (additions, deletions and/or changes to the NUREG) to reflect the plant specific nomenclature.

| △I

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The Bases have been modified to reflect the plant specific analysis.

DB2 The brackets have been removed and the proper plant specific JAFNPP value has been provided.

DB3 The brackets have been removed and the proper plant specific References has been provided.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity $\leq$ 0.2 μCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 0.2 μCi/gm and $\leq$ 2.0 μCi/gm DOSE EQUIVALENT I-131.	------------NOTE---------- LCO 3.0.4 is not applicable. --------------------------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met. OR Reactor coolant specific activity > 2.0 μCi/gm DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	B.2.1 Isolate all main steam lines.	12 hours
	OR	
		(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

10 CFR 100. The limits on the specific activity of the primary coolant also ensure the thyroid dose to the control room operators, resulting from an MSLB outside containment during steady state operation will not exceed the limits specified in GDC 19 of 10 CFR 50, Appendix A (Ref. 3).

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

<u>A.1 and A.2</u>

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 2.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 2.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100.11 and GDC 19 of 10 CFR 50 Appendix A (Ref. 3) during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the

(continued)

DISCUSSION OF CHANGES
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.6.A does not state any Applicability requirements. ITS 3.4.9 is Applicable "At all times". Because the CTS does not specifically state Applicability requirements, and the limitations imposed apply at all times, it can be implied that the Specification is also Applicable "At all times." Since no technical requirements are altered, this change is administrative and has no adverse impact on safety.

A3 CTS 3.6.A.5.a is revised by adding a NOTE (ITS 3.4.9 Condition A Note) which requires that a determination be made whether the RCS is acceptable for continued operation whenever the Condition is entered, regardless of whether compliance with the LCO is restored. This change only provides clarification, because CTS 3.6.A.5.a already contains this requirement. Since no technical requirements are altered, this change is administrative and has no adverse impact on safety.

A4 CTS 3.6.A.5 provides actions appropriate for placing the facility in a condition outside the MODE(S) of Applicability when the Applicability is MODES 1, 2, and 3. Since certain PT limits apply even when not in MODES 1, 2, and 3, Action C was added (refer to DOC M4). To clarify the use and application of applying the appropriate action depending on the MODE of operation, the specific clarification "in MODES 1, 2, or 3" is added. No technical requirements are altered, this change is administrative and has no adverse impact on safety. I

A5 Not used.

ACTIONS (continued)

[M4]

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

[4.6.A.2] [4.6.A.3] [3.6.A.2] [3.6.A.3] [3.6.A.4] INSERT SR-1

[4.6.A.4]

SURVEILLANCE		FREQUENCY
SR 3.4.~~10~~ 9 .1 PA1	---------NOTE--------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --------------------- Verify: a. RCS pressure, and RCS temperature~~, and RCS heatup and cooldown~~ rates are within the limits specified in ~~the PTLR.~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable.	30 minutes
SR 3.4.~~10~~ 9 .2 PA1	Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~the PTLR.~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable CLB1	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

CLB1 | I

| I

(continued)

Revision I

CLB1

Insert SR-1

b. RCS temperature change avereaged over a one hour period is:

1. ≤ 100°F when the RCS pressure and RCS temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; |△I

2. ≤ 20°F when the RCS pressure and RCS temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and |△I

3. ≤ 100°F during other heatup and cooldown operations.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 JAFNPP has not developed the "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)". References to limits in the PTLR are replaced with current requirements.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage", is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Editorial changes have been made to achieve consistency with the Writer's Guide for the Restructured Technical Specifications.

1 I

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The bracketed allowance has been deleted since it does not apply to JAFNPP.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 35, Revision 0, have been incorporated.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 ITS SR 3.4.9.4 has been added to the requirements of ISTS 3.4.10 (ITS 3.4.9) to allow an alternative to the requirements of ITS SR 3.4.9.3. This Surveillance has been added to the CTS in accordance with L1. A Note 2 was added to SR 3.4.9.3 which allows the option to perform SR 3.4.9.4. In addition, subsequent Surveillances have been renumbered, as required.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	------------------NOTE------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --- Verify: a. RCS pressure and RCS temperature are within the limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable. b. RCS temperature change averaged over a one hour period is: 1. ≤ 100°F when the RCS pressure and RCS temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; 2. ≤ 20°F when the RCS pressure and RCS temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and 3. ≤ 100°F during other heatup and cooldown operations.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	-------------------NOTES----------------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be performed if SR 3.4.9.4 is satisfied. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	-----------------NOTES------------------ 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be met if SR 3.4.9.3 is satisfied. -- Verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes.	Once within 15 minutes prior to each startup of a recirculation pump

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(continued)

SUMMARY OF CHANGES TO ITS SECTION 3.5 - REVISION I

Source of Change	Summary of Change	Affected Pages
BSI-F6	The change agreed to during communications with the NRC concerning this issue have been made. Specifically, the LCO Section of the Bases have been modified to clearly state that both pumps per subsystem are required for LPCI subsystem Operability.	**Specification 3.5.1** NUREG Bases markup p B 3.5-5 Retyped ITS Bases p B 3.5-5
Consistency issue	The pressure at which adequate steam pressure is available to perform the HPCI and RCIC low pressure tests (i.e., the pressure at which the 12 hour Note allowance starts) has been changed to > 150 psig, consistent with the pressure allowed in SR 3.5.1.10 and SR 3.5.3.6. This pressure is also consistent with the Applicability for HPCI and RCIC. (The test pressure in the two SRs (SR 3.5.1.9 and SR 3.5.3.5) is not being changed; it is still ≤ 165 psig.)	**Specification 3.5.1** NUREG Bases markup p B 3.5-12 Retyped ITS Bases p B 3.5-13 **Specification 3.5.3** NUREG Bases markup p B 3.5-27 Retyped ITS Bases p B 3.5-30
CTS markup error	Minor editorial correction has been made to the CTS markup. (The words "and 3.5.E" in the last line of CTS 3.5.G.1 have been annotated with "see ITS 3.5.3".)	**Specification 3.5.1** CTS markup p 9 of 17
Typographical errors	Minor typographical errors in the Discussion of Changes have been corrected. (DOC A7, "CTS 3.5.A.b.4" changed to "CTS 3.5.A.4.b"; DOC A10, "(e.g., LPCI and CS)" changed to "(i.e., LPCI, CS, and HPCI)" and "Specifications 3.5.A, 3.5.C and 3.5.E" changed to "3.5.A and 3.5.C"; and DOC M9, "ITS SR 3.4.3.2" changed to "ITS SR 3.5.1.13" and in the first sentence, the word "manual" has been changed to "manually.")	**Specification 3.5.1** DOCs A7, A10, and M9 (DOCs p 3 of 24, 4 of 24, and 9 of 24)
Editorial correction	The DOC describing the administrative change to CTS 3.9.F.2.a has been modified to more accurately describe the change.	**Specification 3.5.1** DOC A12 (DOCs p 5 of 24)
Consistency issue	The Discussion of Change has been modified to be consistent with the actual ITS Note (i.e., the phrase "in MODE 3 with reactor steam dome pressure less than the RHR permissive pressure" has been deleted since the Note only applies in MODES 4 and 5).	**Specification 3.5.2** DOC A2 (DOCs p 1 of 9)
RAI 3.5.3-1	The changes based on the JAFNPP response to RAI 3.5.3-1 were made in Revision D. However, it was later noted that the CTS markup pages were not annotated properly nor submitted to the NRC as part of Revision D. This change completes the changes based on the JAFNPP response to RAI 3.5.3-1.	**Specification 3.5.1** CTS markup p 3 of 17 **Specification 3.5.3** CTS markup p 2 of 4
Typographical error	Minor typographical error in the Discussion of Changes has been corrected. (DOC A3, "injection/spray" changed to "injection" in two places.)	**Specification 3.5.3** DOC A3 (DOCs p 1 of 7)

Specification 3.5.1

A1

3.5.1 ECCS - Operating

3.5 (cont'd)

DELETED

[3.5.1] C. **HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)**

[Note 1 to SR 3.5.1.10]

A2

[LCO 3.5.1] [Applicability] 1. The HPCI System shall be operable whenever the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel, except as specified below:

MODE 1
MODE 2 and 3, except HPCI not required to be OPERABLE with reactor steam dome ≤ 150 psig

4.5 (cont'd)

C. **HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)**

M1

[SR 3.5.1.8
SR 3.5.1.9 Note]

Surveillance of HPCI System shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within 10 days of continuous operation from the time steam becomes available.

12 hours

A3 L1

See ITS: 3.3.5.1

LA7

[SR 3.5.1.2]
[SR 3.5.1.8]
[SR 3.5.1.10]

1. HPCI System testing shall be as specified in 4.5.A.1 a, b, c, d, f, and g except that the HPCI pump shall deliver at least 4,250 gpm against a system head corresponding to a reactor vessel pressure of 1,195 psig to 150 psig.

3400

L7

with reactor pressure ≥ 970 and ≤ 1040 psig

M2

L15

[SR 3.5.1.9]

with reactor pressure ≤ 165 psig

I

Page 3 of 17

Specification 3.5.J

A1

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes of satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

ND

See ITS: 3.5.3

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. These values are specified in the Core Operating Limits Report. If at anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

A6

2. Following any period where the LPCI subsystems or core spray subsystems have not been maintained in a filled condition, the discharge piping of the affected subsystem shall be vented from the high point of the system and water flow observed.

LA6

1

SR 3.5.1.1

See ITS 3.5.3

M11

3. Whenever the HPCI or RCIC System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI or RCIC shall be vented from the high point of the system, and water flow observed on a monthly basis.

LA6

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to ensure they are full shall be functionally tested each month.

LB1

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥25% rated thermal power.

See ITS: 3.2.1

ADMINISTRATIVE CHANGES

A6 (continued)

SR 3.0.2." Thus, anytime where these subsystem or systems had not been maintained in a filled condition SR 3.0.1 would require that the subsystems or systems be verified filled prior to declaring the subsystems or systems operable. Therefore, this change is not a technical change and is considered administrative. The change is consistent with NUREG-1433, Revision 1.

A7 CTS 3.5.A.4.a requires that the "reactor shall not be started up with the RHR System supplying cooling to the fuel pool." CTS 3.5.A.4.b requires that "the RHR System shall not supply cooling to the spent fuel pool when the reactor coolant temperature is above 212°F." | △I

In the proposed ITS presentation the ability to change MODES is generically controlled by the provisions of LCO 3.0.4 which states in part that "when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time." LCO 3.5.1, Condition A and Condition B, requires that the LPCI mode of RHR be Operable in MODES 1, 2, and 3, or if inoperable the reactor would be required to shutdown in 7 days, 12 hours.

Therefore LCO 3.0.4 would prevent plant startup with a LPCI subsystem inoperable (i.e., supplying cooling to the fuel pool results in LPCI being inoperable). Likewise, if the reactor coolant temperature is above 212°F, with the plant in MODE 1, 2, or 3 by definition of the MODEs Table (Table 1.1-1), both subsystems of LPCI would be required to be Operable, and therefore a loop of RHR could not be used to supply cooling water to the fuel pool. Therefore, this proposed change causes no technical or actual change from present specifications. Therefore, the change is considered administrative, and is consistent with NUREG-1433, Revision 1.

A8 A new ACTION has been added to CTS 3.5.A (for the Core Spray Systems and Low Pressure Coolant Injection Systems), CTS 3.5.C (for the High Pressure Coolant Injection System) and CTS 3.5.D (for the Automatic Depressurization System (ADS)) for all other conditions not addressed in the current Specification or in ITS 3.5.1 Conditions A, C, D, E, or F. With so many ECCS Systems inoperable the plant is considered to be outside it design bases and entry into CTS 3.0.C will be required. ITS 3.5.1 ACTION H is being proposed which will require immediate entry into LCO 3.0.3 (Required Action H.1) under the same conditions. Since the current Technical Specifications will also require entry into CTS

ADMINISTRATIVE CHANGES

A8 (continued)

3.0.C, this change is considered administrative. Changes to action requirements of CTS 3.0.C are covered in the Discussion of Changes for ITS LCO 3.0. In addition, CTS 3.9.F.3 requires that "the reactor shall be brought to cold condition within 24 hours" when both LPCI independent power supplies are made or found to be inoperable. This specific default action has been changed to require entry into LCO 3.0.3 since the plant will be outside of its design basis in the condition. This portion of the change may be considered as more restrictive but since the current Completion Times in CTS 3.9.F.3 and CTS 3.0.C are equivalent this change is classified as administrative. These changes are consistent with NUREG-1433, Revision 1.

A9 The requirements in CTS 3.5.A.3.b and CTS 4.5.A.3.b concerning the LPCI cross tie valves have been simplified into one Surveillance which requires the verification that the valves are closed and power is removed from the electrical valve operator every 31 days (ITS SR 3.5.1.4). The details on how this is performed have been relocated to the Bases in accordance with LA4. Since the current requirements in both CTS 3.5.A.3.b and 4.5.A.3.b require the valves to be closed and power to be removed, this change reflects a presentation preference and is considered administrative since identical requirements have been combined into one Surveillance (SR 3.5.1.4) in the ITS. This change is consistent with NUREG-1433, Revision 1.

A10 CTS 3.5.G.1 requires the associated ECCS pump (i.e., LPCI, CS, and HPCI) to be declared inoperable for the purposes of satisfying Specifications 3.5.A and 3.5.C, when the associated pump discharge piping cannot be maintained in a filled condition. This explicit cross reference is not required in ITS 3.5.1 since this CTS requirement is included along with the requirements of the associated system. Failure to meet this Surveillance will require direct entry into the appropriate ITS 3.5.1 ACTION(S). The Operability requirements in CTS 3.5.G.1 and 4.5.G.1 are directly incorporated in the required surveillances of ITS 3.5.1 (SR 3.5.1.1). ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance shall be a failure to meet the LCO. Therefore, incorporating the requirement to verify pump discharge piping is in the filled condition within the SRs associated with ECCS-Operating ensures the associated ECCS pump is declared inoperable when the surveillance is not met. Since there are no changes to any technical requirements this change is considered administrative. This change is consistent with NUREG-1433, Revision 1.

ADMINISTRATIVE CHANGES

A11 CTS 4.5.G.1 requires the discharge piping of the required ECCS subsystem to be vented every month prior to the testing of the LPCI and CS subsystems. This explicit requirement to perform the surveillance prior to the testing of the LPCI and CS subsystems has been deleted. The requirement to perform this surveillance every 31 days (ITS SR 3.5.1.1) is sufficient to ensure the discharge piping is full whenever the system is required to be Operable. This change is necessary since the ECCS subsystems flow rate Surveillances (e.g., CTS 4.5.A.1.b) are no longer tested every month. The Frequency of these Surveillances have been changed to "In accordance with the Inservice Testing Program" in recently approved Technical Specification Licensing Amendment 241. CTS 4.5.G.1 should have been modified during the process of the change. This will make the Surveillance consistent with other parts of the CTS and is therefore considered to be an administrative since the current Surveillance Frequency is every 31 days. This change is consistent with NUREG-1433, Revision 1.

A12 The requirement in CTS 3.9.F.2.a that operations may continue only if the other LPCI independent power supply battery including its battery charger, and distribution system is Operable has been deleted. The ITS includes ACTIONS for when two LPCI subsystems are inoperable, thus this ACTION covers the inverter and bus. In addition, the requirements of the battery and battery charger are included in ITS 3.8.4, while the requirements of battery cell parameters are included in ITS 3.8.6. ITS 3.8.4 requires the associated LPCI subsystem to be declared inoperable immediately when the LPCI independent power supply battery or battery charger are inoperable. When a battery cell parameter is not within limits, either the battery is still Operable, or if not then the ACTIONS of ITS 3.8.6 require the associated LPCI battery to be declared inoperable (which will result in the associated LPCI subsystem being declared inoperable). Therefore, the specific requirement is not necessary to be included in the ITS.

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A13 Not used.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M8 According to CTS 3.9.F.1, the reactor shall not be made critical unless both LPCI MOV Independent Power Supplies are operable which is effectively MODES 1 and 2. ITS 3.5.1 requires the low pressure core injection subsystems to be Operable in MODES 1, 2 and 3. Since the operability of the LPCI MOV Independent Power Supply effects the OPERABILITY of the associated LPCI subsystem, the operability requirements of LPCI MOV Independent Power Supplies have been extended to MODE 3. This ensures that each LPCI subsystem will remain operable with the required uninterruptable power supply during reactor conditions where there is significant core energy. This change is considered more restrictive and has no adverse effect on safety.

M9 CTS 4.6.E.4 requires the safety/relief valves to be manually opened every 24 months. ITS SR 3.5.1.13 requires this same manual opening but requires the actuation to be initiated on a Staggered Test Basis for each valve solenoid. This will ensure that a different solenoid will be used to actuate the valve every 24 months and is considered more restrictive since the current requirement does not specify which solenoid to use. This change is necessary to ensure both solenoids are tested within any 48 month period.

M10 CTS 3.5.A.5 requires all recirculation pump discharge valves to be Operable prior to reactor startup (or closed if permitted elsewhere in these specifications). ITS 3.5.1 and associated SR 3.5.1.6 also require all recirculation pump discharge valves to be Operable. However, if this requirement can not be met, then ITS SR 3.5.1.5 allows the associated recirculation pump discharge valve to be "de-energized" in the closed position. Requiring the inoperable recirculation pump discharge valve to also be "de-energized" in the closed position represents an additional restriction on plant operation. This change is necessary to ensure the proper flow path for the associated LPCI subsystem.

M11 CTS 4.5.G.3 requires the HPCI System discharge piping to be vented from the high point of the system whenever HPCI is lined up to take suction from the condensate storage tank (CST) on a monthly basis. In ITS SR 3.5.1.1 this requirement must be met whenever HPCI is required to be Operable whether it is aligned to the CST or the suppression pool. This change is considered more restrictive on plant operation but necessary to help prevent a water hammer following an initiation signal.

M12 CTS 3.5.A.1 and 3.5.A.3 require the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Systems, respectively to be Operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from cold shutdown (this covers MODES 1, 2 in the ITS). CTS 3.5.A

DB1 recirculation pump suction line

Division 2

BASES

LOCA due to a

APPLICABLE SAFETY ANALYSES (continued)

e. Adequate long term cooling capability is maintained.

DB1

The limiting single failures are discussed in Reference ~~1~~ 5. For a large ~~discharge~~ pipe break LOCA, failure of the ~~LPCI~~ ~~valve on the unbroken recirculation loop~~ is considered the most severe failure. For a small break LOCA, HPCI failure is the most severe failure. ~~One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation.~~ The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

125 VDC battery

Insert ASA

DB2

10 CFR 50.36(c)(2)(ii) Ref. 11

The ECCS satisfy Criterion 3 of ~~the NRC Policy Statement~~. X1

LCO

six DB2

Each ECCS injection/spray subsystem and ~~seven~~ ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

(which includes both pumps per subsystem)

PA1

PA1 active component

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

Alignment and operation for decay heat removal includes when the system is being realigned from or to the RHR shutdown cooling mode.

LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6 (continued)

startup prior to reaching > 25% RTP is an exception to the normal Inservice Testing Program generic valve cycling Frequency of 92 days, but is considered acceptable due to the demonstrated reliability of these valves. If the valve is inoperable and in the open position, the associated LPCI subsystem must be declared inoperable.

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9

PA1 at least

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established during preoperational testing.

PA4

against a system head corresponding to reactor pressure

Insert SR 3.5.1.7

PA7

DB9 970

one

main turbine generator load is greater than 100 MWe.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ ~~920~~ psig to perform SR 3.5.1.8 and ≥ ~~150~~ psig to perform SR 3.5.1.9. Adequate steam flow is represented by at least ~~1.25~~ turbine bypass valves open, or ~~total steam flow ≥ 10^6 lb/hr~~. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal

Adequate

I

DB9

PA7

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 5. For a LOCA due to a large recirculation pump suction line pipe break, failure of the Division 2 125 VDC battery is considered the most severe failure. For a small break LOCA, HPCI failure is the most severe failure. In the analysis of events requiring ADS operation, it is assumed that only five of the seven ADS valves operate. Since six ADS valves are required to be OPERABLE, the explicit assumption of the failure of an ADS valve is not considered in the analysis. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 11).

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems (which includes both pumps per subsystem), and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

I

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting Design Basis LOCA concurrent with the worst case single active component failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop at least the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested at both the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Adequate reactor steam pressure must be ≥ 970 psig to perform SR 3.5.1.8 and > 150 psig to perform SR 3.5.1.9. | △1 Adequate steam flow is represented by at least one turbine bypass valve open or main turbine generator load is greater than 100 MWe. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable.

(continued)

DISCUSSION OF CHANGES
ITS: 3.5.2 - ECCS - SHUTDOWN

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.5.F does not directly address the OPERABILITY status of LPCI during alignment and operation for decay heat removal. A Note has been added to CTS 4.5.F.5 (Note to ITS SR 3.5.2.4) which states that one LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. This allowance is consistent with the CTS 4.5.F Bases description. The Bases states that a LPCI subsystem operating in the shutdown cooling mode of RHR is considered operable for the ECCS function if it can be realigned manually (either remote or local) to the LPCI mode and is not otherwise inoperable. This allowance was approved in Licensing Amendment 168 which clarified and defined the ECCS requirements for when the plant is in the cold condition. Therefore, this change does not present any technical change from the current requirements, only a repositioning of clarifying information from the Bases to an SR Note. As such, the change is considered administrative.

I

A3 CTS 3.5.F.1 requires two low pressure Emergency Core Cooling subsystems to be Operable when work is being performed with the potential for draining the vessel. CTS 3.5.F.2 requires one low pressure Emergency Core Cooling subsystem to be Operable when no work is being performed with the potential for draining the reactor vessel. ITS 3.5.2 is identical although the format of presentation of these requirements are different. ITS LCO 3.5.2 requires two low pressure ECCS injection/spray subsystems to be Operable. It does not distinguish whether work is being performed with the potential for draining the reactor vessel (OPDRVs). If no OPDRVs are occurring and only one ECCS injection/spray subsystem is Operable, the Specification is met since ITS ACTION B allows continuous operation in this condition. Since this change does not change any existing requirements this change is considered administrative.

Specification 3.5.3

A1

3.5 (~~cont'd~~)

4.5 (~~cont'd~~)

[SR 3.5.3.4]

I L6 M3

with reactor pressure ≥ 970 psig and ≤ 1040 psig [SR 3.5.3.4]

with reactor pressure ≤ 165 psig, every 24 months [SR 3.5.3.5]

~~Item~~ SURVEILLANCE	Frequency
d. Flow Rate Test - The RCIC pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of 1195 psig to 150 psig.	Once per 92 Days
e. ~~Testable Check Valves~~	~~Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 92 days.~~
f. Logic System Functional Test	Once per 24 Months

LA4

See ITS: 3.3.5.2

[Required Action A.1]

2. When it is determined that the RCIC System is inoperable at a time when it is required to be operable, the HPCI System shall be verified to be operable immediately ~~and daily thereafter.~~

L4

Page 2 of 4

REVISION ~~D~~ I

DISCUSSION OF CHANGES
ITS: 3.5.3 - RCIC SYSTEM

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.5.G.1 requires the RCIC pump to be considered inoperable when the associated pump discharge piping cannot be maintained in a filled condition. This will require entry into CTS 3.5.E where 7 days (L1) is allowed to restore the RCIC System to Operable status. In the ITS, the requirement that the RCIC discharge piping must be filled is reflected in SR 3.5.3.1. Therefore, since this SR is directly related to the operability requirements of the RCIC System, this cross reference can be deleted and this change considered administrative. This change is consistent with NUREG-1433, Revision 1.

A3 CTS 4.5.E.1.a (ITS SR 3.5.3.6) is modified by Note 2 that excludes
vessel injection during the Surveillance. The Bases indicates that this | △I
test must include actuation of all automatic valves to their required
positions. Since all active components are testable and full flow can
be demonstrated by recirculation through the test line, coolant
injection into the RPV is not required during the Surveillance. This
Note, therefore, is explicit recognition that ITS SR 3.5.3.6 can be
satisfied by a series of overlapping tests. Since surveillance testing
of RCIC (CTS 4.5.E.1.a) does not presently require actual injection, and
is currently satisfied by a series of overlapping tests, the addition of
the Note excluding vessel injection is an administrative change. | △I

A4 Not used.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

CLB2
Insert SR3-A

CLB2

SR 3.5.3.3 ④ and SR 3.5.3.4 ⑤

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ 920 psig to perform SR 3.5.3.3 and ≥ 150 psig to perform SR 3.5.3.4. Adequate steam flow is represented by at least 1.25 turbine bypass valves open, or ~~total steam flow ≥ 10⁶ lb/hr~~. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. ~~Therefore~~, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

DB1
against a system head corresponding to reactor pressure

DB6
970 psig

PA2
Insert SR 3-B
④

PA2
Adequate

DB6
main turbine generator load is greater than 100 MWe

⑤ DB6
one

PA2

Ⓐ PA2

PA2
Insert SR 3-C

CLB2 ⑤ (24) X2

⑤

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The 18 month Frequency for SR 3.5.3.4 is based on the need to perform the Surveillance under conditions that apply ~~just prior to or~~ during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the

PA2

(24) X2

(continued)

REVISION ~~0~~ I

BASES (continued)

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 33~~ UFSAR, Section 16.6 DB4
2. UFSAR, Section ~~[5.5.6]~~ 4.7 DB5 PA3
3. ~~3~~ 4 Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975. PA4 X1

3 10 CFR 50.36(c)(2)(ii) X1

I

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3

During RCIC System operation, the RCIC System motor operated valves must reposition to ensure the RCIC System design function can be met. Cycling each motor specified valve through its range of motion (closed and open) ensures the valve will function when necessary. The functional tests ensure that the motor operated valves are capable of cycling open and closed within the required limits of operation. The Frequency of this SR is 92 days consistent with the requirements of the Inservice Testing Program.

SR 3.5.3.4 and SR 3.5.3.5

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Adequate reactor steam pressure must be ≥ 970 psig to perform SR 3.5.3.4 and > 150 psig to perform SR 3.5.3.5. | I
Adequate steam flow is represented by at least one turbine bypass valve open, or main turbine generator load is greater than 100 MWe. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is

(continued)

BASES

SURVEILLANCE

SR 3.5.3.6 (continued)

The 24 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by Note 1 that says the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The time allowed for this test after required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. Adequate reactor pressure must be available to perform this test. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test. Adequate reactor steam pressure is > 150 psig. Adequate steam flow is represented by at least one turbine bypass valve open. Reactor startup is allowed prior to performing this test because the reactor pressure is low and the time allowed to satisfactorily perform the test is short.

This SR is modified by Note 2 that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Section 4.7.
3. 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

REFERENCES (continued)

4. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.

I

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION I

Source of Change	Summary of Change	Affected Pages
RAI 3.6.1.1-1	The changes agreed to by JAFNPP during a conversation with the NRC concerning RAI 3.6.1.1-1 have been made. The changes concern moving portions of the CTS definition of Primary Containment Integrity to the ITS Bases.	**Specification 3.6.1.1** CTS markup p 1 of 8 and 2 of 8 DOC LA1 (DOCs p 3 of 6)
Technical change related to BSI-F4	The ITS includes an exemption to performing the leakage rate testing of the Primary Containment Leakage Rate Testing Program. Specifically, ITS SR 3.6.1.1.1 excludes the LPCI and CS System injection line air operated testable check valves. The SR should not exempt these valves, since the valves are not exempt from testing. The actual concern is to ensure that the failure of the valves to meet individual leak rate testing criteria does not automatically mean SR 3.6.1.1.1 is failed. The Bases for SR 3.6.1.1.1 adequately covers this concern, similar to the manner in which the MSIV leakage limit is handled in the NUREG SR 3.6.1.1.1 Bases.	**Specification 3.6.1.1** NUREG ITS markup p 3.6-2 JFD CLB3 (deleted) (JFDs p 1 of 1) Retyped ITS p 3.6-2
Typographical errors	Minor typographical errors in the NUREG Bases markup and the retyped ITS have been corrected. (The word "opening" in the NUREG Bases markup for SR 3.6.1.7.3 should not be deleted and the term "guage/minute" has been changed to "gauge per minute" in the retyped SR 3.6.1.1.2)	**Specification 3.6.1.1** NUREG Bases markup p B 3.6-53 Retyped ITS p 3.6-2
Technical change	When the 20 and 24 inch primary containment purge and vent valves are open, current analysis not only require the full flow line to the SGT System to be closed, but also requires one or more SGT System reactor building suction valves to be open. The combination of the two is needed for protection of the SGT filter trains from overpressure concerns. Therefore, this second restriction has been added to the appropriate SR (the first restriction is already in the appropriate SR).	**Specification 3.6.1.3** CTS markup p 9 of 10 DOC M7 (DOCs p 7 of 14) NUREG ITS markup p 3.6-14 JFD CLB4 (JFDs p 1 of 5) NUREG Bases markup p B 3.6-25 Bases JFD CLB4 (Bases JFDs p 1 of 6) Retyped ITS p 3.6-12 Retyped ITS Bases p B 3.6-23

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION I

Source of Change	Summary of Change	Affected Pages
RAI 3.6.1.3-1	The changes agreed to by JAFNPP during conversations concerning RAI 3.6.1.3-1 have been made. Specifically, the leakage limits for the LPCI and CS System vessel injection penetration air operated testable check valves have been moved to the Bases. This is consistent with TSTF-52, Revision 3.	**Specification 3.6.1.3** CTS markup p 7 of 10 DOC LA1 (DOCs p 7 of 14) NUREG ITS markup p 3.6-18 (Insert page 3.6-18 deleted) JFDs CLB11 and X9 (deleted) (JFDs p 2 of 5 and 5 of 5) NUREG Bases markup p B 3.6-31 and Insert page B 3.6-31 Bases JFDs CLB11 and X13 (deleted) (Bases JFDs p 2 of 6 and 6 of 6) Retyped ITS p 3.6-14 Retyped ITS Bases p B 3.6-28
Editorial change	The CTS included the MSIV leakage limit in both the Containment Section and in the Primary Containment Leakage Rate Testing Program. In the JAFNPP ITS submittal, the limit was only maintained in the Program. However, the NUREG includes the limit in the MSIV Specification, not the Program. Therefore, for consistency with the NUREG, the MSIV leakage limit has been added to the MSIV leakage limit SR and deleted from the Primary Containment Leakage Rate Testing Program. (Note - This item is also described in the Section 5.0 Summary)	**Specification 3.6.1.3** CTS markup p 1 of 10 through 10 of 10 DOC A6 (DOCs p 2 of 14) NUREG ITS markup p 3.6-17 JFD CLB10 (JFDs p 2 of 5) NUREG Bases markup B 3.6-31 Retyped ITS p 3.6-14 Retyped ITS Bases B 3.6-27 **Specification 5.5** CTS markup p 8 of 22 NUREG ITS markup p Insert page 5.0-10-2 Retyped ITS p 5.0-11
Typographical error	Minor typographical error in the Discussion of Changes has been corrected. (DOC A2, "ITS SR 3.6.1.7" changed to "ITS SR 3.6.1.7.1".)	**Specification 3.6.1.7** DOC A2 (DOCs p 1 of 6)
RAI 3.6.1.7-2	The change agreed to by JAFNPP during a conversation with the NRC concerning RAI 3.6.1.7-2 has been made. Specifically, the CTS markup and DOC have been corrected to reflect the 2 hour Completion Time.	**Specification 3.6.1.7** CTS markup p 2 of 3 DOC M3 (DOCs p 2 of 6)
Amendment 271	This amendment affects the RHRSW System (CTS 3.5.B.3 footnote *), which is on a CTS mark-up page used by this Specification. However, the Amendment does not affect this Specification.	**Specification 3.6.1.9** CTS markup p 2 of 2

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION I

Source of Change	Summary of Change	Affected Pages
RAI 3.6.1.9-6	The change agreed to by JAFNPP during a conversation with the NRC concerning RAI 3.6.1.9-6 has been made. Specifically, the last two sentences of the LCO section of the Bases (which concerns the alignment and operation of the System in the decay heat removal mode) have been deleted.	<u>Specification 3.6.1.9</u> NUREG Bases markup p Insert page B 3.6-57c and Insert page B 3.6-57d Retyped ITS Bases p B 3.6-53
RAI 3.6.2.1-2	The change agreed to by JAFNPP during a conversation with the NRC concerning RAI 3.6.2.1-2 has been made. Specifically, DOC A5 has been added back into the submittal to properly describe the change to CTS 3.7.A.1.c.(1). Also, DOC M4 has been modified to be consistent with a similar DOC in ITS 3.6.2.2.	<u>Specification 3.6.2.1</u> DOCs A5 and M4 (DOCs p 1 of 6, 2 of 6, and 3 of 6)
Amendment 271	This amendment affects the RHRSW System (CTS 3.5.B.3 footnote *), which is on a CTS mark-up page used by this Specification. However, the Amendment does not affect this Specification.	<u>Specification 3.6.2.3</u> CTS markup p 2 of 2
RAI 3.6.2.3-5	The change agreed to by JAFNPP during a conversation with the NRC concerning RAI 3.6.2.3-5 has been made. Specifically, the last two sentences of the LCO section of the Bases (which concerns the alignment and operation of the System in the decay heat removal mode) have been deleted.	<u>Specification 3.6.2.3</u> NUREG Bases markup p B 3.6-68 and Insert page B 3.6-68 (deleted) Retyped ITS Bases p B 3.6-68
Typographical error	Minor typographical error in the Discussion of Changes has been corrected. (DOC A2, "CTS 4.7.A.7" changed to "CTS 4.7.A.7.a.")	<u>Specification 3.6.2.4</u> DOC A2 (DOCs p 1 of 2)
Typographical error	Minor typographical error in the Discussion of Changes has been corrected. (DOC M6, "CTS RETS Table 4.2-2" changed to "CTS RETS Table 3.10-2.")	<u>Specification 3.6.4.2</u> DOC M6 (DOCs p 5 of 10)
Typographical errors	Minor typographical errors in the Discussion of Changes have been corrected. (DOC L4, "CTS Tables 4.2-1 Note 7" changed to "CTS Table 4.2-1 Note 7 and RETS Table 3.10-2 Note f" and the parenthetical "(Item 5 of Table 4.2-1)" has been deleted.)	<u>Specification 3.6.4.3</u> DOC L4 (DOCs p 8 of 8)

See ITS 1.0

Specification 3.6.1.1

1.0 (cont'd)

A1

1. **Refuel Mode** - The reactor is in the refuel mode when the Mode Switch is in the Refuel Mode position. When the Mode Switch is in the Refuel position, the refueling interlocks are in service.

2. **Run Mode** - In this mode the reactor system pressure is at or above 850 psig and the Reactor Protection System is energized with APRM protection (excluding the 15 percent high flux trip) and the RBM interlocks in service.

3. **Shutdown Mode** - The reactor is in the shutdown mode when the Reactor Mode Switch is in the Shutdown Mode position.

   a. Hot shutdown means conditions as above with reactor coolant temperature >212°F.

   b. Cold shutdown means conditions as above with reactor coolant temperature ≤212°F. and the reactor vessel vented.

4. **Startup/Hot Standby** - In this mode the low pressure main steam line isolation valve closure trip is bypassed, the Reactor Protection System is energized with APRM (15 percent) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

J. **Operable** - A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

K. **Operating** - Operating means that a system or component is performing its intended functions in its required manner.

L. **Operating Cycle** - Interval between the end of one refueling outage and the end of the subsequent refueling outage.

LCO 3.6.1.1

M. **Primary Containment Integrity** - OPERABLE Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

LA1

1. All manual containment isolation valves on lines connected to the Reactor Coolant System or containment which are not required to be open during plant accident conditions are closed. These valves may be

See ITS 3.6.1.3

I

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REVISION ~~F~~ I

Specification 3.6.1.1 A1

See ITS: 3.6.1.3

opened to perform necessary operational activities.

See ITS: 3.6.1.2

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

see ITS; 3.6.1.3

4. All blind flanges and manways are closed. LA1

N. **Rated Power** - Rated power refers to operation at a reactor power of 2,536 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power (Reference 1).

O. **Reactor Power Operation** - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. **Reactor Vessel Pressure** - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. **Refueling Outage** - Refueling outage is the period of time between the shutdown of the unit prior to refueling and the startup of the Plant subsequent to that refueling.

R. **Safety Limits** - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational

See ITS: Chapter 1.0

see ITS: Chapter 1.0

deficiency subject to regulatory review.

S. **Secondary Containment Integrity** - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

See ITS; 3.6.4.1

I

1. At least one door in each access opening is closed.

2. The Standby Gas Treatment System is operable. See ITS; 3.6.4.3

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

See ITS: 3.6.4.2

T. **Surveillance Frequency Notations / Intervals**

The surveillance frequency notations / intervals used in these specifications are defined as follows:

Notations	Intervals	Frequency
D	Daily	At least once per 24 hours
W	Weekly	At least once per 7 days
M	Monthly	At least once per 31 days
Q	Quarterly or every 3 months	At least once per 92 days
SA	Semiannually or every 6 months	At least once per 184 days
A	Annually or Yearly	At least once per 366 days
18M	18 Months	At least once per 18 months (550 days)
R	Operating Cycle	At least once per 24 months (731 days)
S/U		Prior to each reactor startup
NA		Not applicable

see ITS: Chapter 1.0 Section 5.5

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TECHNICAL CHANGES - MORE RESTRICTIVE

M4 The CTS 4.7.A.5.d requirement that the drywell to suppression chamber leak rate test be conducted at 1 psid is being changed to a differential pressure of ≥ 1 psi. Performing the test at precisely 1 psid is not possible and actual test performance is conducted at slightly higher differential pressure to ensure test differential pressure does not decrease to less than 1 psi. The higher test differential pressure increases leakage resulting in conservative (more restrictive) test results. Therefore, this change is considered to be more restrictive but necessary to allow test performance in strict compliance with the SR and in a conservative manner.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details of the CTS 1.0.M definition of Primary Containment integrity concerning the manual isolation valves (CTS 1.0.M.1), automatic isolation valves (CTS 1.0.M.3), that the drywell and pressure suppression chamber are intact (CTS 1.0.M) and the requirement that manways are closed (CTS 1.0.M.4) are proposed to be relocated to the Bases. The requirement in ITS LCO 3.6.1.1 that the Primary Containment shall be OPERABLE (see A2) and the definition of Operability is sufficient to ensure the requirements are met. The ITS 3.6.1.1 LCO Bases states that compliance with this LCO will ensure a primary containment configuration, including hatches (manways), that is structurally sound and that will limit leakage to those leakage rates assumed in the analysis. This requirement ensures the existing requirements are retained. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

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LA2 Not used.

LA3 The details in CTS 3.7.A.5.e that the drywell to suppression chamber leakage rate limit of ≤ 71 scfm shall be monitored via the suppression chamber 10 minute pressure transient is proposed to be relocated to the Bases. The requirement in ITS SR 3.6.1.1.2 to verify the suppression chamber pressure increase is ≤ 0.25 in. water gauge/minute for a 10 minute period is sufficient to ensure the requirement is met. The details in the Bases of SR 3.6.1.1.2 will ensure the test is performed consistent with the current requirements. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - MORE RESTRICTIVE (GENERIC)

LA4 The requirement of CTS 4.7.A.3 (Continuous Leak Rate Monitoring) that when the primary containment is inerted, it shall be continuously monitored for gross leakage by review of the inerting system makeup requirements is proposed to be relocated to the UFSAR. The requirements in ITS LCO 3.6.1.1, that the Primary Containment shall be Operable, the requirement in ITS LCO 3.6.1.2, that two primary containment air locks shall be Operable, the definition of Operability, and the requirements in SR 3.6.1.1.1 and SR 3.6.1.2.1 to perform required visual examinations and leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program are sufficient to ensure all Primary Containment Leakage limits are met. As such, this Surveillance is not required to be in the ITS to provide adequate protection of public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.7.A.2 does not provide any time to restore the primary containment to Operable status if it is found to be inoperable. Entry into CTS 3.7.A.8 is required and the plant is required to be in cold shutdown within 24 hours. ITS 3.6.1.1 ACTION A has been added to allow 1 hour to restore primary containment to OPERABLE status. ITS 3.6.1.1 ACTION A provides 1 hour to restore the primary containment to OPERABLE before proceeding to ACTION B and the subsequent MODE 3 in 12 hours (M1) and MODE 4 in 36 hours (L2). The additional one hour allowed to restore primary containment provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. Additionally, the one hour period ensures the probability of an accident (requiring containment OPERABILITY) occurring during periods where primary containment is inoperable is maintained at a minimum.

L2 CTS 3.7.A.8 requires the reactor to be in the cold condition (MODE 4) within 24 hours if the requirements of CTS 3.7.A.2 (primary containment integrity) cannot be met. ITS 3.6.1.1 Required Action B.2 requires the plant to be in MODE 4 in 36 hours if the Required Action and associated Completion Time (primary containment restored to OPERABLE status in 1 hour) of ITS 3.6.1.1 ACTION A (L1) is not met. However, ITS 3.6.1.1 Required Action B.1 requires the plant to be in MODE 3 in 12 hours (M1). This change is less restrictive because it extends the time for the plant to be in MODE 4 from 24 hours to 37 hours (1 hour from Required Action A.1 (L1) and 36 hours from Required Action B.1). The allowed Completion Times in Required Actions B.1 and B.2 are reasonable, based on operating experience, to reach the required plant conditions from

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[4.7.A.2.a] SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Primary Containment Leakage Rate Testing Program. ~~The leakage rate acceptance criterion is ≤ 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L_a for the Type B and Type C tests, and < 0.75 L_a for the Type A test.~~	~~-----NOTE----- SR 3.0.2 is not applicable ----------------~~ In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ TA1
[4.7.A.5.d] SR 3.6.1.1.2 [M2]	Verify ~~drywell to~~ suppression chamber ~~differential~~ pressure ~~does not decrease at a rate ≥~~ increase is ≤ [0.25] inch water gauge per minute tested over a [10] minute period ~~at an initial~~ with a drywell to suppression chamber differential pressure of [1] psi~~d~~. Z	[~~18~~ 24 months] CLB2 AND -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ---------------- PA1 [~~9~~ 12 months] X1

Revision I

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.1.1 - PRIMARY CONTAINMENT

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 Not used.

CLB2 The brackets have been removed on the ITS SR 3.6.1.1.2 Frequency and changed from 18 months to 24 months as currently required by CTS 4.7.A.5.d.

| I

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 ITS SR 3.6.1.1.2 has been revised to reflect UFSAR Section 5.2.4.4.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 52, Revision 3, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 ITS SR 3.6.1.1.2, second Frequency to verify drywell to suppression chamber differential pressure leakage rate, in accordance with the Note condition, when two consecutive tests fail and continues until two consecutive tests pass, has been included. The Frequency of 12 months is half of the normal Frequency of ITS SR 3.6.1.1.2 (CTS 4.7.A.5.d) which is consistent with the philosophy utilized in NUREG-1433.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program	I
SR 3.6.1.1.2	Verify suppression chamber pressure increase is $\leq$ 0.25 in. water gauge per minute over a 10 minute period with a drywell to suppression chamber differential pressure of $\geq$ 1 psi.	24 months AND -----NOTE------ Only required after two consecutive tests fail and continues until two consecutive tests pass -------------- 12 months	I

[See 3.6.1.3.7]

A9

See ITS: 3.3.6.1

JAFNPP

Specification 3.6.1.3

A1

TABLE 4.2-1 (Cont'd)

PRIMARY CONTAINMENT ISOLATION SYSTEM INSTRUMENTATION TEST AND CALIBRATION REQUIREMENTS

Actual or Simulated automatic activation

see ITS: 3.3.5.1

Logic System Functional Test (Notes 7 & 9)		Frequency
1)	Main Steam Line Isolation Valves Main Steam Line Drain Valves Reactor Water Sample Valves	R
2)	RHR - Isolation Valve Control Shutdown Cooling Valves	R
3)	Reactor Water Cleanup Isolation	R
4)	Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves	R
5)	Standby Gas Treatment System Reactor Building Isolation	R
6)	HPCI Subsystem Auto Isolation	R
7)	RCIC Subsystem Auto Isolation	R

See ITS: 3.3.6.1

see ITS: 3.3.6.2

see ITS: 3.3.6.1

NOTE: See notes following Table 4.2-5.

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Amendment No. ~~14, 43, 53, 89, 106, 120, 160, 181, 190, 227~~, 248

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Specification 3.6.1.3

(A1)

NOTES FOR TABLES 4.2-1 THROUGH 4.2-5

See ITS: 3.4.5

See ITS: 3.3.2.1

1. Initially once every month until acceptance failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instruments operate in a environment similar to that of JAFNPP.

See ITS: 3.3.2.1 3.3.5.1 3.3.6.1 3.3.5.2

2. Functional tests are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed within seven (7) days prior to each startup.

3. Calibrations are not required when these instruments are not required to be operable or are tripped. Calibration tests shall be performed within seven (7) days prior to each startup or prior to a pre-planned shutdown.

4. Instrument checks are not required when these instruments are not required to be operable or are tripped.

See ITS 3.3.2.1 3.3.5.1 3.3.6.1 3.3.5.2 3.3.3.2

5. This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

6. These instrument channels will be calibrated using simulated electrical signals once every three months.

See ITS: 3.3.2.1

[SR 3.6.1.3.7]

7. "actual" or Simulated automatic actuation shall be performed once per 24 months.

L1

See ITS 3.3.6.1

8. Reactor low water level, and high drywell pressure are not included on Table 4.2-1 since they are listed on Table 4.1-2.

See ITS 3.3.5.1 3.3.6.1

9. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

~~10. (Deleted).~~

See ITS: 3.3.6.1 3.3.7.2

11. Perform a calibration once per 24 months using a radiation source. Perform an instrument channel alignment once every 3 months using a current source.

~~12. (Deleted)~~

~~13. (Deleted)~~

~~14. (Deleted)~~

See ITS 3.3.5.1 3.3.6.1

15. Sensor calibration once per 24 months. Master/slave trip unit calibration once per 6 months.

16. The quarterly calibration of the temperature sensor consists of comparing the active temperature signal with a redundant temperature signal.

See ITS: 3.3.6.1

Specification 3.6.1.3

A1

See ITS: 1.0

1.0 (cont'd)

I. 1. **Refuel Mode** - The reactor is in the refuel mode when the Mode Switch is in the Refuel Mode position. When the Mode Switch is in the Refuel position, the refueling interlocks are in service.

2. **Run Mode** - In this mode the reactor system pressure is at or above 850 psig and the Reactor Protection System is energized with APRM protection (excluding the 15 percent high flux trip) and the RBM interlocks in service.

3. **Shutdown Mode** - The reactor is in the shutdown mode when the Reactor Mode Switch is in the Shutdown Mode position.

   a. Hot shutdown means conditions as above with reactor coolant temperature >212°F.

   b. Cold shutdown means conditions as above with reactor coolant temperature ≤212°F. and the reactor vessel vented.

4. **Startup/Hot Standby** - In this mode the low pressure main steam line isolation valve closure trip is bypassed, the Reactor Protection System is energized with APRM (15 percent) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

J. **Operable** - A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

K. **Operating** - Operating means that a system or component is performing its intended functions in its required manner.

L. **Operating Cycle** - Interval between the end of one refueling outage and the end of the subsequent refueling outage.

[3.6.1.3]

M. **Primary Containment Integrity** - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

PCIVs — All

See ITS: 3.6.1.1

1. All manual containment isolation valves on lines connected to the Reactor Coolant System or containment which are not required to be open during plant accident conditions are closed. These valves may be

M4 — [SR 3.6.1.3.2] [SR 3.6.1.3.3]

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2
Note 2 to SR 3.6.1.3.3

add Surveillance Frequency — M4

Amendment No. ~~95~~, ~~122~~ , 134

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Specification 3.6.1.3 (A1)

1.0 (cont'd)

NOTE 1 to ACTIONS
Note 2 to SR 3.6.1.3.2
Note 2 to SR 3.6.1.3.3

opened to perform necessary operational activities.

See ITS: 3.6.1.2

2. At least one door in each airlock is closed and sealed.

[LCO 3.6.1.3] 3. All automatic containment isolation valves are operable or de-activated in the isolated position.

LA5

See ITS 3.6.1.1

[SR 3.6.1.3.2]
[SR 3.6.1.3.3]
M4

4. All blind flanges and manways are closed.

M4
add Surveillance Frequency

N. **Rated Power** - Rated power refers to operation at a reactor power of 2,536 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power (Reference 1).

O. **Reactor Power Operation** - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. **Reactor Vessel Pressure** - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. **Refueling Outage** - Refueling outage is the period of time between the shutdown of the unit prior to refueling and the startup of the Plant subsequent to that refueling.

R. **Safety Limits** - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational

see ITS Chapter 1.0

See ITS: Chapter 1.0

deficiency subject to regulatory review.

S. **Secondary Containment Integrity** - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed. See ITS: 3.6.4.1

2. The Standby Gas Treatment System is operable. See ITS: 3.6.4.3

3. All automatic ventilation system isolation valves are operable or secured in the isolated position. See ITS: 3.6.4.2

T. **Surveillance Frequency Notations / Intervals**

The surveillance frequency notations / intervals used in these specifications are defined as follows:

Notations	Intervals	Frequency
D	Daily	At least once per 24 hours
W	Weekly	At least once per 7 days
M	Monthly	At least once per 31 days
Q	Quarterly or every 3 months	At least once per 92 days
SA	Semiannually or every 6 months	At least once per 184 days
A	Annually or Yearly	At least once per 366 days
18M	18 Months	At least once per 18 months (550 days)
R	Operating Cycle	At least once per 24 months (731 days)
S/U		Prior to each reactor startup
NA		Not applicable

See ITS: Chapter 1.0
Section 5.5

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Specification 3.6.1.3 (A1)

(A8) except reactor building-to-suppression chamber vacuum breakers

3.7 (cont'd)

add second Applicability (M1) MODES 1, 2 and 3

[3.6.1.3] D. Primary Containment Isolation Valves

[Applicability] 1. Whenever primary containment integrity is required per 3.7.A.2, [LCO 3.6.1.3] containment isolation valves ~~and all instrument line excess flow check valves~~ shall be operable, ~~except as specified in 3.7.D.2.~~ (A8) The containment vent and purge valves shall be limited to opening angles less than or equal to that specified below:

Valve Number	Maximum Opening Angle
27AOV-111	40°
27AOV-112	40°
27AOV-113	40°
27AOV-114	50°
27AOV-115	50°
27AOV-116	50°
27AOV-117	50°
27AOV-118	50°

(LA6)

add ACTIONS Note 2, 3 and 4 (A2)

4.7 (cont'd)

see JTS: 3.6.4.1

c. Secondary containment capability to maintain a 1/4 in. of water vacuum under calm wind conditions with a filter train flow rate of not more than 6,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

D. [3.6.1.3] Primary Containment Isolation Valves

(A1)

1. The primary containment isolation valves surveillance shall be performed as follows:

(A9) 24 months (A9)

	Item	Frequency [SR 3.6.1.3.5]
a. [3.6.1.3.7]	The operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic (actual or (L1)) initiation and for closure time. [SR 3.6.1.3.5]	In accordance with the Inservice Testing Program
b. [SR 3.6.1.3.8]	Instrument line excess flow check valves shall be tested for proper operation.	In accordance with the Inservice Testing Program
[3.6.1.3.5] c.	All normally open power-operated isolation valves (except for the main steam isolation valves) shall be fully closed and reopened.	In accordance with the Inservice Testing Program

to actuate to the isolation position on a simulated instrument line break (M2)

(A10)

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Specification 3.6.1.3 (A1)

3.7 (cont'd)

add Proposed ACTION B (L3)

add Proposed ACTION C (L4)

add Required Action A.2 and Required Action C.2 Notes (L11)

add Note to Condition A (A3)

[ACTION A] 2. With one or more of the containment isolation valves inoperable, ~~maintain at least one isolation valve operable in each affected penetration that is open and:~~

~~a. Restore the inoperable valve(s) to operable status within 4 hours; or~~ (A4)

[Required Action A.1]

b. Isolate each affected penetration within ~~4~~ hours by use of at least one deactivated automatic valve secured in the closed position. Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control; or

8 hours for main steam lines (L5)

c. Isolate each affected penetration within ~~4~~ hours by use of at least one closed manual valve or a blind flange. ^ or check valve with flow through the valve secured or closed and deactivated automatic valve. (L6)

3. If Specifications 3.7.D.1 or 3.7.D.2 cannot be met the ^ reactor shall be in the cold condition within ~~24~~ 36 hrs. (L7)

(M5) MODE 3 in 12 hours

add ACTION G (M1)

4.7 (cont'd)

~~Item~~	Surveillance	Frequency
[SR 3.6.1.3.6] d.	Fast close each main steam isolation valve, and verify closure time. ^ add limits (M3)	In accordance with the Inservice Testing Program

~~2.~~ Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be ~~recorded~~ ~~daily~~. (A5) (L8)

Required Action A.2 and C.2

add proposed SRs:
SR 3.6.1.3.2
SR 3.6.1.3.3
SR 3.6.1.3.4
SR 3.6.1.3.9
(M4)

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Specification 3.6.1.3 (A1)

~~3.7~~ (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

see ITS 3.6.2.1

[LCO 3.6.1.3] 2. ~~Primary containment integrity shall be maintained~~ at all times when the reactor is critical or when the reactor water temperature ~~is above 212°F, and fuel is in the reactor vessel~~, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE (AB)

add second Applicability (M1)

see ITS: 3.10.8

[Applicability MODES 1, 2 and 3] (M1)

add proposed ACTION D for MSIV leakage not within limits (L9)

add proposed ACTION E for any LPCI or CS AOV PCIV leakage not within limits (L10)

~~4.7~~ (Con't'd)

(L13) In accordance with the primary Containment Leakage Rate Testing Program

see ITS: 3.6.1.1

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

[SR 3.6.1.3.10] b. Demonstrate leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

(L12)

[SR 3.6.1.3.11] c. ~~Once per 24 months~~, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system ~~to be less than 11 scfm per valve when pneumatically tested at ≥ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at ≥ 1,035 psig at ambient temperature.~~

within limits. (A1)

| I

| I

Amendment No. ~~16, 234~~, 239

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Revision I

Specification 3.6.1.3

A1

3.7 (Cont'd)

4.7 (Cont'd)

(1) The drywell to torus differential pressure shall be established within 24 hours of exceeding 15% rated thermal power during startup. The differential pressure may be reduced to less than the limit up to 24 hours prior to reducing thermal power to less than 15% of rated before a plant shutdown.

(2) The differential pressure may be decreased to less than 1.7 psid for a maximum of four (4) hours during required operability testing of the HPCI, RCIC, and Suppression Chamber - Drywell Vacuum Breaker System.

(3) If 3.7.A.7.a above cannot be met, restore the differential pressure to within limits within eight hours or reduce thermal power to less than 15% of rated within the next 12 hours.

See ITS: 3.6.2.4

8. If the specifications of 3.7.A.1 through 3.7.A.5 cannot be met the reactor shall be in the cold condition within ~~24~~ 36 hours.

MODE 3 12 hours

M5

L7

[ACTION F]

~~8. Not applicable.~~

Page 8 of 10 (I)

A1

3.7 (cont'd)

b. If in Refuel or Cold Shutdown mode, reactor operation or irradiated fuel handling is permissible only during the succeeding 31 days unless such circuit is sooner made operable, provided that during such 31 days all active components of the other Standby Gas Treatment Circuit shall be operable.

3. If Specifications 3.7.B.1 and 3.7.B.2 are not met, the reactor shall be placed in the cold condition and irradiated fuel handling operations and operations that could reduce the shutdown margin shall be prohibited.

4. Whenever primary containment integrity is required as specified in Section 3.7.A.2. Valve 27MOV-121 shall be used for inerting or deinerting.

[Note to SR 3.6.1.3.1]

4.7 (cont'd)

See ITS 3.6.4.3

3. Intentionally Blank

M7

[SR 3.6.1.3.1 (including Note)]

20 and 24 inch vent and purge valves

4. ~~Valve 27MOV-120~~ shall be verified closed when containment integrity is established, and then once per month.

1 I

Specification 3.6.1.3

A1

6.19 POSTACCIDENT SAMPLING PROGRAM

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,

B) Procedures for sampling and analysis,

C) Provisions for maintenance of sampling and analysis

6.20 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the Primary Containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the exception that Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

1

A. The peak Primary Containment internal pressure for the design basis loss of coolant accident (P_a), is 45 psig.

B. The maximum allowable Primary Containment leakage rate (L_a), at P_a, shall be 1.5% of primary containment air weight per day.

C. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests;

2. Airlock testing acceptance criteria are:

    a. Overall airlock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

    b. For each door seal, leakage rate is ≤ 120 scfd when tested at $\geq P_a$.

See ITS: Section 5.5

[SR 3.6.1.3-10]

3. MSIV leakage rate acceptance criteria is ≤ 11.5 scfh for each MSIV when tested at ≥ 25 psig.

D. The provisions of Specification 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

E. The provisions of Specification 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

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DISCUSSION OF CHANGES
ITS: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES (PCIVs)

ADMINISTRATIVE CHANGES

A3 CTS 3.7.D.2 requirement, to maintain at least one isolation valve operable in each affected penetration that is open, is being deleted. Proposed ITS 3.6.1.3 Condition A Note has been provided to restrict the applicability to penetrations with two PCIVs, where a second valve is available. This Note is consistent with the Notes provided in the new proposed ITS 3.6.1.3 Condition B (L3) for two valves inoperable in a penetration with two PCIVs, and ITS 3.6.1.3 Condition C (L4) for penetrations with only one PCIV. The addition of this Note identifying the applicable configuration, in conjunction with the separate and specific requirements provided in the proposed Conditions, is consistent with the format of NUREG-1433, Revision 1. Since there is no change in any technical requirements, this change is considered administrative.

A4 The requirement in CTS 3.7.D.2.a, to "restore the inoperable valve(s) to operable status within 4 hours," has been deleted since this is always an option. Since the time requirements on the alternative actions (CTS 3.7.D.2.b and 3.7.D.2.c are identical this change is considered administrative.

A5 The requirement to record the results in CTS 4.7.D.2 (ITS 3.6.1.3 Required Actions A.2 and C.2) is proposed to be deleted. This requirement duplicates the requirements of 10 CFR 50 Appendix B, Section XVII (Quality Assurance Records) to maintain records of activities affecting quality, including the results of tests/verifications. Compliance with 10 CFR 50 Appendix B is required by the JAFNPP Operating License. The details of the regulations within the Technical Specifications are repetitious and unnecessary. Therefore, retaining the requirement to perform the associated verifications and eliminating the details from Technical Specifications that are found in 10 CFR 50 Appendix B is considered a presentation preference, which is administrative.

A6 Not Used. I

A7 Not Used.

A8 CTS 3.7.A.2 (3.7.D.1) requirement for primary containment isolation valves (PCIVs) to be Operable, has been revised. Proposed ITS LCO 3.6.1.3 provides an exception for reactor building-to-suppression

TECHNICAL CHANGES - MORE RESTRICTIVE

M7 CTS 4.7.B.4 requirement, that 27MOV-120 (12 inch, full-flow valve) be verified closed when containment integrity is established, and then once per month, is being revised. ITS SR 3.6.1.3.1, requires verification that each 20 and 24 inch primary containment purge and vent valve is closed every 31 days. Since the purge and vent valves are the actual primary containment isolation valves (PCIVs) associated with these penetrations, this change is appropriate. Since CTS 3.7.B.4 allows inerting and de-inerting operations only with valve 27MOV-121 (6 inch, low flow valve) it is understood that the primary containment purge and vent valves must be opened for these operations. Therefore, a Note has been added to proposed SR 3.6.1.3.1 which allows these operations to occur as long as the full-flow line (27MOV-120) is closed and one or more Standby Gas Treatment (SGT) System reactor building suction valves are open. This provides protection of the SGT filter trains from over pressure concerns. This change is considered more restrictive since the primary containment vent and purge valves are required to be closed when these operations are not underway. This is consistent with current practice and in accordance with the UFSAR safety analyses. This assures that the requirements of the LOCA are met and ensures these valves are opened for a valid reason. This change is not considered to result in any reduction to safety.

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TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 Requirements in CTS 4.7.A.2.c concerning the leakage limit and test pressure for LPCI/CS air operated testable check valves are proposed to be relocated to the Bases. The leakage limits and test pressure are not necessary for ensuring the test is performed. The requirements of ITS 3.6.1.3 and SR 3.6.1.3.11 are adequate to ensure the OPERABILITY of these valves and that they are tested properly. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

I

LA2 Not Used.

LA3 Not Used.

LA4 Not Used.

SURVEILLANCE REQUIREMENTS (continued)

CLB1

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.~~2~~ 1	-------------------NOTE~~S~~------------------- ~~1. Only required to be met in MODES 1, 2, and 3.~~ ~~2.~~ Not required to be met when the ~~[18]~~ 20 and 24 inch primary containment vent and purge valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the full flow line to Standby Gas Treatment (SGT) System is closed and one or more SGT System reactor building suction valves are open. -- Verify each ~~[18]~~ 20 and 24 inch primary containment vent and purge valve is closed.	31 days
SR 3.6.1.3.~~3~~ 2	-------------------NOTES------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. -- Verify each primary containment isolation manual valve and blind flange that is located outside primary containment, and not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed.	31 days

PA1
PA3
X3
DB1
CLB4
CLB1
PA3
TA1
I
I

[M7]
[3.7.B.4]
[1.0.M.4]
[1.0.M.1]
[M4]

(continued)

Revision I

SURVEILLANCE REQUIREMENTS (continued)

CLB9

SURVEILLANCE		FREQUENCY
~~SR 3.6.1.3.12~~	~~-------------------NOTES-------------------~~ ~~[1. Only required to be met in MODES 1, 2, and 3.]~~ ~~2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~---~~ ~~Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ [L_a] when pressurized to ≥ [psig].~~	~~------NOTE-----~~ ~~SR 3.0.2 is not applicable~~ ~~---------------~~ ~~In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~
CLB9 10 SR 3.6.1.3.~~13~~	Verify leakage rate through each MSIV is ≤ [11.5] scfh when tested at ≥ [28.8] psig. 25 PA3 the Primary Containment Leakage Rate Testing Program TA4	~~----NOTE-----~~ ~~SR 3.0.2 is not applicable~~ ~~-------------~~ TA4 ~~In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~

[4.7.A.2.b]
[6.20.C.3]

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(continued)

Revision I

SURVEILLANCE REQUIREMENTS (continued)

CLB9 AA1

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.~~14~~ 11	~~NOTE~~ ~~Only required to be met in MODES 1, 2, and 3.~~	
[4.7.A.2.c] CLB11	Verify ~~combined~~ the leakage rate of ~~[1 gpm times the total number of PCIVs] through hydrostatically tested lines that penetrate the primary containment~~ each airoperated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is ~~not exceeded when these isolation valves are tested at ≥ [63.25] psig~~ within limits.	~~NOTE~~ ~~SR 3.0.2 is not applicable~~ TA4 In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Primary Containment Leakage Rate Testing Program
[L13]		
~~SR 3.6.1.3.15~~	~~NOTE~~ ~~Only required to be met in MODES 1, 2, and 3.~~ ~~Verify each [ ] inch primary containment purge valve is blocked to restrict the valve from opening > [50]%.~~	~~[18] months~~

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K7

Revision I

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES (PCIVs)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS 3.6.1.3 has been revised to reflect the current licensing requirements of JAFNPP, that no special vent and purge valve leakage limits, flow path exceptions, or Surveillance Requirements exist in the CTS 3/4.7. The bracketed ISTS 3.6.1.3 Action E, SR 3.6.1.3.1, SR 3.1.6.3.7, and references to purge valve leakage limits are not applicable and have been deleted. Subsequent ACTIONS and Surveillance Requirements have been renumbered as applicable.

CLB2 ISTS 3.6.1.3 ACTION G and ACTION H have been deleted to reflect the current licensing requirement of JAFNPP that no PCIVs are required to be OPERABLE during movement of irradiated fuel or during CORE ALTERATIONS. Subsequent ACTIONS have been renumbered as applicable.

CLB3 Not Used.

CLB4 ITS SR 3.6.1.3.1 Note 2 has been revised to reflect the current licensing requirement of JAFNPP, CTS 3.7.B.4, that for periods when primary containment integrity is required, inerting and de-inerting be performed using the 27MOV-121 (low flow, 6 inch) valve, and the 27MOV-120 (full-flow, 12 inch) valve shall be closed. The Note also includes the design basis requirement that one or more Standby Gas Treatment (SGT) System reactor building suction valves be open to protect the SGT filters from excessive differential pressure in the event of a LOCA during vent and purge operations.

△I

CLB5 ITS SR 3.6.1.3.5 has been revised to reflect current licensing requirements at JAFNPP (CTS 4.7.D.1.a) that the Frequency for verifying isolation time of each automatic PCIV except for MSIVs is in accordance with the Inservice Testing Program.

CLB6 ITS SR 3.6.1.3.6 has been revised to reflect current licensing requirements at JAFNPP (CTS 4.7.D.1.d) that the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds in accordance with UFSAR Table 7.3-1, Primary Containment Isolation Valves, and the Frequency for the Surveillance is in accordance with the Inservice Testing Program.

CLB7 Not Used.

CLB8 ITS SR 3.6.1.3.8 has been revised to reflect current licensing requirements at JAFNPP, CTS 4.7.D.1.b, that the Frequency for verifying each reactor instrument line EFCV actuates to the isolation position on an actual or simulated (M2) isolation instrument line break is in accordance with the Inservice Testing Program. In addition, the requirement to restrict flow to ≤ 1 gph has been deleted since the JAFNPP analysis does not assume a specific leakage through the EFCVs. The leakage will be controlled administratively and will be based on valve design leakage.

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB9 ITS 3.6.1.3 has been revised to reflect the current licensing requirements of JAFNPP, that since no separate secondary containment bypass leakage is considered with respect to the primary containment leakage, no specific leakage rates or Surveillance Requirements exist in the CTS 3/4.7. The bracketed ISTS 3.6.1.3 Action D reference to secondary containment bypass leakage and the bracket SR 3.6.1.3.12 to verify secondary containment bypass leakage path limits are not applicable and have been deleted. Subsequent Surveillance Requirements have been renumbered as applicable.

CLB10 Not Used. | I

CLB11 ITS SR 3.6.1.3.11 (ISTS SR 3.6.1.3.14) has been revised to reflect the current licensing requirement of JAFNPP, CTS 4.7.A.2.c, to determine the leakage rate of hydrostatically tested valves. | I

CLB12 ITS SR 3.6.1.3.7 has been revised to reflect the requirements at JAFNPP that the Frequency for verifying each automatic PCIV actuates to the isolation position on an actual (L1) or simulated isolation signal is 24 months (A9) consistent with CTS Table 4.2-1, Primary Containment Isolation Instrumentation Test and Calibration Requirements.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The words "in MODES 1, 2, and 3" have been deleted from ITS 3.6.1.3 ACTIONS Note 4 since there are no PCIV leakage tests required in MODES other than MODES 1, 2, and 3 for JAFNPP (i.e., there are no PCIVs required to be OPERABLE in MODES other than MODES 1, 2, and 3 that have specific leakage limits). In addition, ITS SR 3.6.1.3.1, Note 1 and SR 3.6.1.3.11 Note 1, have been deleted for the same reason. The subsequent Notes have been renumbered, as applicable.

PA2 Editorial changes have been made to enhance clarity.

<u>DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE</u> (X)

X7 ISTS SR 3.6.1.3.15, to verify each primary containment purge valve is blocked to restrict valve opening, has been deleted. In accordance with the ISTS Bases SR 3.6.1.3.15 Reviewers Note, this Surveillance is not required for valves which have blocking devices permanently installed. JAFNPP blocking devices are permanently installed.

X8 ITS 3.6.1.3 ACTION E has been added to address the condition when leakage rate specified in SR 3.6.1.3.11 (CTS 4.7.A.2.c) is exceeded. The addition of this Action is similar to ACTION D for other leakage limits not within limits. The Completion Time of 72 hours is adequate as described in L10. In addition, the bracketed exceptions of ITS 3.6.1.3 ACTION A and ACTION B, have been revised by replacing the bracketed valve listing with the phrase "for reasons other than Conditions D and E." The change reflects TSTF-207, R5. Subsequent Conditions and Required Actions have also been renumbered to reflect addition of Condition E accordingly.

CLB1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves or the release of radioactive material will exceed limits prior to the closing of the purge valves. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are allowed to be open.

X3 X3

SR 3.6.1.3.~~2~~ 1 CLB1

This SR ensures that the primary containment vent and purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. [The SR is also modified by a Note (Note 1), stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves, or the release of radioactive material will exceed limits prior to the purge valves closing. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are allowed to be open.] The SR is modified by a Note ~~(Note 2)~~ stating that the SR is not required to be met when the vent and purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the full-flow 12 inch line (with valve 27MOV-120) to the SGT is closed and one or more SGT System suction valves are open. This will ensure there is no damage to the filters if a LOCA were to occur with the vent and purge valves open since excessive differential pressure is not expected with the full-flow 12 inch line closed and one or more SGT System suction valves open. The ~~[18]~~ 20 and 24 inch purge valves are capable of closing ~~in the environment following~~ against the dynamic effects of a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other PCIV requirements discussed in SR 3.6.1.3.3.

PA2 PA2 PA2 PA3 DB1 CLB4 I

(continued)

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PA2

CLB9

BASES

SURVEILLANCE REQUIREMENTS (continued)

more

CLB10

25

TA4

SR 3.6.1.3.1~~2~~ 0

CLB10 10 DB7

The analyses in References ~~2 and 6~~ are based on leakage that is ~~less~~ than the specified leakage rate. Leakage through each MSIV must be ≤ [11.5] scfh when tested at ≥ ~~[P_t]~~ ~~(28.8)~~ psig. ~~The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.~~ ~~Note 1 is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.~~ This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is ~~required by 10 CFR 50, Appendix J, as modified by approved exemptions; thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

PA3

I

TA4

In accordance with the Primary Containment Leakage Rate Testing Program

CLB13

resulting radiation dose rate that would result if the Reactor Coolant were released to the reactor building at the specified limit will be small (Ref. 11)

Insert SR 3.6.1.3.11 — CLB11

CLB9

SR 3.6.1.3.1~~0~~ 1

CLB13

Surveillance of ~~hydrostatically tested lines~~ provides assurance that the ~~calculation assumptions of Reference 2 are met~~. The ~~combined~~ leakage rates must be demonstrated in accordance with the leakage rate test Frequency ~~of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions; thus SR 3.0.2 (which allows Frequency extensions) does not apply.~~

I

TA4

required by the Primary Containment Leakage Rate Testing Program

PA3

~~[This SR has been modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3, since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.]~~

X7

~~**SR 3.6.1.3.15**~~

~~[Reviewer's Note: This SR is only required for those plants with purge valves with resilient seals allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices that are not permanently installed on the valves.~~

(continued)

The acceptance criteria for each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is < 10 gpm when hydrostatically tested at ≥ 1035 psig or < 11 scfm when pneumatically tested at ≥ 45 psig, at ambient temperature.

CLB11 TA4

Revision I

CLA 11

INSERT SR 3.6.1.3.11

each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations. I

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JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES (PCIVs)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS 3.6.1.3 has been revised to reflect the current licensing requirements of JAFNPP, that no special vent and purge valve leakage limits, flow path exceptions, or Surveillance Requirements exist in the CTS 3/4.7. The bracketed, ISTS 3.6.1.3 Action E, SR 3.6.1.3.1, SR 3.6.1.3.7, and references to purge valve leakage limits are not applicable and have been deleted. Subsequent Surveillance Requirements have been renumbered as applicable. The Bases has been revised to reflect this change.

CLB2 ISTS 3.6.1.3 ACTION G and ACTION H have been deleted to reflect the current licensing requirement of JAFNPP, that no PCIVs are required to be OPERABLE during movement of irradiated fuel, or CORE ALTERATIONS. Subsequent ACTIONS have been renumbered as applicable.

CLB3 Not Used.

CLB4 ISTS SR 3.6.1.3.2 Note 2 (ITS SR 3.6.1.3.1 Note 1) has been revised to reflect the current licensing requirement of JAFNPP, CTS 3.7.B.4, that for periods when primary containment integrity is required, inerting and de-inerting be performed using the 27MOV-121 (low-flow, 6 inch) valve, and the 27MOV-120 (full-flow, 12 inch) valve shall be closed. The Bases Background and the discussion of SR 3.6.1.3.1 has been revised to reflect this current licensing requirement. The Note also includes the design basis requirement that one or more Standby Gas Treatment (SGT) System reactor building suction valves be open to protect the SGT filters from excessive differential pressure in the event of a LOCA during vent and purge operations.

CLB5 ITS SR 3.6.1.3.5 has been revised to reflect current licensing requirements at JAFNPP, CTS 4.7.D.1.a, that the Frequency for verifying isolation time of each automatic PCIV except for MSIVs is in accordance with the Inservice Testing Program.

CLB6 ITS SR 3.6.1.3.6 has been revised to reflect current licensing requirements at JAFNPP, CTS 4.7.D.1.d, that the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds in accordance with UFSAR, Table 7.3-1, Primary Containment Isolation Valves, and the Frequency for the Surveillance is in accordance with the Inservice Testing Program.

CLB7 Not Used.

CLB8 ITS SR 3.6.1.3.8 has been revised to reflect current licensing requirements at JAFNPP, CTS 4.7.D.1.b, that the Frequency for verifying each reactor instrument line EFCV actuates to the isolation position on a simulated (M2) instrument line break is in accordance with the Inservice Testing Program. In addition, the requirement to restrict flow to ≤ 1 gph has been deleted since the JAFNPP analysis does not assume a specific leakage through the EFCVs.

<u>RETENTION OF EXISTING REQUIREMENT</u> (CLB)

CLB8 (continued)

The leakage will be controlled administratively and will be based on valve design leakage.

CLB9 ITS 3.6.1.3 has been revised to reflect the current licensing requirements of JAFNPP, that since no separate secondary containment bypass leakage is considered with respect to the primary containment leakage, no specific leakage rates or Surveillance Requirements exist in the CTS 3/4.7. The bracketed, ISTS 3.6.1.3 Action D reference to secondary containment bypass leakage, and SR 3.6.1.3.12 to verify secondary containment bypass leakage path limits are not applicable and have been deleted. Subsequent Surveillance Requirements have been renumbered and other places in the Bases have been modified, as applicable.

CLB10 ITS SR 3.6.1.3.10 (ISTS SR 3.6.1.3.13) has been revised to reflect the current licensing requirements of JAFNPP, that the MSIV leakage rate and Frequency are contained in the Primary Containment Leakage Rate Testing Program. In addition, the Note to the ISTS SR 3.6.1.3.13 Frequency has been deleted since SR 3.0.2 does not apply to the Primary Containment Leakage Rate Testing Program as stated in the Bases of SR 3.0.2. Therefore, it is not necessary to include this Note in the ITS. The wording in the Bases has been modified to reflect this change.

CLB11 ITS SR 3.6.1.3.11 (ISTS SR 3.6.1.3.14) has been revised to reflect the current licensing requirement of JAFNPP, CTS 4.7.A.2.c, to determine the leakage rate of each air operated testable check valve associated with the LPCI and CS System vessel injection penetrations. I

CLB12 ITS SR 3.6.1.3.7 has been revised to reflect the requirements at JAFNPP, that the Frequency for verifying each automatic PCIV actuates to the isolation position on an actual (L1) or simulated isolation signal is 24 months (A9) consistent with CTS Table 4.2-1, Primary Containment Isolation Instrumentation Test and Calibration Requirements.

CLB13 ITS SR 3.6.1.3.11 (ISTS SR 3.6.1.3.14) has been revised to reflect the current licensing basis as reflected in the safety evaluation of Amendment 40 (Ref. 8). The appropriate reference has been added.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES (PCIVs)

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X6 ITS SR 3.6.1.3.9 Frequency of 24 months to remove and test the explosive squib from each shear isolation valve of the TIP System has been included (M4). This Frequency is consistent with similar testing which is performed at the refueling cycle frequency.

X7 ISTS SR 3.6.1.3.15, to verify each primary containment purge valve is blocked to restrict valve opening, has been deleted. In accordance with the ISTS Bases SR 3.6.1.3.15 Reviewers Note, this Surveillance is not required for valves which have blocking devices permanently installed. JAFNPP blocking devices are permanently installed.

X8 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X9 This change to ITS 3.6.1.3 Bases A.1 and A.2 was approved to be made in NUREG-1433, Revision 1 per change package BWR-15, C.5, but apparently was not made.

X10 Not used.

X11 ITS 3.6.1.3 has been revised to include reference to the Technical Requirements Manual (TRM) and the Inservice Testing (IST) Program. The TRM will include the PCIV listing while the Inservice Testing Program will include the valve stroke times.

X12 ITS 3.6.1.3 ACTION E has been added to address the condition when the leakage rate specified in SR 3.6.1.3.11 (CTS 4.7.A.2.c) is exceeded for LPCI or CS System testable check valves. The addition of this Action is similar to ACTION D for other leakage limits not within limits (i.e., MSIVs). The Completion Time of 72 hours is adequate as described in L10. The Bases have been revised to reflect this added Condition including modifications to the description for Required Actions A.1 and A.2, Required Actions B.1 and B.2, and Required Actions C.1 and C.2.

I

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	-----NOTE----- Not required to be met when the 20 and 24 inch primary containment vent and purge valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the full-flow line to Standby Gas Treatment (SGT) System is closed and one or more SGT System reactor building suction valves are open. ---------- Verify each 20 and 24 inch primary containment vent and purge valve is closed.	31 days
SR 3.6.1.3.2	-----NOTES----- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. ---------- Verify each primary containment isolation manual valve and blind flange that is located outside primary containment and not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on a simulated instrument line break.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify the leakage rate of each air operated testable check valve associated with the LPCI and CS System vessel injection penetrations is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

ACTIONS (continued)

G.1 and G.2

If any Required Action and associated Completion Time cannot be met for PCIV(s) required to OPERABLE during MODE 4 or 5, the plant must be placed in a condition in which the LCO does not apply. Action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended and valve(s) are restored to OPERABLE status. If suspending an OPDRV would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valve(s) to OPERABLE status. This allows RHR shutdown cooling to remain in service while actions are being taken to restore the valve.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR ensures that the primary containment vent and purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. The SR is modified by a Note stating that the SR is not required to be met when the vent and purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided that full-flow line (with valve 27MOV-120) to the SGT System is closed and one or more SGT System reactor building suction valves are open. This will ensure there is no damage to the filters if a LOCA were to occur with the vent and purge valves open since excessive differential pressure is not expected with the full-flow line closed and one or more SGT System suction valves open. The 20 and 24 inch vent and purge valves are capable of closing against the dynamic effects of a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other PCIV requirements discussed in SR 3.6.1.3.2.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.2

This SR ensures that each primary containment isolation manual valve and blind flange that is located outside primary containment and not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits.

This SR does not require any testing or valve manipulation. Rather, it involves verification that those PCIVs outside primary containment, and capable of being mispositioned, are in the correct position. Since verification of valve position for PCIVs outside primary containment is relatively easy, the 31 day Frequency was chosen to provide added assurance that the PCIVs are in the correct positions.

Two Notes have been added to this SR. The first Note allows valves, blind flanges or equivalent isolation methods located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since the primary containment is inerted and access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these PCIVs, once they have been verified to be in the proper position, is low. A second Note has been included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

SR 3.6.1.3.3

This SR ensures that each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed or otherwise

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve actuates to the isolation position on a simulated instrument line break. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event evaluated in Reference 9. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in-place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 24 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in Reference 8 are based on leakage that is more than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ 25 psig. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program. I

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

Surveillance of each air operated testable check valve associated with the LPCI and CS System vessel injection penetrations provides assurance that the resulting radiation dose that would result if the reactor coolant were released to the reactor building at the specified limit will be small (Ref. 11). The acceptance criteria for each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is < 10 gpm when hydrostatically tested at ≥ 1035 psig or < 11 scfm when pneumatically tested at ≥ 45 psig, at ambient temperature. The leakage rates must be demonstrated in accordance with the leakage rate test Frequency required by the Primary Containment Leakage Rate Testing Program.

I

REFERENCES

1. UFSAR, Section 14.6.
2. UFSAR, Section 6.5.3.2.
3. UFSAR, Section 14.5.2.3.
4. UFSAR, Section 7.3.3.1
5. UFSAR, Table 7.3-1
6. 10 CFR 50.36(c)(2)(ii)
7. Technical Requirements Manual.
8. UFSAR, Section 16.3.2.5.
9. UFSAR, Section 5.2.3.5.
10. UFSAR, Section 14.8.2.1.1.
11. NRC Letter to NYPA, November 9, 1978 NRC Safety Evaluation Supporting Amendment 40 to the Facility Operating License No. DPR-59.

Specification 3.6.1.7

A1

see ITS: 3.6.1.1

3.7 (cont'd)

e. Leakage between the drywell and suppression chamber shall not exceed a rate of 71 scfm as monitored via the suppression chamber 10 min pressure transient of 0.25 in. water/min.

[SR 3.6.1.7.3]

f. The self actuated vacuum breakers shall open when subjected to a force equivalent to 0.5 psid acting on the valve disc.

[ACTION B] M3

M3

g. From and after the date that one of the pressure suppression chamber/drywell vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding ~~seven days~~ unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

12 hours

M3

[ACTION A] 72 hours M2

[for OPENING]

4.7 (cont'd)

~~e. Not applicable~~

~~f. Not applicable~~

[SR 3.6.1.7.3]

g. Once per 24 months, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.5.f and each vacuum breaker shall be inspected and verified to meet design requirements.

LA1

I

Page 2 of 3

Revision I.

DISCUSSION OF CHANGES
ITS: 3.6.1.7 - SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted that do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.7.A.5.a and ITS 3.6.1.7, Suppression Chamber-to-Drywell Vacuum Breakers, require that all of the vacuum breakers be closed. However, ITS SR 3.6.1.7.1 Note 2 makes the exception "except when performing their intended function." This is an explicit recognition that the automatic cycling of the vacuum breakers does not violate the intent of the LCO and is considered an administrative change. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new Surveillance Requirement has been added to CTS 4.7.A.5. ITS SR 3.6.1.7.1 will require the verification that each suppression chamber-to-drywell vacuum breaker is closed every 14 days. The addition of a new Surveillance Requirement constitutes a more restrictive change necessary to ensure the vacuum breakers are in the correct position and the design bases analyses can be met.

M2 CTS 3.7.A.5.c provides an allowance that one drywell suppression chamber vacuum breaker may be inoperable for opening with no specific limitation on the Completion Time. However, the limitation on the Completion Time is provided in CTS 3.7.A.5.g. The vacuum breaker must be restored within 7 days. ITS 3.6.1.7 ACTION A will allow only 72 hours to restore the vacuum breaker to OPERABLE status. This time is permitted since four vacuum breakers can perform the required safety function however the overall system reliability is reduced. Therefore, the 72 hour limit imposed is more restrictive but is acceptable due to the low probability of an event during this time period requiring the remaining vacuum breaker to function. This change is consistent with NUREG-1433, Revision 1.

DISCUSSION OF CHANGES
ITS: 3.6.1.7 - SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS

TECHNICAL CHANGES - MORE RESTRICTIVE

M3 CTS 3.7.A.5.g imposes limitations if one pressure suppression chamber/drywell vacuum breaker is made or found to be inoperable for any reason. If a vacuum breaker is inoperable the valve must be locked closed and operation is allowed for seven days. This action has been divided into two separate conditions. As discussed in comment M2 a Completion Time of 72 hours is given if a valve is found to be inoperable for opening (ITS 3.6.1.7 ACTION A). In CTS 3.7.A.5.g there is a requirement to "lock close" the inoperable vacuum breaker and operation is permissible for seven days, however there is no specific time requirement to close the valve. ITS ACTION B allows 2 hours to close an opened vacuum breaker to reduce the probability of an event that could pressurize primary containment and to allow sufficient time for vacuum breaker to be leak tested. The requirement to "lock" close the valve has been deleted since if the Completion Time is met the valve is assumed to be OPERABLE for opening and therefore the valve must not be locked. This reduction in Completion Time constitutes a more restrictive change necessary to ensure the vacuum breaker is closed. The time provided is necessary to perform the drywell to suppression chamber bypass leakage test of SR 3.6.1.1.2. This test ensures that each suppression chamber-to-drywell vacuum breakers are closed. The suppression chamber-to-drywell vacuum breaker instrumentation may be inoperable or undergoing maintenance and therefore proper suppression chamber-to-drywell vacuum breaker position indication may not be available at the time of the performance of SR 3.6.1.7.1. Local verification is possible, however this type of verification may not be convenient due to ALARA concerns. If excessive leakage existed, the suppression chamber and drywell pressure instrumentation would have indicated whether the primary containment was inoperable. ITS SR 3.0.1 will require all SRs to be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Therefore, as a result of ITS SR 3.6.1.1.2, the associated ACTIONS of ITS 3.6.1.1 (1 hours for primary containment inoperability), and SR 3.0.1, the 2 hour allowance is acceptable since entry into ITS 3.6.1.1 ACTION A is required if primary containment is inoperable.

M4 CTS 3.7.A.8 requires the reactor to be in the cold condition within 24 hours if the requirements of CTS 3.7.A.5 cannot be met. ITS 3.6.1.7 Required Action C.1 places the plant in MODE 3 in 12 hours if the Required Action and Associated Completion Times are not met. In addition, Required Action C.2 places the plant in MODE 4 in 36 hours (see L2). The allowed Completion Times in Required Actions C.1 and C.2

7 PA1

BASES

7 PA1

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.2 (continued)

X6

CLB4

~~Additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after either a discharge of steam to the suppression chamber from the safety/relief valves or after an operation that causes any of the vacuum breakers to open.~~

CLB2

7 PA1

SR 3.6.1.8.3

1 I

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of [0.5] psid is valid. ~~The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ ~~For this facility,~~ the ~~[18]~~ 24 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because ~~of~~ 4 ~~other surveillances~~ SR 3.6.1.7.2 is performed at a shorter ~~Frequencies~~ Frequency that convey 5 the proper functioning status of each vacuum breaker.

DB8 CLB3 PA3 PA3

REFERENCES

PA5

1. ~~U~~ FSAR, Section ~~[6.2]~~ 14.6.1.3.3. DB5

2. UFSAR, Section 5.2.4.2
3. Preliminary Hazards Summary Report, Bodega Bay Atomic Park Unit Number 1, Docket No. 50-205, Appendix I, December 28, 1962

DB3

4. 10 CFR 50.36 (c)(2)(ii)

X1

Revision I

Specification 3.6.1.9

A1

3.~~5 (cont'd)~~

4.~~5 (cont'd)~~

		Item Surveillance A1	Frequency
[SR 3.6.1.9.1]	e.	a verification that each valve (manual, power operated, or automatic) in the flowpath that is not locked, sealed or otherwise secured in position, is in the correct position. or can be aligned to the correct position A2	Once per 31 Days
[SR 3.6.1.9.3]	f.	~~an air test shall be performed~~ LA3 on the containment spray headers and nozzles is unobstructed	Once per ~~5~~ 10 L3 Years

See ITS: 3.7.1

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

[ACTION A] 3. Should one of the containment cooling subsystems become inoperable or should one RHRSW pump in each subsystem become inoperable, continued reactor operation is permissible for a period not to exceed 7 days*.

See ITS: 3.7.1

L5

add ACTION B

[ACTION C] 4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within ~~24~~ 36 hr.

MODE 3 in 12 hours M1

L2

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature <212°F with an inoperable component(s) as specified in 3.5.B above.

See ITS 3.10.B

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter. A4 When one RHRSW pump in each subsystem becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.

****During the installation of modification 00-125 to the "B" RHRSW strainer, continued reactor operation is permissible for a period not to exceed 11 days.***

See ITS: 3.7.1

CTS AMD271

I

Page 2 of 2
Revision I

⟨Insert B 3.6.1.9⟩ CLB1

CLB1

BASES

APPLICABLE SAFETY ANALYSES (continued)

DB2

INSERT ASA

with containment spray operation the primary containment pressure remains within design limits.

X1

The RHR Containment Spray System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 7).

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure and temperature below design limits. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

DB2

DB6

DB2

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization and heating of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR containment spray subsystems OPERABLE is not required in MODE 4 or 5.

I

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE RHR containment spray subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time was chosen in light of the redundant RHR containment capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

PA2

(continued)

Insert Page B 3.6-57c

QB2

INSERT ASA

Steam line breaks have been analyzed to develop a drywell temperature history for use in equipment qualification (Refs. 3, 4 and 5). The RHR containment sprays are assumed to be initiated at a minimum time of 10 minutes. The RHR containment spray flow rates were assumed to be 7,150 gpm for drywell sprays and 600 gpm for suppression chamber sprays. The highest temperature envelope is 330°F for the first 200 seconds and this is as a result of a 0.75 ft^2 steam line break (Ref. 5). This temperature exceeds the containment design temperature of 309°F but is acceptable since the drywell design temperature limit is applicable coincident with a drywell design pressure of 56 psig (Ref. 6).

APPLICABLE SAFETY ANALYSES (continued)

The maximum allowable equivalent flow path area for bypass leakage has been specified to be 0.032 ft^2. The analysis demonstrates that with containment spray operation the primary containment pressure remains within design limits.

Steam line breaks have been analyzed to develop a drywell temperature history for use in equipment qualification (Refs. 3, 4 and 5). The RHR containment sprays are assumed to be initiated at a minimum time of 10 minutes. The RHR containment spray flow rates were assumed to be 7,150 gpm for drywell sprays and 600 gpm for suppression chamber sprays. The highest temperature envelope is 330°F for the first 200 seconds and this is as a result of a .75 ft^2 steam line break (Ref. 5). This temperature exceeds the containment design temperature of 309°F but is acceptable since the drywell design temperature limit is applicable coincident with a drywell design pressure of 56 psig (Ref. 6).

The RHR Containment Spray System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure and temperature below design limits. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

I

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization and heating of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR containment spray subsystems OPERABLE is not required in MODE 4 or 5.

(continued)

BASES (continued)

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE RHR containment spray subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time was chosen in light of the redundant RHR containment capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

B.1

With two RHR containment spray subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this Condition, there is a substantial loss of the primary containment bypass leakage and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to remove heat from primary containment are available.

C.1 and C.2

If any Required Action and associated Completion Time is not met 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 12 hours and to MODE 4 within 36 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.

(continued)

DISCUSSION OF CHANGES
ITS: 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted that do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 Not used.

A3 CTS 4.7.A.1 requires the torus temperature to be monitored as specified in CTS Table 4.2-8. The Frequency of the Surveillance in CTS Table 4.2-8 is daily. The cross reference in CTS 4.7.A.1 to Table 4.2-8 is being deleted and the Frequency of 24 hours is being included in ITS SR 3.6.2.1.1. Since the current Surveillance Frequency in Table 4.2-8 is daily, this change is administrative. This change is consistent with the requirements and format of NUREG-1433, Revision 1.

A4 During testing that adds heat to the suppression pool, CTS 4.7.A.1 requires the pool temperature to be continuously recorded until heat is terminated or in lieu of continuously recording, the operator shall log the temperature every 5 minutes. In addition, the CTS requires the operator to verify the average temperature is within applicable limits every 5 minutes. Under the same conditions, ITS SR 3.6.2.1.1 requires the suppression pool temperature to be verified to be within the applicable limit once per 5 minutes when performing testing that adds heat to the suppression pool. The requirements to record or log the suppression pool temperature has been deleted from the Technical Specifications. This requirement duplicates the requirements of 10 CFR 50 Appendix B, Section XVII (Quality Assurance records): maintain records of activities affecting quality, including the results of tests (i.e., Technical Specification Surveillances). Compliance with 10 CFR 50 Appendix B is required by the JAFNPP Operating License. The details of the regulations within the Technical Specifications are repetitious and unnecessary. Therefore, retaining the requirement to perform the associated surveillances and eliminating the details from Technical Specifications that are found in 10 CFR 50 Appendix B is considered a presentation preference, which is administrative.

A5 CTS 3.7.A.1.c.(1) requires the maximum water temperature of the suppression pool to be ≤ 95°F during normal power operation. CTS 1.0.0 defines "Reactor Power Operation" to be any operation with the Reactor Mode Switch in the Startup/Hot Standby or Run position with the reactor

ADMINISTRATIVE CHANGES

A5 (continued)

critical and above 1% rated thermal power. ITS LCO 3.6.2.1.a requires the suppression pool average temperature to be ≤ 95°F when THERMAL POWER is > 1% RTP and no testing that adds heat to the suppression pool is being performed. The addition of the words concerning testing (no testing) is considered an administrative change since testing is accounted for in the suppression pool temperature limits in CTS 3.7.A.1.c.(2) (ITS LCO 3.6.2.1.b). Since testing is not mentioned in CTS 3.7.A.1.c.(1) it has been added for clarity in ITS LCO 3.6.2.1.a. The exclusion of the details concerning the Reactor Mode Switch position and whether or not the reactor is critical is also considered to be an administrative change since the ITS 3.6.2.1 Applicability (MODES 1, 2, and 3), the MODES Table (ITS Table 1.1-1), and the requirement that the LCO is applicable when THERMAL POWER is > 1% RTP is sufficient.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 ITS 3.6.2.1 Required Action A.1 is proposed to be added to CTS 3.7.A.1.c to verify that temperature is ≤ 110°F once per hour, anytime temperature has exceeded 95°F and no testing that adds heat to the suppression pool is being performed. This is an additional restriction on plant operation but is necessary to ensure the suppression pool temperature remains ≤ 110°F since additional Actions are required at suppression pool temperatures greater than 110°F (see ACTION D).

M2 CTS 3.7.A.1.c.(3) requires the reactor to be scrammed if the pool temperature reaches 110°F. ITS 3.6.2.1 ACTION D requires, in addition to scramming the reactor by placing the reactor mode switch in the shutdown position immediately (ITS 3.6.2.1 Required Action D.1), that suppression pool temperature be verified ≤ 120°F once per 30 minutes (ITS 3.6.2.1 Required Action D.2) and that the reactor be placed in MODE 4 within 36 hours (ITS 3.6.2.1 Required Action D.3) if the suppression pool temperature is > 110°F but ≤ 120°F. These changes are more restrictive but necessary since the new requirement places the plant outside the conditions of the LCO. This is an additional restriction on plant operation necessary to ensure plant operations remain within the bounds of the containment analyses.

M3 CTS 3.7.A.1.c.(4) and ITS 3.6.2.1 Required Action E.1 require that the reactor pressure vessel be depressurized to less than 200 psig if pool temperature reaches 120°F. However, CTS 3.7.A.1.c.(4) is applicable

TECHNICAL CHANGES - MORE RESTRICTIVE

M3 (continued)

only "during reactor isolation conditions" when the only methods available for depressurizing (cooling) the reactor vessel rely on the suppression pool and require that this depressurization (cooldown) be performed "at normal cooldown rates." ITS 3.6.2.1 ACTION E is applicable whether or not the reactor is isolated. Additionally, ITS 3.6.2.1 Required Action E.2 requires the reactor be in MODE 4 within 36 hours. Therefore, the proposed change is more restrictive. The completion time for depressurizing the reactor to less than 200 psig is changed from proceeding "at normal cooldown rates" to within 12 hours because it is a reasonable time considering that cooling the reactor (if isolated) may involve adding additional heat to the suppression pool that is already greater than 120°F. This change ensures the appropriate actions are taken in the event the plant operates outside the bounds of the containment analysis.

M4 CTS 3.7.A.1 requires the torus (suppression pool) water temperature to be within limits whenever the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel. The scope of the current Applicability covers MODE 1, 3 and portions of MODE 2 operations. The Applicability in ITS 3.6.2.1 is MODES 1, 2 and 3. This change is considered more restrictive since the suppression pool water temperature will be required to be Operable at all times in MODE 2 even prior to any plant startup when reactor coolant temperature may be below 212°F. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.7.A.1.c.(1) requires the suppression pool temperature to be ≤ 95°F during normal power operation. If this limit is exceeded CTS 3.7.A.8 must be entered and the reactor must be in cold condition within 24 hours. ITS 3.6.2.1 Required Action A.2 requires that the suppression pool temperature be restored to ≤ 95°F within 24 hours if temperature is > 95°F but ≤ 110°F, power is > 1% RTP and no testing that adds heat to the suppression pool is being performed. This change is less restrictive than CTS 3.7.A.1.c.(1), which does not allow any time to restore the temperature to within limits. This change is consistent

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

with CTS 3.7.A.1.c.(2), which allows 24 hours to restore temperature only in connection with testing which adds heat to the suppression pool. The proposed Required Action is reasonable based on the fact that the CTS currently allow 24 hours to restore suppression pool temperature for the condition most likely to result in temperatures > 95°F in the suppression pool.

L2 CTS 3.7.A.1.c.(1) requires the suppression pool temperature to be ≤ 95°F during normal power operation. If this limit is exceeded CTS 3.7.A.8 must be entered and the reactor must be in cold condition within 24 hours. ITS 3.6.2.1 ACTION B requires power to be reduced to ≤ 1% RTP within 12 hours if the Required Actions and associated Completion Times of Condition A (see L1 and M1) are not met. Currently the plant would be required to enter CTS 3.7.A.8 and the reactor placed in a cold condition within 24 hours. Proposed ACTION B is less restrictive in that it deletes the requirement that the reactor must be in a cold condition. The 12 hour Completion Time is reasonable, based on operating experience, to reduce power from full power conditions in an orderly manner and without challenging plant systems. The requirement to only reduce power to ≤ 1% RTP is acceptable since it places the plant outside of the conditions of the LCO.

L3 CTS 4.7.A.1 requires an external visual inspection of the suppression chamber whenever there is indication of relief valve operation with the local suppression pool temperature reaching 160°F or greater and the primary coolant system pressure greater than 200 psig. This surveillance is being deleted in accordance with NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," dated December 1984. The basis for deleting this surveillance is that testing has demonstrated that there are no undue loads on the suppression pool or its components at elevated temperatures and pressures when SRVs discharge through "quenchers" (spargers). At JAFNPP each relief valve discharge line terminates in a T-quencher (sparger). Therefore, the requirement for an external visual inspection of the suppression chamber is being deleted.

L4 CTS 4.7.A.1 requires monitoring suppression pool temperature when "there is indication of relief valve operation or testing which adds heat to the suppression pool." ITS SR 3.6.2.1.1 requires frequent monitoring of the suppression pool while performing testing which adds heat to the suppression pool. The requirement to monitor suppression pool temperature whenever there is indication of relief valve operation is proposed to be deleted. If a relief valve is not opened for testing,

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

monitoring suppression pool temperature is part of the coordinated response to an unplanned transient which is governed by plant procedures. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. ITS SR 3.0.1 also states that failure to meet a Surveillance even if experienced between performances of the Surveillance, shall be failure to meet the LCO. Therefore, the limits on suppression pool temperature in ITS 3.6.2.1 and the associated Surveillance Requirement, to periodically monitor suppression pool temperature, are still applicable during the transient and are adequate to ensure the suppression pool temperature is appropriately monitored.

L5 CTS 3.7.A.1.c.(2) allows the suppression pool normal power operation temperature limit of ≤ 95°F specified in CTS 3.7.A.1.c.(1) to be exceeded by no more than 10°F during testing that adds heat to the suppression pool and requires the temperature of the suppression pool to be restored to ≤ 95°F within 24 hours if the limit of CTS 3.7.A.1.c.(1) is exceeded. In addition, if the CTS 3.7.A.1.c (2) limit of ≤ 105°F (95°F plus the 10°F increase allowed during testing) is exceeded, or if the suppression pool temperature is not restored to ≤ 95°F within 24 hours, CTS 3.7.A.8 requires the reactor to be placed in the cold condition within 24 hours.

In the proposed ITS, when suppression pool temperature exceeds 95°F (with thermal power > 1% RTP and during testing that adds heat to the suppression pool), no action is required until the suppression pool temperature exceeds 105°F (ITS 3.6.2.1, ACTION C.1). Once testing that adds heat to the suppression pool is suspended (due to action taken as required by ITS 3.6.2.1, ACTION C.1 or for any other reason such as test completion), ITS 3.6.2.1, CONDITION A becomes applicable and if the suppression pool temperature is > 95°F, ACTION A.2 requires the suppression pool temperature be restored to ≤ 95°F within 24 hours. In the proposed ITS the time period that the suppression pool temperature may be > 95°F may be more than 24 hours since no action is required to restore the temperature to ≤ 95°F until: 1) the temperature exceeds 105°F during testing (ACTION C.1 is applicable) and, 2) ACTION A.2 becomes applicable when testing is suspended as required by ACTION C.1. This sequence may result in the suppression pool temperature being > 95°F for more than 24 hours because action to restore the suppression pool temperature to ≤ 95°F is not required to be initiated until testing is terminated. While this combination of conditions allowed in ITS 3.6.2.1 is less restrictive than CTS 3.7.A.1.c (2) and CTS 3.7.A.8, the proposed ITS 3.6.2.1 ACTIONS are acceptable for the following reasons.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 (continued)

The ITS 3.6.2.1 ACTION C.1 requirement to only require the immediate suspension of testing that adds heat to the suppression pool (rather than also requiring the plant be placed in the cold condition) when the suppression pool temperature is > 105°F (and with thermal power > 1 % RTP while testing that adds heat to the suppression pool is being performed) is acceptable because once the testing that adds heat is suspended the suppression pool cooling function of the Residual Heat Removal (RHR) System is capable of restoring the temperature to within limits.

The ITS 3.6.2.1 ACTION A.2 requirement to restore suppression pool temperature to ≤ 95°F within 24 hours is the same as the CTS 3.7.A.1.c.(2) requirement (once ITS 3.6.2.1, CONDITION A is entered).

The ACTIONS associated with ITS 3.6.2.1, CONDITIONS A and C, maintain the suppression pool temperature within the bounds of the assumptions used in the containment analyses and the changes are consistent with NUREG-1433, Revision 1.

TECHNICAL SPECIFICATIONS - RELOCATIONS

None

Specification 3.6.2.3

A1

~~3.5 (cont'd)~~

See ITS: 3.7.1

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

[ACTION A]

3. Should one of the containment RHR Suppression Pool cooling subsystems become inoperable ~~or should one RHRSW pump in each subsystem become inoperable,~~ continued reactor operation is permissible for a period not to exceed 7 days.*

See ITS: 3.7.1

add ACTION B L4

[ACTION C]

4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within ~~24~~ 36 hr. C3

MODE 3 in 12 hours M2

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature <212°F with an inoperable component(s) as specified in 3.5.B above.

See ITS: 3.10.8

~~4.5 (cont'd)~~

	~~Item~~ Surveillance	Frequency
e.	a verification that each valve (manual, power operated, or automatic) in the flowpath that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position	Once per 31 Days
f.	an air test shall be performed on the containment spray headers and nozzles.	Once per 5 Years

[SR 3.6.2.3.1]

A2

See ITS: 3.6.1.9

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. ~~When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter.~~ ~~When one RHRSW pump in each subsystem~~ becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.

A3

During the installation of modification 00-125 to the "B" RHRSW strainer, continued reactor operation is permissible for a period not to exceed 11 days.

See ITS: 3.7.1

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APPLICABLE SAFETY ANALYSES (continued)

suppression pool temperature is calculated to remain below the design limit.

XI

The RHR Suppression Pool Cooling System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii) (Ref. 4).

LCO

Following PA2 ~~During~~ a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Ref. ~~1~~ 3). DB1 To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE with power from two safety related ~~independent~~ redundant power supplies. PA2 Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active component failure. An RHR suppression pool cooling subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

I

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment and cause a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single active component failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool

(continued) PA2

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2 DB2

required PA4

PA1 performance

PA2 tests

Verifying that each required RHR pump develops a flow rate ≥ [7700] gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 2 5). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice ~~inspections~~ tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is [in accordance with the Inservice Testing Program or 92 days].

5 DB1

I

CLB1

REFERENCES

PA1

1. [U]FSAR, Section [6.2]. 14.6.1.3.3 DB3

2. 5 DB1 ASME, Boiler and Pressure Vessel Code, Section XI.

INSERT REF DB1

X1

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APPLICABLE SAFETY ANALYSES (continued)

bulk suppression pool temperatures following certain events including small break LOCAs and a stuck open S/RV. The analyses indicates that the heat removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The suppression pool temperature is calculated to remain below the design limit.

The RHR Suppression Pool Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

Following a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Ref. 2). To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE with power from two safety related redundant power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active component failure. An RHR suppression pool cooling subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE. 1/I

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment and cause a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the

(continued)

ACTIONS

A.1 (continued)

overall reliability is reduced because a single active component failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

B.1

With two RHR suppression pool cooling subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment pressure and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and the potential avoidance of a plant shutdown transient that could result in the need for the RHR suppression pool cooling subsystems to operate.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, 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 12 hours and to MODE 4 within 36 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 3.6.2.3.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each required RHR pump develops a flow rate ≥ 7700 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 5). This test confirms one point on the pump performance curve, and the results are indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

I

REFERENCES

1. UFSAR, Section 14.6.1.3.3.
2. GE-NE-T23-0737-01, James A. FitzPatrick Nuclear Power Plant Higher Service Water Temperature Analysis, August 1996.

(continued)

BASES

REFERENCES (continued)	3.	NEDC-24361-P, James. A FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response, August 1981.
	4.	10 CFR 50.36 (c)(2)(ii).
	5.	ASME, Boiler and Pressure Vessel Code, Section XI.

DISCUSSION OF CHANGES
ITS: 3.6.2.4 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted that do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 The reference in CTS 4.7.A.7.a to surveillance requirements of Table 4.2-8 is being deleted since the ITS does not use cross references. The surveillances in current Table 4.2-8 and the proposed Surveillances in ITS 3.3.3.1 are adequate to ensure the instrumentation is functioning properly. Any changes to the current Surveillance Requirements in Table 4.2-8 are discussed in the Discussion of Changes for ITS 3.3.3.1, "Post Accident Monitoring Instrumentation." Since the removal of this cross reference does not change any technical requirements this change is considered administrative and is consistent with the format of NUREG-1433, Revision 1. 1/I

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 3.7.A.7.a.(2) allows the differential pressure to be outside its limit for a maximum of 4 hours as a result of required operability testing of HPCI, RCIC, and the Suppresssion Chamber - Drywell Vacuum Breaker System. The details of which Surveillance Tests this allowance is provided for is proposed to be relocated to the Bases. The allowance in the Note to ITS SR 3.6.2.4.1 that the limit is not required to be met for 4 hours during Surveillances that cause the drywell-to-suppression chamber differential pressure to be outside the limit is adequate to ensure the allowance is taken only during planned testing. The specific details of the which Operability Surveillance is not necessary to be in the Specification. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - MORE RESTRICTIVE

M6 An actual or simulated automatic isolation test (ITS SR 3.6.4.2.3) has been added to the requirements of CTS RETS Table 3.10-2 Item 2 (Refuel Area Exhaust Monitors and Recorders) to ensure both a Logic System Functional Test as well as an actual or simulated automatic isolation test is performed for this Secondary Containment Isolation Instrumentation Function. The new Surveillance will ensure the Function is properly tested throughout their operating sequence. This surveillance is not currently required to be performed, therefore, this change is considered more restrictive on plant operation but is added to enhance plant safety. I

M7 ITS SR 3.6.4.2.1, the requirement to verify that each secondary containment isolation manual valve, blind flange, or equivalent that is required to be closed during accident conditions is closed, every 31 days, is being added to CTS 4.7.C. This Surveillance verifies the secondary containment isolation devices are in the correct position to ensure the secondary containment will perform as assumed in the safety analysis. Since the SCIVs are readily accessible to personnel during normal operation and position verification is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions. For clarification Note 1 has been added to the SR which allows the verification of these devices in high radiation areas to be performed by administrative means. This is acceptable since access to these areas is typically restricted during MODES 1, 2 and 3 for ALARA reasons. Note 2 is also included in the SR which does not require the SR to be met for SCIVs that are open under administrative control. This is acceptable since the Bases says that the administrative controls will require stationing a dedicated operator at the controls of the valve who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. The addition of a new Surveillance Requirement imposes added operational requirements and, therefore, constitutes a more restrictive change. This change is not considered to result in any reduction to safety.

M8 ITS SR 3.6.4.2.2, the requirement to verify that the isolation time of each power operated automatic SCIV is within limits in accordance with the Frequency requirements of the Inservice Testing Program, is being added to CTS 4.7.C. This Surveillance verifies the secondary containment isolation valves function to ensure the secondary containment will perform as assumed in the safety analysis. The addition of new Surveillance Requirements imposes additional operational requirements and, therefore, constitutes a more restrictive change. This change is not considered to result in any reduction to safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 (continued)

monitored. As a result, leaks or pipe breaks would typically be detected before significant inventory loss occurred. These activities would typically be performed after refueling when few noncondensible gases remain in the reactor coolant. The temperature limitation of 212°F will ensure that water not steam would be emitted from the postulated leak or pipe break. In addition under these conditions, stored energy is sufficiently low that even with a loss of inventory following a recirculation line break, core coverage would be maintained by the low pressure Emergency Core Cooling systems required per ITS 3.5.2 and the fuel would not exceed its peak clad temperature limit. As a result, the potential for failed fuel and a subsequent increase in reactor coolant activity is minimized and significant releases of radioactive material would not be expected to occur. Therefore, it is considered acceptable to eliminate the requirement to maintain secondary containment Operability with the Reactor Coolant System not vented in MODE 4 or 5.

L4 CTS Table 4.2-1 Note 7 and RETS Table 3.10-2 Note f requires the performance of a simulated automatic actuation of the Standby Gas Treatment System. In addition CTS 4.7.B.1.d requires an automatic initiation test of the Standby Gas Treatment System. These test requirements are identical. ITS SR 3.6.4.3.3 includes the phrase "actual or," in reference to the Standby Gas Treatment automatic initiation signal. This allows satisfactory automatic system initiations to be used to fulfill the Surveillance Requirements. Operability is adequately demonstrated in either case since the SGT subsystem itself can not discriminate between "actual" or "simulated" signals.

L5 Not used.

TECHNICAL CHANGES - RELOCATIONS

None

MODIFIED RAI RESPONSES FOR ITS SECTION 3.6

3.6.4.3-2 DOC A4
JFD CLB3
JFD Bases CLB3
CTS 4.7.B.1.e
STS SR 3.6.4.2.4 and Associated Bases
ITS SR 3.6.4.2.4 and Associated Bases

CTS 4.7.B.1.e requires that manual operability of the bypass valve for filter cooling shall be demonstrated once per 24 months. The corresponding ITS SR is ITS SR 3.6.4.3.4 which requires verifying the SGT System filter cooling cross tie valves are open. There are a number of changes made to the CTS SR in order to arrive at the ITS SR wording. All the changes are considered and described as Administrative changes (DOC A4, JFD CLB 3, and JFD Bases CLB 3). This is incorrect. The changes associated with the valve nomenclature and the addition of the ITS SR Note are discussed in Comment Numbers 3.6.4.3-1 and 3.6.4.3-3 respectively. CTS 4.6.B.1.e requires demonstrating manual operability of the valves which implies operating or stroking the valve. The ITS only verifies that the valve is open. STS SR 3.6.4.3.4 excluding fan starting would more accurately be a reflection of the CTS requirements of opening and closing the valve. Thus, the change to the CTS would be Administrative only if the STS words were used. Thus the proposed ITS change would be considered as a Less Restrictive (L) change.

Comment: Revise the CTS/ITS markups as appropriate and provide the appropriate discussion and justification for this change. See Comment Numbers 3.6.4.3-1 and 3.4.3.3-2.

Licensee Response:

1. Refer to response to item 3.6.4.3-1 and Figures RAI 3.6.4.3-1.1 and RAI 3.6.4.3-1.2

2. As shown on Figure RAI 3.6.4.3-1.1, the manual valves associated with filter decay heat cooling are labeled 3A and 3B. CTS 4.7.B.1.e and ITS SR 3.6.4.3.4 are the applicable surveillance requirements. Valves 3A and 3B must be capable of being full open to provide the decay heat cooling air flow path for the idle subsystem filter (see Figure RAI 3.6.4.3-1.2) and must also be capable of full closure to allow isolation of one subsystem filter for maintenance purposes while maintaining operability of the other subsystem.

3, NYPA will revise the NUREG markup of SR 3.6.4.3.4 to require cycling of valves 3A and 3B as suggested.

Revision I Changes to Section 3.6 RAI Responses

[Revised Response provided with Revision I package]

1. The Licensee has revised the ITS 3.6.4.3 Conversion Package to reflect the current licensing basis with respect to surveillance testing associated with filter decay heat cooling valves (cooling cross-tie valves) 3A and 3B. The CTS markup was revised at 4.7.B.1.e by deletion of the Note previously added (in Revision A) and associated DOC A4 was revised to reflect the retention of the current requirement for filter cooling capability by demonstrating manual operability of the cooler bypass valves (cooling cross-tie valves). NUREG SR 3.6.4.3.4 markup was revised by deletion of the Note which had been added in Revision A and the text of the SR was revised to be consistent with the current licensing basis by requiring the cooling valves be manually cycled at a Frequency of 24 months. NUREG SR 3.6.4.3 Bases markup was revised to reflect the changes to the text of the SR, deletion of "Insert SR 3.6.4.3.4 Note" for the SR Note discussion.

2. See RAI 3.6.4.3-3 below for additional discussion of the ITS SR 3.6.4.3.4 Note.

3.6.4.3-3 DOC A4
JFD CLB 3
JFD Bases CLB 3
CTS 4.7.B.1.e
ITS SR 3.6.4.2.4 and Associated Bases

CTS 4.7.B.1.e requires that manual operability of the bypass valve for filter cooling shall be demonstrated once per 24 months. The corresponding ITS SR is ITS SR 3.6.4.3.4 which requires verifying the SGT System filter cooling cross tie valves are open. There are a number of changes made to the CTS SR in order to arrive at the ITS SR wording. All the changes are considered and described as Administrative changes (DOC A4, JFD CLB3, and JFD Bases CLB3). This is incorrect. The changes associated with the valve nomenclature and manual operability of the valve are discussed in Comment Numbers 3.6.4.3-1 and 3.6.4.3-2 respectively. CTS 4.7.B.1.e is modified by a Note which exempts performance of ITS SR 3.6.4.3.4 when one SGT subsystem is isolated. The CTS does not currently allow this exemption. Thus the addition of the Note would be a Less Restrictive (L) change. In addition insufficient information is provided to make a determination that the addition of the Note is plant specific. There is the potential that this change is generic and thus beyond the scope of review for this conversion.

Comment: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change based on plant specific system design or operational constraints.

Licensee Response:

1. NYPA does not agree that the addition of the Note to ITS SR 3.6.4.3.4 should be classified as a less restrictive change.

2. While CTS does not contain explicit wording that states that closure of a decay heat cooling cross-tie valve (see response to item 3.6.4.3-1 and valves 3A and 3B as shown on Figure RAI 3.6.4.3-1.1) when performing maintenance on a filter is allowed, it is obvious that the failure to close a cross-tie valve would result in the inoperability of the SGT subsystem which is not out of service for maintenance (and thus both subsystems would be inoperable whenever maintenance is performed on either subsystem). Closing a cross-tie valve (either valve 3A or 3B) to preserve the operability of the other subsystem during maintenance on a SGT subsystem filter is explicitly addressed in plant procedures and is consistent with the interpretation of CTS.

3. Note that an essentially identical Administrative change was included in the Cooper ITS conversion. While there are some differences between the Cooper and FitzPatrick SGT system configurations, the flow path for a subsystem filter decay heat cooling is essentially the same. The location of cooling air introduction to a subsystem filter and use of a cross-tie line to the other subsystem fan suction in both the Cooper and FitzPatrick SGT systems makes isolation of the cross-tie line mandatory

if inoperability of both subsystems is to be avoided during maintenance of either subsystem filter.

[Revised Response provided with Revision I package]

1. Refer to the Revised Response to RAI 3.6.4.3-2 above.

3.7.A.3-2 DOC R1
CTS 3.7.A.3
ITS SR 3.6.1.3.1 and Associated Bases

CTS 3.7.A.3 states that "The containment shall be purged through the Standby Gas Treatment System whenever the primary containment integrity is required." Based on the CTS and the ITS markup, the staff does not conclude that CTS 3.7.A.3 can be relocated out of the JAFNPP TS. CTS 3.7.A.3 has a direct relationship to the OPERABILITY of the containment vent and purge valves, in that purging shall be through the SGT System. If purging cannot be through the SGT System then based on CTS 3.7.A.3 the vent and purge valves need to be closed. Thus CTS 3.7.A.3 needs to be retained in ITS SR 3.6.1.3.1 either as part of Note 1 or as a separate Note.

Comment: Revise the CTS/ITS markups and provide, as appropriate, additional discussion and justification adding CTS 3.7.A.3 to ITS SR 3.6.1.3.1.

Licensee Response:

1. NYPA does not agree that CTS 3.7.A.3 (or a modified form of CTS 3.7.A.3) should be added to ITS SR 3.6.1.3.1 or the associated Note. Requiring primary containment purge through the Standby Gas Treatment (SGT) system (or discontinuation of purging without delay if the requirement is not met) is to cause filtering of any radioactive materials that might be present in the primary containment atmosphere when purging. Filtering the purge path in the SGT system and releasing the effluent through the plant stack (elevated release point) minimizes the resulting dose consistent with the objectives of Radiological Effluent Technical Specifications.

   Creation of a purge path that is not through the Standby Gas Treatment (SGT) System would not have any effect on the automatic or manual closure capabilities of the primary containment vent and purge valves under either normal or accident conditions. Similarly, the inoperability of the vent and purge valves has no effect on the purge path (if any path is assumed to exist with inoperable vent and purge valves). In other words, there is no relationship between the purge path and operability of the vent and purge valves. Thus, no modification ITS SR 3.6.1.3.1 or the associated Note to reflect CTS 3.7.A.3 is necessary.

2. CTS 3.7.A.3 was added to CTS as part of Facility Operating License Amendment 93 dated May 29, 1985. Amendment 93 added Radiological Effluent Technical Specifications (RETS) to bring the License into compliance with 10 CFR 50, Appendix I.

   As discussed in the NRC letter that transmitted Amendment 93, NYPA initially submitted proposed RETS to the Commission on April 29, 1983. The April 29, 1983 submittal did not include any proposed changes to CTS 3.7.A.3. The April 29, 1983 submittal was subsequently superseded by a

December 21, 1984 submittal and finally supplemented by a February 19, 1995 submittal. The December 21, 1984, submittal reflected the results of an August 8, 1984 meeting between the NRC and NYPA and included the addition of the requirements of CTS 3.7.A.3.

Attachments to the December 21, 1984 submittal letter (Attachment III, titled "Safety Evaluation for Proposed RETS and Associated Changes" and Attachment IV, titled "Additional Information Concerning the Differences Between the Revised RETS and NUREG-0473, Revision 3, Draft 7") do not include any discussions that are specific to the requirements contained in CTS 3.7.A.3.

The NRC approved the RETS submittal (as Amendment 93) without any changes to the requirements of CTS 3.7.A.3. The NRC safety evaluation for Amendment 93 (and the associated "Technical Evaluation Report" which was prepared by a contractor) do not include any specific discussion of CTS 3.7.A.3.

Based on the recollection of NYPA personnel involved in the preparation of the RETS submittal and a number of meetings and discussions with NRC personnel concerning the implementation of Technical Specifications intended to bring the plant into compliance with 10 CFR 50, Appendix I, and the absence of documentation to the contrary, it is concluded that CTS 3.7.A.3 was apparently considered to be a necessary component of RETS implementation. It follows that relocation of RETS requirements should include the relocation of CTS 3.7.A.3 to the ODCM as proposed by NYPA in DOC R1.

3. It should be noted that the only primary containment purge pathway that exists (without modification of the plant) is through the SGT System so that primary containment vent and purge effluents are filtered prior to release (Reference UFSAR 5.2.3.6). During initial construction of the plant the design of the Vent and Purge System, the Reactor Building Ventilation System, the SGT System, and the associated control systems was modified so that the only purge pathway that physically exists is through the SGT System. These changes were made prior to the issuance of the plant operating license in October, 1974.

[Revised Response provided with Revision I package]

1. The Licensee has revised DOC R1 associated with CTS 3.7.A.3 by providing a brief discussion that notes that the only primary containment purge path that exists, by design, is via the SGT System.

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION I

Source of Change	Summary of Change	Affected Pages
mendment 271	This amendment deleted the Note concerning modification 99-095 and replaced it with a similar Note concerning modification 00-125. However, similar to the previous Discussion of Change about the modification, this new modification has also been completed, thus the amendment only affects the CTS markup and Discussion of Changes; not the ITS.	**Specification 3.7.1** CTS markup p 2 of 2 DOC A2 (DOCs p 1 of 6)
Consistency issue	The CTS requires the verification of gross gamma activity rate after a 50% increase in the nominal steady state fission gas release if the gross gamma activity rate is "≥ 5000 μCi/second". Therefore, the Frequency of SR 3.7.5.1, which specifies the activity rate as "> 5000 μCi/second", is being changed. Also, for consistency with the manner in which these types of modifiers are used in the Frequency column (see SR 3.6.1.1.2), the activity rate modifier has been made into a Note to the Frequency.	**Specification 3.7.5** NUREG ITS markup p 3.7-17 NUREG Bases markup p B 3.7-32 Retyped ITS p 3.7-15 Retyped ITS Bases p B 3.7-31
Typographical errors	Minor typographical errors in the Discussion of Changes have been corrected. (DOC L3, ""met until after 31 days" changed to "performed until 31 days"; DOC R1, "CTS RETS 3.5.1.b" changed to "CTS RETS 3.5.b" in three locations and "able 3.10-2" changed to "Table 3.10-2" in the first sentence of the second paragraph.)	**Specification 3.7.5** DOCs L3 and R1 (DOCs p 2 of 4, 3 of 4, and 4 of 4)
Technical change	The change to the Frequency for the cycling of the main turbine bypass valves has been made as agreed to by JAFNPP during a phone conversation with the NRC.	**Specification 3.7.6** NUREG ITS markup p 3.7-18 JFD X3 (JFDs p 2 of 2) NUREG Bases markup p B 3.7-35 Retyped ITS p 3.7-16 Retyped ITS Bases p B 3.7-35
Editorial change	The location of the response time limits (which are not currently specified in the CTS) has been changed from the COLR to the Technical Requirements Manual.	**Specification 3.7.6** NUREG Bases markup p B 3.7-36 and Insert page B 3.7-36 Retyped ITS Bases p B 3.7-36
Amendment 268	This amendment provided the title of the licensee (i.e., Entergy Nuclear Operations, Inc.). However, this entire Specification is being relocated to the TRM, thus the ITS is not affected.	**Current Specification 3/4.8** CTS markup p 1 of 1
Amendment 269	This amendment affects the Control Room Emergency Ventilation System, which is on a CTS mark-up page used by this Specification. However, the Amendment does not affect this Specification.	**Current Specification 3/4.11.C** CTS markup p 1 of 1

Specification 3.7.1

A1

~~3.5~~ (cont'd)

[ACTION A] [M2] 2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days ~~provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.~~ L1

add ACTION C Note

[ACTION C] 3. Should one of the containment cooling subsystems become inoperable or should one RHRSW pump in each ~~subsystem become~~ inoperable, continued reactor operation is permissible for a period not to exceed 7 days* A2 [ACTION B] [ACTION C]

add ACTION D L2

[ACTION E] 4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within 24 hr.

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature <212°F with an inoperable component(s) as specified in 3.5.B above.

See ITS: 3.10.8

~~4.5~~ (cont'd)

	Surveillance ~~Item~~	Frequency
[SR 3.7.1.1] e.	a verification that each valve (manual, power operated, or automatic) in the flowpath that is not locked, sealed or otherwise secured in position, is in the correct position. or can be aligned to the correct position L5	Once per 31 Days
See ITS: 3.6.1.9 f.	an air test shall be performed on the containment spray headers and nozzles.	Once per 5 Years

L4

~~2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.~~

~~3. When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter. When one RHRSW pump in each subsystem becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.~~

~~***During the installation of modification 00-125 to the "B" RHRSW strainer, continued reactor operation is permissible for a period not to exceed 11 days.***~~ A2

CTS: Amd 271

4

DISCUSSION OF CHANGES
ITS: 3.7.1 - RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.5.B.3, Footnote *, provides an operational allowance for installation of modification 00-125. This footnote allowance is deleted since the modification has been completed. As such, this is an administrative change.

I

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.5.B.4 requires the reactor to be placed in a cold condition within 24 hours when CTS 3.5.B.2 or 3.5.B.3 cannot be met. CTS 3.5.B.2 covers the conditions with one RHRSW pump inoperable while CTS 3.5.B.4 covers the condition with one containment cooling subsystem (in this case one RHRSW subsystem) or one RHRSW pump in each subsystem inoperable. When two RHRSW subsystems are inoperable the plant must enter CTS 3.0.C and the plant must be placed in a cold shutdown within 24 hours. In ITS 3.7.1, all of the default actions are covered in ACTION E. An additional ACTION has been added to allow time to restore one RHRSW subsystem to Operable status when two RHRSW subsystems are found to be inoperable (ACTION D), however this change is addressed in L2. ITS 3.7.1 Required Action E.1 will require the plant be in MODE 3 within 12 hours when the Required Action and associated Completion Time of ACTIONS A, B, C, or D are not. In addition, ITS 3.7.1 Required Action E.2 requires the plant to be in MODE 4 in 36 hours (L3). This change is more restrictive because it provides an additional requirement to place the plant in MODE 3 in 12 hours. The allowed Completion Times in Required Action E.1 and E.2 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. However, the 12 hour Completion Time ensures timely action is taken to place the plant in a shutdown condition (MODE 3). The consequences of any design bases event is significantly reduced when plant is shutdown. These Completion Times are consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS RETS Table 3.10-1 Note (e) requires the plant to isolate the SJAE or all main steam lines within the next 12 hours if the SJAE release rate is not below the trip level within 72 hours. These actions have been included in ITS 3.7.5 as ACTION A and B. An option has been included in proposed ACTION B allowing the plant to be in MODE 3 within 12 hours and in MODE 4 within 36 hours. This is acceptable since these alternative actions will result in a power reduction which will reduce the coolant activity levels and place the plant in a condition where the Specification does not apply (MODE 4). This change is less restrictive on plant operation since the option is provided and the overall time to exit the applicability is longer.

L2 CTS RETS 3.5.a (LCO) specifies that the limits of gross radioactivity rate of noble gases is given on Table 3.10-1. CTS RETS Table 3.10-1 specifies the trip level setting for the SJAE Radiation Monitors. This limit has been increased from 500,000 to 600,000 μCi/sec consistent with the value used in the Offgas System Failure accident of UFSAR, Section 11.4.7.2. Since a higher value has been included in proposed Specification 3.7.5 this change is considered less restrictive but acceptable since the limit is consistent with the analysis. The trip level setting of the SJAE Radiation Monitors has been relocated as identified in the Discussion of Changes for CTS 3/4.2.D, "Radiation Monitoring Systems - Isolation and Initiation Functions". This change to include the analytical limit in the ITS is consistent with the requirements and format of NUREG-1433, Revision 1.

L3 CTS RETS 3.5.a (Surveillance Requirement) must be performed prior to entry into the mode of applicability in accordance with CTS 3.0.D. A Note has been added to CTS RETS 3.5.a (proposed SR 3.7.5.1) which clarifies when the surveillance must be performed. The Note specifies that the surveillance is not required to be performed until 31 days after any main steam line is not isolated and the SJAE are in operation since in this condition radioactive fission gases may be in the Main Steam Offgas System at significant rates. This change is considered less restrictive since CTS 4.0.D (ITS SR 3.0.4) requires the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 (continued)

surveillance to be met prior to entry into the modes of Applicability. This change is acceptable since a test with the valves isolated provides no meaningful information. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - RELOCATIONS

R1 CTS RETS 3.5.b (LCO and Surveillance Requirement), CTS RETS Table 3.10-1, and Table 3.10-2 specify the requirements for the Steam Jet Air Ejector (SJAE) System radiation monitors. This instrumentation is neither a safety system nor is it connected to the reactor coolant. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The SJAE System monitors are used to provide a continuous check on the releases of radioactive gaseous effluents from the Main Condenser Steam Jet Air Ejector. These Technical Specifications require the Licensee to maintain Operability of various effluent monitors and establish setpoints in accordance with the Offsite Dose Calculation Manual (ODCM). The alarm/trip setpoints are established to ensure that the alarm/trip will occur to prevent exceeding the limits of 10 CFR 20. Plant Design Basis Accident (DBA) analyses do not assume any action, either automatic or manual, resulting from the Steam Jet Air Ejector (SJAE) monitors. ITS 3.7.5, Main Condenser Steam Jet Air Ejector Offgas, will be included in the ITS to ensure the SJAE Offgas failure event will remain within the calculated values of UFSAR, Section 11.4.7.2. Additional administrative controls are also proposed to be added to the Technical Specifications to ensure compliance with the applicable regulatory requirements is maintained. ITS 5.5.1 specifies that future changes to the ODCM will be reviewed to ensure that such changes will "maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations."

CTS RETS 3.5.b (LCO and Surveillance Requirement), CTS RETS Table 3.10-1 and Table 3.10-2 do not identify a parameter which is an initial condition or assumption for a DBA or transient, identify a significant abnormal degradation of the reactor coolant pressure boundary, provide any mitigation of a design basis event and is not a structure system or component which operating experience or PRA has shown to be significant to public health and safety. Therefore, the requirements specified in

TECHNICAL CHANGES - RELOCATIONS

R1 (continued)

CTS RETS 3.5.b (LCO and Surveillance Requirement), CTS RETS Table 3.10-1 and Table 3.10-2 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the JAFNPP Technical Specifications and will be relocated to the ODCM. Changes to the ODCM will be controlled by the provisions of the ODCM change control process described in Chapter 5 of the ITS. This change is consistent with Generic Letter 89-01 for removal of Radiological Effluent Technical Specification (RETS) and relocation to the ODCM. I

PA1

SJAE

PA2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE DB1	FREQUENCY
SR 3.7.~~6~~.1 ⑤ [L3] PA2 → ⑤ [3.5.a (SR) RETS] [M1]	-------------------NOTE------------------- Not required to be performed until 31 days after any ~~[~~main steam line not isolated and~~]~~ SJAE in operation. -- Verify the gross gamma activity rate of the noble gases is ≤ ~~[240]~~ 600,000 µCi/second ~~[after decay of 30 minutes]~~. DB1	CLB1 ---- NOTE ---- Only required when gross gamma activity rate is ≥ 5000 µCi/second ------------ 31 days AND Once within 4 hours after a ≥ 50% increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level I

Revision I

BASES

ACTIONS **B.1, B.2, B.3.1, and B.3.2** (continued)

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.

plant PA1

CLB1

SURVEILLANCE REQUIREMENTS SR 3.7.~~6~~.1 5 PA2

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample, taken at the discharge (prior to dilution and/or discharge) of the SJAE, or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity) PA4 to ensure that the required limits are satisfied. ~~The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85, Kr-87, and Kr-88.~~ X2 If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. As noted, this Frequency is only required when gross gamma activity rate is ≥ 5000 µCi/second. 4 The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, providing offgas isolation on excessive activity PA3, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. DB4 Only in this condition can radioactive fission gases be in the Main Condenser (SJAE) PA1 Offgas System at significant rates.

REFERENCES

1. U PA1 FSAR, Section ~~15.1.35~~ 11.4.7.2 DB2.
2. 10 CFR 100.
3. 10 CFR 50.36 (c)(2)(ii) X1
4. 10 CFR 50, Appendix I CLB1

CLB1 The 5,000 µCi/second threshold level is an administrative control to reduce the number of unnecessary grab samples. This value is approximately 1% of the SJAE trip level setting and operating at or below the threshold level will ensure ~~the~~ site boundary annual radiation exposures remain within the 10 CFR 50, Appendix I guidelines (Ref. 4) CLB1

Revision I

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	------------------NOTE------------------ Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. -- Verify the gross gamma activity rate of the noble gases is ≤ 600,000 μCi/second.	31 days **AND** -----NOTE------ Only required when gross gamma activity rate is ≥ 5,000 μCi/second -------------- Once within 4 hours after a ≥ 50% increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level

I

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain primary containment isolation valve is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging plant systems.

An alternative to Required Actions B.1 and B.2 is to place the plant in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 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 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample, taken at the discharge (prior to dilution and/or discharge) of the SJAE, or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity) to ensure that the required limits are satisfied. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. As noted, this Frequency is only required when the gross gamma activity rate is ≥ 5,000 μCi/second. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas providing offgas isolation on excessive activity, and is acceptable, based on operating experience. The 5,000 μCi/second threshold level is an administrative control to reduce the number of unnecessary grab samples. This value is 1% of the SJAE trip level setting and operating at or below the threshold level will ensure the site boundary annual radiation exposures remain within the 10 CFR 50, Appendix I guidelines (Ref. 4).

(continued)

3.7 PLANT SYSTEMS

3.7.6 The Main Turbine Bypass System

PA1

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

INSERT LCO-1 TA1

[OR

a. TA1

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, ~~are made applicable~~. ; and TA1 DB1]

DB1

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

Insert LCO-2 TA1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. [Requirements of the LCO not met ~~or Main Turbine Bypass System inoperable~~.] DB1	A.1 [Satisfy the requirements of the LCO ~~or restore Main Turbine Bypass System to OPERABLE status~~.] DB1	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify one complete cycle of each main turbine bypass valve. PA1	~~31 days~~ X3 Prior to entering MODE 2 or 3 from MODE 4 L

(continued)

Revision I

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.7.6 - MAIN TURBINE BYPASS SYSTEM

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X3 The Frequency of ITS SR 3.7.6.1 has been changed from 31 days to prior to entering MODE 2 or 3 from MODE 4. Currently, this test is not required in the CTS and is only performed prior to a startup from MODE 4, as required by plant procedures. Monthly full stroke testing causes wear of both the bypass valves and the condenser intervals, leading to leaks and reduced efficiency. In addition, a review of historical maintenance and testing data for approximately the past 10 years has shown that this test normally passes its Surveillance at the proposed Frequency. The above data reviewed determined that there have been no failures of a bypass valve to cycle during this test.

BASES

ACTIONS (continued)

B.1

PA2 LHGR limit and TA1 — operating PA2

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limit~~s~~ for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the ~~turbine generator load rejection~~ abnormal operational DB3 transient. ~~The 4 hour Completion~~ 5 Time is reasonable, based on operating experience, to reach the required ~~unit~~ conditions from full power conditions in an orderly manner and without challenging ~~unit~~ plant PA2 systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~7~~6.1 PA1 (Prior to entering MODE 2 or 3 from MODE 4) X4 I

Cycling each main turbine bypass valve through one complete cycle of full travel ~~demonstrates that the valves are mechanically OPERABLE~~ and will function when required. The ~~31 day~~ specified X4 Frequency ~~is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions~~ X4 , and ensures the valves are OPERABLE prior to each reactor startup from MODE 4. Operating experience has shown that these components ~~usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability~~ standpoint. I

SR 3.7.~~7~~6.2 PA1

required DB3

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. X2 The ~~[18]~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~unit~~ plant PA2 outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the ~~[18]~~ 24 X2 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

(continued)

Revision I

PA1

BASES 6 PA1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in ~~[unit specific documentation]~~ the Technical Requirements Manual (Reference 5) DB2. The ~~[18]~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the ~~[18]~~ 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint. X2 I

REFERENCES PA2

1. UFSAR, Section [7.~~7.4~~ 11]. DB2
2. UFSAR, Section ~~[15.1.1]~~ 14.5. DB2

Insert REF DB3 X1

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DB3

INSERT REF

3. 10 CFR 50.36(c)(2)(ii).

4. J11-03757SRL, Revision 0, Supplemental Reload Licensing Report for James A. Fitzpatrick Reload 14 Cycle 15, August 2000.

5. Technical Requirements Manual. I

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify one complete cycle of each main turbine bypass valve.	Prior to entering MODE 2 or 3 from MODE 4
SR 3.7.6.2	Perform a system functional test.	24 months
SR 3.7.6.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

ACTIONS

A.1 (continued)

accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the LHGR limit and MCPR operating limit for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable safety analyses transients. The 4 hour Completion Time is 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 3.7.6.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The specified Frequency (prior to entering MODE 2 or 3 from MODE 4) is based on engineering judgment, is consistent with the procedural controls governing valve operation, ensures correct valve positions, and ensures the valves are OPERABLE prior to each reactor startup from MODE 4. Operating experience has shown that these components usually pass the SR when performed at the specified Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

I

I

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the required valves will actuate to their required position. The 24 month Frequency is based on the need to

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.2 (continued)

perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

SR 3.7.6.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the Technical Requirements Manual (Reference 5). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. USFAR, Section 7.11.

2. UFSAR, Section 14.5.

3. 10 CFR 50.36(c)(2)(ii).

4. J11-03757SRL, Revision 0, Supplemental Reload Licensing Report for James A. FitzPatrick Reload 14 Cycle 15, August 2000.

5. Technical Requirements Manual.

CTS 3/4.8

3.8 LIMITING CONDITIONS FOR OPERATION

3.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability:

Applies to the handling and use of sealed special nuclear, source and by-product material at all times.

Objective:

To assure that leakage from byproduct, source and special nuclear radioactive material sources does not exceed allowable limits.

Specification:

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall have removable contamination of less than or equal to 0.005 microcuries.

A. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use, and either:

1. Decontaminate and repair the sealed source, or
2. Dispose of the sealed source in accordance with applicable regulations.

4.8 SURVEILLANCE REQUIREMENTS

4.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability:

Applies to the surveillance requirements of sealed special nuclear, source and by-product materials.

Objective:

To specify the surveillances to be applied to sealed special nuclear, source and by-product materials.

Specification:

Tests for leakage and/or contamination shall be conducted as follows:

A. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen-3, with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

B. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed source shall not be put into use until tested.

C. Startup sources shall be leak tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

D. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample. Testing shall be performed by Entergy Nuclear Operations Inc. or by other persons specifically authorized by the NRC or an agreement state.

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R1

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CTS 3/4.11.C

3.11 (cont'd)

3. The control room emergency ventilation system shall not be out of service for a period exceeding 3 days during normal reactor operation or refueling operations. In the event that the system is not returned to service within 3 days, the reactor shall be in cold shutdown within 24 hours and any handling of irradiated fuel, core alterations, and operations with a potential for draining the reactor vessel shall be suspended as soon as practicable

4. Not Used

B. DELETED

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

3. Operability of the main control room air intake radiation monitor shall be tested once/3 months.

See ITS: 3.3.7.1

See ITS: 3.7.3

4. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

5. Main control room emergency ventilation air supply system capacity shall be tested once every 18 months to assure that it is ± 10% of the design value of 1000 cfm.

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B. DELETED

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

LA1

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SUMMARY OF CHANGES TO ITS SECTION 3.8 - REVISION I

Source of Change	Summary of Change	Affected Pages
Amendment 272	This amendment added an allowance that the 7 day allowed outage time for a single offsite circuit could be extended to 14 days (for line #3 and/or one reserve station service transformer only) during the period from 9/9/01 through 9/23/01. Since this time period has passed, the allowance is not being included in the ITS, and this change only affects the CTS markup and Discussion of Changes. The amendment also impacts ITS 3.8.3, since the CTS page has an ITS 3.8.3 requirement on it (however, only the CTS markup page for ITS 3.8.3 is affected, since the amendment is not related to ITS 3.8.3).	**Specification 3.8.1** CTS markup p 2 of 11 DOC A9 (DOCs p 3 of 12) **Specification 3.8.3** CTS markup p 5 of 6
F-RAI 3.8.1-05	The change agreed to by JAFNPP during a meeting with the NRC concerning F-RAI 3.8.1-05 has been made. Specifically, ITS SR 3.8.1.2, the EDG start Surveillance, has been modified to allow a warmup period prior to loading. This is consistent with NUREG-1433.	**Specification 3.8.1** DOC A8 (DOCs p 2 of 12) NUREG ITS markup p 3.8-6 JFD DB5 (JFDs p 6 of 10) NUREG Bases markup p B 3.8-16 Retyped ITS p 3.8-5 Retyped ITS Bases p B 3.8-17
F-RAI 3.8.1-07	The change agreed to by JAFNPP during a meeting with the NRC concerning F-RAI 3.8.1-07 has been made. Specifically, the reason for why the SR is only required to be met for each offsite circuit that is not energizing its respective 4.16 kV bus has been provided in the Bases.	**Specification 3.8.1** NUREG Bases markup p B 3.8-20 Retyped ITS Bases p B 3.8-19 and B 3.8-20
F-RAI 3.8.1-09	The change agreed to by JAFNPP during a meeting with the NRC concerning F-RAI 3.8.1-09 has been made. Specifically, the largest load reject test will be performed with both EDGs in the EDG subsystem in parallel.	**Specification 3.8.1** NUREG ITS markup p 3.8-9 NUREG Bases markup p B 3.8-20 and B 3.8-21 Retyped ITS p 3.8-7 Retyped ITS Bases p B 3.8-20 and B 3.8-21
F-RAI 3.8.1-10	The change agreed to by JAFNPP during a meeting with the NRC concerning F-RAI 3.8.1-10 has been made. Specifically, a statement has been added to the Bases describing what is inoperable if the north and south bus disconnect is inoperable and that the automatic open feature is periodically tested in accordance with plant procedures.	**Specification 3.8.1** NUREG Bases markup p Insert page B 3.8-4 Retyped ITS Bases p B 3.8-5
Editorial change	The entity "New York Power Pool" does not exist anymore. Niagara Mohawk now controls the 115kV equipment in the switchyard. Therefore, changes have been made to SR 3.8.1.1 Bases to reflect this change.	**Specification 3.8.1** NUREG Bases markup p Insert page B 3.8-16 Retyped ITS p B 3.8-16

SUMMARY OF CHANGES TO ITS SECTION 3.8 - REVISION I

Source of Change	Summary of Change	Affected Pages
ypographical errors	Minor typographical errors in the NUREG Bases markup and retyped ITS Bases have been corrected. (The number "10" has been changed to "11 in both the NUREG Bases markup and retyped ITS Bases for the LCO section and word "itscorrect" has been changed to "its correct" in the retyped SR 3.8.1.1 Bases)	**Specification 3.8.1** NUREG Bases markup p B 3.8-4 Retyped ITS Bases p B 3.8-5 and B 3.8-16
Technical change	The minimum air start receiver pressure at which 5 EDG starts can be accomplished has been changed from 180 psig to 150 psig. The previous value was incorrect, since plant data demonstrates 5 starts can be accomplished starting as low as 150 psig in the associated air start receiver. Accordingly, the air start pressure at which one EDG start can be accomplished has been changed from 150 psig to 110 psig. In addition, these new values are consistent with UFSAR Table 16.3-6.	**Specification 3.8.3** CTS markup p 5 of 6 DOCs M4 and L3 (DOCs p 3 of 5, 4 of 5, and 5 of 5) NSHC L3 (NSHCs p 5 of 6 and 6 of 6) NUREG ITS markup p 3.8-22 and 3.8-23 JFD X3 (JFDs p 2 of 2) NUREG Bases markup p B 3.8-45 Retyped ITS p 3.8-17 and 3.8-18 Retyped ITS Bases p B 3.8-38
RAI 3.8.3-01	The change agreed to by JAFNPP during a meeting with the NRC concerning F-RAI 3.8.3-01 has been made. Specifically, the JFD for describing why the JAFNPP ITS submittal is not including the 10 year storage tank test has been modified to reflect that not including the SR is consistent with the CTS.	**Specification 3.8.3** JFD TA1 (JFDs p 1 of 2)
Technical change	The proper ASTM Standard for performing the particulate concentration test has been included in the ITS. Specifically, the old standard "ASTM D5452-1998" has been replaced with "ASTM D6217-1998." In addition, the actual ASTM standard has been deleted from the Fuel Oil Testing Program since it is included in the Bases of ITS 3.8.3. Also, the Bases of SR 3.8.3.3 has been modified to be consistent with the Program exception concerning filter size. (Note - This item is also described in the Section 5.0 Summary)	**Specification 3.8.3** NUREG Bases markup p B 3.8-47 and Insert page B 3.8-49 Bases JFD DB3 (Bases JFDs p 1 of 2) Retyped ITS Bases p B 3.8-41 and B 3.8-43 **Specification 5.5** NUREG ITS markup p 5.0-15 and Insert page 5.0-15 JFD X7 (JFDs p 5 of 5) Retyped ITS p 5.0-17
Typographical errors	Minor typographical errors in the Discussion of Changes have been corrected. (DOC A4, deleted "and CTS 3.9.F.3, LPCI MOV Independent Power Supplies," and changed "requirements" to "requirement" in the first sentence, and changed "these Conditions" to "this Condition" and "are" to "is" in the second sentence.)	**Specification 3.8.4** DOC A4 (DOCs p 1 of 8)

SUMMARY OF CHANGES TO ITS SECTION 3.8 - REVISION I

Source of Change	Summary of Change	Affected Pages
Typographical error	Minor typographical error in the Discussion of Changes has been corrected. (DOC A4, "CTS 3.9.E station" changed to "CTS 3.9.E and 3.9.F.")	**Specification 3.8.6** DOC A4 (DOCs p 1 of 5)
Typographical error	Minor typographical error in the Discussion of Changes has been corrected. (DOC A3, "CTS 3.9.F" changed to "CTS 3.9.E."	**Specification 3.8.7** DOC A3 (DOCs p 1 of 4)
Editorial correction	The transformer numbers in the Background section of the Bases have been changed from "T2", "T3", and "T4" to "71T-2", "71T-3", and "71T-4", respectively, to be consistent with plant terminology and ITS 3.8.1 terminology. Also, the AC safety bus numbers in Table B 3.8.7-1 have been changed from "71H05", "71H06", "71L15", "71L25", "71L16", and "71L26" to "10500", "10600", "11500", "12500", "11600", and "12600", respectively, to be consistent with the CTS.	**Specification 3.8.7** NUREG Bases markup p B 3.8-79 and Insert page B 3.8-88 Retyped ITS Bases p B 3.8-65 and B 3.8-73

Specification 3.8.1

(A1)

3.9 (cont'd)

[LCO 3.8.1] B. ~~Emergency A-C Power System~~ AC Power Sources-Operating

The availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3, and 3.9.B.4, ~~except when the reactor is in the cold condition~~.

[Applicability] MODES 1, 2, and 3 (M1)

add Required Action A.2 Completion Time (L7)

[Action A] 1. From and after ~~the time that incoming power is available~~ from only one line or through only one reserve station service transformer, continued reactor operation is permissible for a period not to exceed 7* days unless the line or reserve transformer is made operable earlier (A9) provided that during such ~~7~~ days both Emergency Diesel Generator Systems are operable. At the end of the 7*th day, if the condition still exists, the reactor shall be placed ~~in a cold condition within~~ ~~24~~ 36 (L2) hours. in MODE 3 in 12 hours (M3)

[R.A. A.3] (L1) [Action D] [Action G] (A6) [Action F]

and 21 days from discovery of failure to meet LCO (M4)

[Action C] 2. From and after the time that incoming power is not available from any line or through either reserve station transformer, continued reactor operation is permissible for a period not to exceed 7 days, provided that both redundant Emergency Diesel Generator Systems are operable, all core and containment cooling systems are operable and the shutdown cooling systems are operable. At the end of the seventh day, if the condition still exists, the Reactor shall be placed ~~in a cold condition within~~ ~~24~~ 36 (L2) hours. in MODE 3 in 12 hours (M3)

[R.A. C.2] (A6) [Action G] [R.A. C.1] [Action F]

add Required Action C.1 Completion Time (L7)

~~*From September 9, 2001 through September 23, 2001, with 115 kV line #3 and/or one reserve station service transformer inoperable, continued reactor operation under this condition is permissible for a period not to exceed 14 days, provided 115 kV line #4 is operable.~~ (A9)

add Action D (L1)

add Action G (A6)

4.9 (cont'd)

(L8) add Note 1 to SR 3.8.1.3

(L9) add Note 2 to SR 3.8.1.3

(M8) add Note 3 to SR 3.8.1.3

(A3) add Note 4 to SR 3.8.1.3

B. ~~Emergency A-C Power System~~

(M7) add time, voltage, frequency

(A8) add Note to SR 3.8.1.2

[SR 3.8.1.2] 1. Once each month, each pair of diesel generators which forms a redundant Emergency Diesel Generator System shall be manually initiated to demonstrate its ability to start, accelerate, and force parallel; [SR 3.8.1.3] after connection to the bus, the paralleled pair will be loaded to 5,200 KW, this load will be maintained until ~~both generators are at steady state temperature conditions~~. ~~During this period the generators' load sharing capability will be checked.~~ ≥ 1 hour (A4)

each EDG loaded to ≥ 2340 and ≤ 2600 (L3)

~~2. Once per month the diesel starting air compressors shall be checked for proper operation and their ability to recharge air receivers.~~ See ITS 3.8.3

(I)

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Revision I

ADMINISTRATIVE CHANGES

A5 The details of CTS 3.9.C.2 related to the Diesel Fuel Oil Transfer System OPERABILITY have been deleted. The CTS 4.9.C.2 Surveillance Requirement of the Diesel Fuel Oil Transfer System necessary to support EDG OPERABILITY is maintained by ITS SR 3.8.1.6. Additional technical changes to the diesel fuel oil transfer system are discussed in L6, L10, and M13. The applicability of the Diesel Fuel Oil Transfer System is currently and will remain associated with EDG OPERABILITY, therefore this change in presentation is administrative.

A6 CTS 3.9.B does not provide specific Actions for the condition of three or more AC sources inoperable and therefore entry into CTS 3.0.C is required and the plant must be placed in COLD SHUTDOWN within 24 hours. ITS 3.8.1 Condition G has been included which specifies all other possible combinations of inoperable AC sources not addressed in the other proposed conditions in ITS 3.8.1. Since the ITS format allows multiple Conditions to be entered simultaneously, with three or more AC sources inoperable, ACTIONS would be taken in accordance with ITS 3.8.1, and ITS LCO 3.0.3 entry conditions would not be met. However, consistent with the CTS default to CTS 3.0.C, ITS 3.8.1 Required Action G.1 will require direct entry into ITS LCO 3.0.3. The changes in time requirements to shutdown of CTS 3.0.C are addressed in ITS Section 3.0 Discussion of Changes M1 and L1. Therefore this change is considered administrative.

A7 The wording in CTS 4.9.B.5.a, b and c related to the criteria for determining whether a potential common cause EDG failure exists are being replaced with ITS 3.8.1 Required Action B.3.1 which requires the determination whether the other Operable EDG and EDG subsystems are not inoperable due to common cause failure. The allowance to extend the CTS Completion Time of 8 hours to 24 hours is discussed in L5. Since the intent of the CTS and ITS wording are identical, this change is considered administrative. This change is consistent with NUREG-1433, Revision 1.

A8 A Note has been added to CTS 4.9.B.1 (ITS SR 3.8.1.2) and 4.9.B.4 (ITS SRs 3.8.1.9, 10, and 12) which allows all EDG subsystem starts to be preceded by an engine prelube period to minimize wear and tear on the EDGs during testing. The addition of the Note is considered administrative since the EDGs at JAFNPP run in a continuous prelube mode of operation. In addition, the Note to ITS SR 3.8.1.2 also allows a warmup period prior to loading. The addition of this part of the Note is also considered administrative since the EDGs are not immediately loaded upon startup, but are allowed to warmup for a short time after startup while the operations staff performs post startup EDG checks.

ADMINISTRATIVE CHANGES

A9 The CTS provides an allowance in CTS 3.9.B.1 footnote * that the 7 day allowed outage time for a single offsite circuit can be extended to 14 days (for line #3 and/or one reserve station service transformer only) during the period from 9/9/01 through 9/23/01. Since the time period has passed, this allowance is not applicable anymore, and is not included in the ITS.

I

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.9.A requires the LCO to be met prior to making the reactor critical. This effectively implies MODES 1 and 2. ITS 3.8.1 applicability requires AC sources be OPERABLE in MODES 1, 2, and 3. This change expands the Applicability of AC electric power source OPERABILITY requirements to more MODES of operation. The addition of MODE 3 establishes requirements for the OPERABILITY of AC sources consistent with the OPERABILITY requirements for the functions that these sources are required to support including Emergency Core Cooling Systems and Primary Containment Isolation. The addition of MODE 3, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. The proposed Applicability in ITS 3.8.1 is consistent with the Applicability statement in CTS 3.9.B (except when the reactor is in the cold condition). This statement implies that the 3.9.B is applicable in MODES 1, 2 and 3. Since CTS 3.9.B provides exception to the Operability requirements in CTS 3.9.A, this portion of the change is considered administrative since it clearly states when the AC sources are required to be Operable and fixes an inconsistency in the current Specification.

M2 The JAFNPP CTS 3.9.B does not contain certain Surveillance Requirements specified in the Standard Technical Specifications, General Electric Plants, BWR/4, NUREG-1433, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)). JAFNPP has included the following Surveillance Requirements since they directly impact the OPERABILITY of the AC Sources, and should be performed to ensure the accident analysis can be met.

The following 31 day Surveillance Requirements are adopted in the ITS:

SR 3.8.1.4, Verify each day tank contains ≥ 327 gal of fuel oil; and

SR 3.8.1.5, Check for and remove accumulated water from each day tank.

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

The following 24 month Surveillance Requirements are adopted in the ITS:

SR 3.8.1.7, Verify automatic and manual transfer of plant power supply from the normal station service transformer to each offsite circuit. The normal station service transformer, which is not a qualified offsite circuit, is the normal power source for the 4.16 kV emergency buses when the main generator is on line. However, automatic residual transfer ensures that power can be transfered to the qualified source (reserve circuit) if a main generator trip occurs;

SR 3.8.1.8, Verify each EDG subsystem rejects a load greater than or equal to a load equivalent to one core spray pump;

SR 3.8.1.11, Verify each EDG pair operating within the power factor limit (except when grid conditions will not permit, and then as close to the limit as practicable) operates for ≥ 2 hours loaded from 105% to 110% of the continuous rating, and for 6 hours loaded from 90% to 100% of the continuous rating.

SR 3.8.1.13, Verify interval between each sequenced load block is greater than or equal to the minimum design interval.

An evaluation has been performed due to the added testing requirements imposed by SR 3.8.1.7, SR 3.8.1.8, SR 3.8.1.11, and SR 3.8.1.13. The evaluation has concluded that the additional tests will not impact the reliability of the EDG subsystems.

Since no similar Specifications exist, the addition of these SRs imposes additional operational requirements, and is considered more restrictive. This change, consistent with current operating practice, is considered to have no adverse impact on safety.

M3 CTS 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.B.4, and 3.0.E requirements, that the plant be placed in cold shutdown (MODE 4) within 24 hours (L2) if the corresponding AC sources were not restored within the current Completion Times or all of its redundant system(s), subsystem(s), train(s), component(s), or device(s) are OPERABLE, are being added to. ITS 3.8.1 Required Action F.1 requires the plant to be in MODE 3 (Hot Shutdown) within 12 hours. This action will ensure that the plant is placed in a MODE outside of the Applicability in a timely manner. Based on operating experience, the 12 hour Completion Time limit is acceptable

TECHNICAL CHANGES - MORE RESTRICTIVE

M3 (continued)

since it allows sufficient time for an orderly transition to MODE 3 without challenging plant systems. The additional requirement, to be in MODE 3 in 12 hours, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

M4 CTS 3.9.B.1 and 3.9.B.3 Completion Times are being added to. ITS 3.8.1 Required Actions A.3 and B.4 include a second Completion Time of 21 days from the discovery of failure to meet the LCO. This second Completion Time imposes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. This restriction is intended to prevent exceeding the assumptions regarding allowed out of service times for an inoperable AC source as a result of sequential inoperable EDG subsystems and reserve sources. The additional requirement, limiting the maximum continuous allowed out of service time, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

M5 Not Used.

M6 Not Used.

M7 CTS 4.9.B.1 requirement, to demonstrate the ability of each EDG subsystem to start, accelerate, and force parallel, is being supplemented. ITS SR 3.8.1.2 requires the EDG subsystem also meet specific values for time, voltage and frequency. These requirements are acceptable based on meeting the values for time, voltage, and frequency consistent with existing plant design, and regulatory requirements. The addition of the requirement for the EDG subsystems to meet specific values for time, voltage, and frequency, is necessary to ensure EDG OPERABILITY and safety analysis assumptions are maintained, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

M8 CTS 4.9.B.1 requirement, to load each EDG subsystem, is being restricted. ITS SR 3.8.1.3 Note 3 precludes this Surveillance from being performed on more than one EDG subsystem at a time. This change will ensure that at least one EDG subsystem is available to minimize the consequences of a design basis accident. This change is considered acceptable since although not required by CTS it is consistent with

TECHNICAL CHANGES - MORE RESTRICTIVE

M8 (continued)

current JAFNPP practice. The addition of ITS SR 3.8.1.3 Note 3 to permit loading of only one EDG subsystem at a time, is necessary to help avoid common cause failures of both EDG subsystems that might result from offsite circuit or grid perturbations, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

M9 CTS 4.9.B.4 requires, once every 24 months, the conditions under which the EDG system is required (loss of power signal in conjunction with an ECCS initiation signal) will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel and accept the emergency loads in the prescribed sequence. This condition, loss of power signal in conjunction with an ECCS initiation signal is addressed in ITS SR 3.8.1.12 (A2). In addition, JAFNPP has included ITS SR 3.8.1.9 and SR 3.8.1.10 which require, once each 24 months, verification that the EDG subsystem, in response to a loss of power signal or ECCS initiation signal respectively, will auto-start from a standby condition and energize, as required, permanently connected loads within the required time and auto-connected emergency loads in the prescribed sequence. These Surveillance Requirements have been added to verify EDGs respond in accordance with the variations in the applicable design requirements. Addition of these Surveillance Requirements is consistent with the format in NUREG-1433, Revision 1, imposes additional operational requirements, and therefore is considered more restrictive. This change is considered to have no adverse impact on safety.

M10 Not Used.

M11 CTS 4.9.B.4 requirement, to demonstrate the ability under required conditions (loss of power signal, ECCS initiation signal, loss of power signal in conjunction with ECCS signal) of each EDG subsystem to start, accelerate, force parallel and accept emergency loads in the prescribed sequence, is being supplemented. ITS SR 3.8.1.9, SR 3.8.1.10, and SR 3.8.1.12 require also that the EDG subsystem meet specific values for time, voltage, frequency and loading duration. These requirements verified by current Surveillances are acceptable based on meeting the values for time, voltage, and frequency consistent with existing plant design and regulatory requirements. The addition of these requirements for the EDG subsystems to meet specific values for time, voltage, frequency and loading duration is necessary to ensure EDG OPERABILITY is maintained, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

TECHNICAL CHANGES - MORE RESTRICTIVE

M12 CTS 4.9.B.6 requires that, in the event an EDG subsystem or reserve source is inoperable, the availability of reserve power be assured by verification of correct breaker alignment and that the associated reserve electrical line is energized, once within one hour and every 8 hours thereafter. There is no CTS requirement to verify correct breaker alignment and indicated power availability for each reserve circuit on a regular basis during MODES 1,2 and 3 if the EDG subsystems reserve sources are available. ITS SR 3.8.1.1, a new surveillance, has been included to verify the offsite circuits breaker alignment and power availability once every 7 days. This Frequency is considered adequate since alarms and annunciators exist which will identify problems with the offsite circuits. The addition of the SR 3.8.1.1 to verify correct breaker alignment and indicated power availability for each offsite circuit every 7 days, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

M13 CTS 3.9.C.2.b, 3.9.C.2.c and CTS 4.9.C.2 do not specify which EDG fuel oil transfer pumps must be OPERABLE. ITS SR 3.8.1.6 will require each diesel generator to have at least one transfer pump capable to transfer oil from its corresponding storage tank to day tank automatically. If this requirement is not met the associated diesel generator must be declared inoperable. The allowance in CTS 3.9.C.2.b to repair the transfer pumps in 30 days is deleted and the allowed outage time for the associated EDG applies. This allowance is 14 days for an EDG subsystem therefore this change is more restrictive on plant operation. In addition CTS 3.9.C.2.c has been deleted since the association between the fuel oil transfer system will now correspond to the associated EDG and not the EDG subsystem. The specification of required transfer pumps for each EDG and the elimination or reduction of time to restore the pumps to OPERABLE consistent with EDG OPERABILITY, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details of OPERABILITY in CTS 3.9.A.1, that "power is available to the emergency buses from the following power sources", and specific design details in CTS 3.9.A.1.a, referring to the "115 kV lines and reserve station transformers", are proposed to be relocated to the Bases. These details for system OPERABILITY and design are not necessary in the LCO. ITS 3.8.1 LCO which requires two offsite circuits between the offsite transmission network and the plant class 1E AC

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

Electrical Power Distribution System and the associated ACTIONS and Surveillance Requirements and the definition of OPERABILITY are sufficient to ensure the OPERABILITY of the AC Sources - Operating. Therefore, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program in Chapter 5 of the Technical Specifications.

LA2 The CTS 4.9.B.3 requirement to check the EDG subsystem instrumentation during the monthly generator test is proposed to be relocated to the Technical Requirements Manual. The requirement does not specify which instrumentation is required to be checked. Based on the definition of OPERABILITY all instrumentation necessary for the EDGs to operate must be OPERABLE. The definition of OPERABILITY ensures that all instrumentation not specified in the Technical Specifications and required for proper diesel generator operation are OPERABLE. As a result, this relocated requirement is not necessary to be included in the ITS to provide adequate protection of the public health and safety. At ITS implementation, the relocated requirement will be incorporated by reference into the UFSAR. Changes to the relocated requirements in the Technical Requirements Manual will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.9.B does not provide a specific Action for the condition of one inoperable offsite circuit and one inoperable EDG subsystem. Therefore, if this Condition existed the plant would be required to enter LCO 3.0.C and place the plant in cold shutdown in 24 hours. Therefore, ITS 3.8.1 ACTION D, and associated Required Actions and Completion Times, for one offsite circuit and one EDG subsystem inoperable, have been added, since in this Condition either the OPERABLE offsite circuit or OPERABLE EDGs are more than sufficient to bring the plant to a safe shutdown condition. ITS 3.8.1 Required Action D.1 is to restore the offsite circuit to OPERABLE status within 12 hours or Required Action D.2 is to restore the EDG subsystem to OPERABLE status within a Completion Time of 12 hours. The 12 hour Completion Time is based on the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period. In addition, if the Completion Times are not satisfied entry into ITS 3.8.1 Required Action F.1, MODE 3 in 12 hours(M3) and Required Action F.2, MODE 4 in 36 hours (L2), will be required instead of ITS LCO 3.0.3 since the plant is not outside its Design Bases.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

The addition of ITS 3.8.1 Condition D is considered less restrictive since it allows an additional 12 hours to restore the inoperable sources. However, it is considered acceptable since it reduces the risk of subjecting the plant to a shutdown and since other equipment will still be available to support any abnormal operational transient or a design bases accident. This change is consistent with NUREG-1433, Revision 1, and is appropriate, based on the design of the JAFNPP AC Sources.

L2 CTS 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.B.4, and 3.0.E requirement, that the plant to be placed in cold shutdown (MODE 4) within 24 hours is being relaxed. ITS 3.8.1 Required Action F.2, extends the time allowed for the plant to be in MODE 4 from 24 to 36 hours. This change is in association with the addition of a new interim requirement, ITS 3.8.1 Required Action F.1, which requires the plant to be in MODE 3 in 12 hours (M3). The 36 hour Completion Time is based on providing the necessary time for the plant to cool down and reduce pressure in a controlled and orderly manner, and the low probability of a DBA occurring during this period. The additional time to reach MODE 4 (36 hours) in association with the interim requirement to be in MODE 3 (12 hours), is consistent with NUREG-1433, Revision 1, reduces the potential for a plant event that could challenge plant safety systems, relaxes operational time requirements, and therefore is considered to be less restrictive.

L3 CTS 4.9.B.1 requirement, that each EDG subsystem be force paralleled and loaded to 5200 KW, and load sharing capability checked, are being relaxed. ITS SR 3.8.1.3 requires each EDG subsystem be paralleled with normal, reserve, or backfeed power and each EDG be loaded and operated for ≥ 60 minutes at a load ≥ 2340 KW and ≤ 2600 KW. Maintaining the load of each EDG within the specified limits ensures the load sharing capability is demonstrated, since, if the load sharing capability was not functioning properly, it would be difficult, if not impossible, to maintain each EDG within the required load range. Therefore, the explicit words to check the load sharing capability are not necessary. The EDG subsystem load requirements have also been relaxed to ensure that the EDGs continuous rating is not required to be exceeded every 31 days. The new load range requirement of 90%-100% of continuous rating is consistent with the recommendations of Regulatory Guide 1.9, Revision 3. This change is acceptable since the EDG full load carrying capability will be demonstrated during the 8 hour load test in ITS SR 3.8.1.11.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CTS 4.9.B.4 stipulates that an automatic starting test, under the conditions which the EDG is required to perform will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence. ITS SRs 3.8.1.9, 3.8.1.10, and 3.8.1.12, include the phrase "actual or," in reference to the automatic initiation signal, for verifying that each EDG starts on an automatic initiation signal. This change is acceptable since the EDGs cannot discriminate between "actual" or "simulated" signals and OPERABILITY is adequately demonstrated in either case. This change allows satisfactory automatic system initiations to be used to fulfill the Surveillance Requirements, is consistent with the NUREG-1433, Revision 1, allows alternative operational requirements, and therefore is less restrictive.

L5 CTS 4.9.B.5 provides a Completion Time of 8 hours when an EDG subsystem is inoperable to demonstrate, by manual start and force paralleling, the availability of the OPERABLE EDG subsystem or determine that the inoperability is due to preplanned maintenance or testing, an inoperable support system with no potential common mode failure, or an independently testable component with no potential common mode failure. ITS 3.8.1 Required Action B.3.1 provides a Completion Time of 24 hours to determine the OPERABLE EDG subsystem is not inoperable due to common cause failure or ITS 3.8.1 Required Action B.3.2, perform SR 3.8.1.2 for OPERABLE EDG subsystem. This change provides an additional 16 hours to determine that the inoperabilities are not due to common cause failure or to demonstrate OPERABLE EDG subsystem availability. The 24 hour Completion Time, in accordance with Generic Letter 84-15, is reasonable time to confirm that the OPERABLE EDG subsystem is not affected by the same problem as the inoperable EDG subsystem based on the low probability of an event during the additional 16 hours. If the cause of the inoperable EDG subsystem cannot be confirmed not to exist on the OPERABLE EDG subsystem, then performance of ITS SR 3.8.1.2 is required to provide assurance of continued OPERABILITY of the remaining EDG subsystem. This change is consistent with Generic Letter 84-15 and NUREG-1433, Revision 1, relaxes operational requirements, and therefore is considered to be less restrictive.

L6 CTS 3.9.C.2.a allowance, for operation to continue for 60 days with one inoperable fuel oil transfer pump associated with a Diesel Generator subsystem provided the remaining fuel oil transfer pumps are demonstrated OPERABLE immediately and weekly thereafter, has been deleted. ITS SR 3.8.1.6 Bases requires only one pump to be OPERABLE for each EDG. Each individual diesel generator is provided with an independent fuel oil system consisting of a main fuel oil storage tank, a day tank and two full-capacity motor-driven pumps which are designed to transfer fuel oil from the storage tank to the associated day tank automatically. The selected pump will start automatically when the day

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L6 (continued)

tank level is < 70 % and the standby pump will start when the day tank level is < 50%. The fuel oil transfer system design includes provisions so that the transfer pumps can be aligned manually to transfer oil from any storage tank to the day tank within each Diesel Generator subsystem. Since one transfer pump is capable of maintaining adequate level in the day tank with the diesel generator operating at full load, allowing continuous operation with one OPERABLE transfer pump is acceptable.

L7 CTS 3.9.B.1 allows operation to continue for 7 days with one reserve power source inoperable provided both Emergency Diesel Generator subsystems are OPERABLE. CTS 3.9.B.2 allows operation to continue for 7 days with two reserve power sources inoperable provided both Emergency Diesel Generator subsystems, all core and containment cooling systems as well as the shutdown cooling systems are OPERABLE. CTS 3.9.B.3 allows operation to continue for 14 days with one EDG subsystem inoperable provided that the two reserve power sources are available and the remaining EDG subsystem is OPERABLE. CTS 3.0.E requires the plant to be in cold shutdown within 24 hours when a redundant feature is inoperable and the reserve or onsite power source of a system, subsystem, train, component or device is inoperable.

ITS 3.8.1 includes all of the requirements for redundant system cross checks when AC sources are inoperable. ITS 3.8.1, Required Action A.2 allows 24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s) to declare required feature(s) with no offsite power available inoperable when the required feature(s) are inoperable. ITS 3.8.1 Required Action B.2 allows 4 hours from discovery of one inoperable EDG subsystem concurrent with inoperability of redundant required feature(s) to declare the required features supported by the inoperable EDG subsystem inoperable when the redundant required feature(s) are inoperable. ITS 3.8.1 Required Action C.1 allows 12 hours from discovery of two offsite circuits inoperable concurrent with inoperability of redundant required feature(s) to declare required feature(s) inoperable when the redundant required feature(s) are inoperable.

Thus, adding the new ITS 3.8.1 Required Actions A.2, B.2, and C.1 will allow the operator time to evaluate and correct any discovered inoperabilities. This will reduce the risk of subjecting the plant to a shutdown while the redundant safety features are inoperable. This change is acceptable because the remaining OPERABLE power supplies are adequate to supply electrical power to the onsite emergency distribution system. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, the safety function has not been lost. The 4 hour, 12 hour, and 24 hour Completion

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L7 (continued)

Times take into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. By allowing featuresassociated with the inoperable offsite circuit(s) to be declared inoperable, the appropriate ACTIONS can be taken. Additionally, the Completion Times take into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and a low probability of a DBA occurring during this period. Therefore, this change is considered acceptable.

L8 A Note is added to CTS 4.9.B.1 requirement for loading the emergency diesel generators (EDGs). ITS SR 3.8.1.3 Note 1, allows gradual loading of the EDGs as recommended by the manufacturer. EDG loading should be gradual whenever possible to minimize mechanical stress and wear on the diesel engine. This change is considered to be acceptable since the starting, loading, subsequent full load operation, and automatic start and loading testing required by other Technical Specification Surveillances is adequate to confirm the EDG's capability.

L9 Note is added to CTS 4.9.B.1 requirement for loading the emergency diesel generators (EDGs). ITS SR 3.8.1.3 Note 2, allows that momentary load transients outside of the load range do not invalidate performance of the SR. Momentary transients may occur for various reasons during loading, unloading, and steady state operation of the EDG. However, these transients are quickly restored to within the limits and do not reflect an inability of the EDG to fulfill its function. Therefore, these transients should not be considered as a failure of the Surveillances.

TECHNICAL CHANGES - RELOCATIONS

None

PA1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.1.1 [4.9.B.6] Verify correct breaker alignment and indicated power availability for each DB7 [required] offsite circuit.	7 days
SR 3.8.1.2 [4.9.B.1] -------------------NOTES------------------- 1. ~~Performance of SR 3.8.1.7 satisfies this SR.~~ DB5 2. All ~~DG~~ E subsystem starts ~~may~~ be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. ~~A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met.~~ DB5 --- [4.9.B.1] [M7] DB1 Verify each ~~DG~~ E subsystem starts from standby conditions, force parallels, and achieves steady state voltage ≥ ~~[3740]~~ 3900 V and ≤ ~~[4580]~~ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz. ; a. in ≤10 seconds DB5 voltage ≥ 3900V and frequency ≥ 58.8Hz; and TA2 b.	31 days CLB5 ~~As specified in Table 3.8.1-1~~ X2

1 I

(continued)

Revision I

PA1

SURVEILLANCE REQUIREMENTS (continued)

DB5

[M2]

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9 8 X7	CLB9 TA1 -----------------------NOTES----------------------- 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with the E DG synchronized paralleled with offsite normal, reserve or backfeed power, it shall be performed at a power factor ≤ [0.9]. -- Verify each E DG subsystem rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ [65.5] 66.75 Hz; b. Within [3] seconds following load rejection, the voltage is ≥ [3740] V and ≤ [4580] V; and c. Within [6] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz. CLB4	CLB3 within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable [18 months] X7 24
~~SR 3.8.1.10~~	~~-----------------------NOTE-----------------------~~ ~~This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR.~~ ~~--~~ ~~Verify each DG operating at a power factor ≤ [0.9] does not trip and voltage is maintained ≤ [4800] V during and following a load rejection of ≥ [1710] kW and ≤ [2000] kW.~~	CLB2 ~~[18 months]~~

(continued)

I

Revision C I

11/22/98

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.8.1 - AC SOURCES - OPERATING

PLANT SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB4 Words have been changed to properly reflect the JAFNPP fuel oil transfer system. Each EDG is considered to have its own dedicated system even though the systems within an EDG subsystem can be manually tied together. In addition, the EDGs do not have an engine mounted tank.

DB5 Note 1 and Note 3 of ISTS SR 3.8.1.2 have been deleted since the idling and gradual acceleration feature is not included in the JAFNPP EDG design. ISTS SR 3.8.1.7 has been deleted since the quick start (i.e., starting and accelerating to rated speed and voltage within a specific time period) will be performed every 31 days, however, the ISTS SR 3.8.1.7 requirement to reach rated speed and voltage in ≤ 10 seconds has been added to ITS SR 3.8.1.2. Subsequent surveillances have been renumbered as applicable. | △I

DB6 Not Used.

DB7 ITS 3.8.1 has been revised to reflect the specific AC sources design at JAFNPP, in that JAFNPP has two offsite (reserve circuit) sources and 4 emergency diesel generators (EDGs) in two EDG subsystems all of which are required to be OPERABLE. As such, the term required has been deleted to reflect the OPERABILITY requirement of all available AC sources.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 8, Revision 2, have been incorporated into the revised Improved Technical Specifications. TSTF-8, Revision 2, revises NUREG-1433 SRs 3.8.1.8 Note, 3.8.1.9 Note 1, 3.8.1.11 Note 2, 3.8.1.12 Note 2, 3.8.1.14 Note 2, and 3.8.1.19 Note 2 by deleting the statement allowing credit to be taken for unplanned events that satisfy this SR. The removal of this statement is consistent with additional changes to the Bases for SR 3.0.1 which clarify that credit may be taken for unplanned events to satisfy any SR, not just those in Section 3.8.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler 163, Revision 2, have been incorporated into revised Improved Technical Specifications. TSTF-163, Revision 2, revises EDG SR start acceptance criteria to specify only

PA3

BASES

LCO (continued)

~~[In addition, [one required automatic load sequencer per ESF bus] shall be OPERABLE.]~~ PB6

qualified

emergency

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ~~ESF~~ buses. ~~Each offsite circuit consists of incoming breaker and disconnect to the respective 2C and 2D SATs, the 2C and 2D transformers, and the respective circuit path including feeder breakers to 4.16 kV ESF buses. Feeder breakers from each circuit are required to the 2E ESF bus; however, if 2C SAT is connected to ESF bus 2E (or 2G) and 2D SAT is connected to 2G (or 2E), the remaining breakers to 2E and 2G are not required.~~

Insert B 3.8.1 LCO-1

I

E

subsystem

force paralleling

(11)

E

subsystem

emergency

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ~~ESF~~ bus on detection of bus undervoltage. This sequence must be accomplished within ~~12~~ seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ~~ESF~~ buses. These capabilities are required to be met ~~from a variety of initial conditions, such as DG in standby with the engine hot and DG~~ in standby with the engine at ambient condition. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of ~~the DG~~ to ~~revert to standby status on an ECCS signal while operating in parallel test mode.~~

with the EDGs

s

E

each EDG

reject a load greater than or equal to the load of a core spray pump

X2

E

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

E

qualified

The AC sources must be separate and independent (to the extent possible) of other AC sources. For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical. ~~A circuit may be connected to more than one ESF bus, with fast transfer capability to the other circuit OPERABLE, and not violate separation criteria.~~ A circuit that is not connected to an ~~ESF~~ bus is required to have OPERABLE ~~fast~~ transfer interlock mechanisms to ~~at least two ESF~~ bus~~es~~ to support OPERABILITY of that circuit.

qualified offsite

emergency

automatic

its associated emergency

(continued)

Revision I

PA3

Insert B 3.8.1 LCO-1

Each qualified offsite circuit consists of the incoming switchyard breakers and and disconnect devices to reserve station service transformer (RSST) 71T-2 or 71T-3, and the respective circuit path including feeder breakers to the 4.16 kV emergency bus 10500 or 10600. In addition, to ensure a fault on one qualified offsite circuit does not adversely impact the other qualified offsite circuit, the 115 kV North and South bus disconnect (10017) automatic opening feature must be OPERABLE if the disconnect is closed. If the automatic opening feature is inoperable, then one of the offsite circuits must be declared inoperable. In addition, due to the unique nature of this design, the automatic opening feature is periodically demonstrated in accordance with plant procedures.

I

BASES AA3

SURVEILLANCE REQUIREMENTS (continued)

minimum and maximum frequencies of the E DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations found in ~~Regulatory~~ Safety Guide ~~1.~~9 (Ref. 3). DB6

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the ~~onsite~~ plant distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that ~~distribution~~ emergency buses and loads offsite can be or are connected to their ~~preferred~~ power source and that appropriate independence of offsite circuits is maintained. Insert B SR 3.8.1.1-A The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

INSERT B SR 3.8.1.1-B

SR 3.8.1.2 ~~and SR 3.8.1.7~~ DB4

Th~~ese~~ is SR~~s~~ help s to ensure the availability of the ~~standby~~ onsite electrical power supply to mitigate DBAs and transients and maintain the ~~unit~~ plant in a safe shutdown condition.

To minimize the wear on moving parts ~~that do not get lubricated when the engine is not running~~, th~~ese~~ is SR~~s~~ ~~have~~ is been modified by a Note ~~(Note 2 for SR 3.8.1.2 and Note 1 for SR 3.8.1.7)~~ to indicate that all E DG starts for th~~ese~~ is Surveillance~~s~~ may be preceded by an engine prelube period and followed by a warmup prior to loading. DB4 E 1 (I)

For the purposes of this testing, the E DGs are started from standby conditions. Standby conditions for a DG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. N

~~[In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.]~~ DB4

(continued)

Revision I

PA3

Insert B SR 3.8.1.1-A

Reserve circuit alignment verification can be accomplished by verifying that an offsite circuit is energized and that the status of offsite circuit supply breakers and disconnects displayed in the control room is correct. Offsite source power availability can be verified by communication with Niagara Mohawk for the Nine Mile Point Unit One switchyard, South Oswego substation, and Light House Hill substation. I

PA3

Insert B SR 3.8.1.1-B

In addition, the Frequency is adequate since administrative controls are in place that require plant notification by Niagara Mohawk of distribution system problems that affect power availability. I

PA3

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

~~part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.~~

CLB10

~~SR 3.8.1.7~~

~~See SR 3.8.1.2.~~

DB4

SR 3.8.1.~~8~~ 7

Transfer of each 4.16 kV ~~ESF~~ emergency bus power supply from the normal ~~offsite circuit~~ to ~~the~~ ~~alternate~~ offsite circuit demonstrates the OPERABILITY of the ~~alternate~~ circuit distribution network to power the shutdown loads. The ~~18~~ 24 month Frequency of the Surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed on the ~~18~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

each

offsite

or backfeed

automatic residual

source (NSST 71T-4)

This SR is only required to be met for each offsite circuit that is not energizing its respective 4.16 kV emergency bus (i.e., the bus is being energized by the NSST)

24

X8

~~This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems.~~ ~~Credit may be taken for unplanned events that satisfy this SR.~~

PA1

Insert B SR 3.8.1.7-A

CLB8

TA1

, Since the automatic transfer must be OPERABLE when the 4.16 kV emergency bus is being supplied by the main generator

SR 3.8.1.~~9~~ 8

DB4

Each E DG is provided with an engine overspeed trip subsystem to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG ~~load response characteristics and~~ capability to reject the largest single load without exceeding predetermined ~~voltage and~~ frequency and while maintaining a specified margin to the overspeed trip. The largest single load for each E DG is a ~~residual heat removal service water~~ core spray pump (12~~25~~ 50 bhp). This subsystem Surveillance may be accomplished by:

a

CLB4

I

(continued)

I

Revision ~~0~~ I

PA3

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~9~~ 8 DB4 (continued)

E 5 E Subsystem I

a. Tripping the ~~DG~~ output breaker with the ~~DG~~ carrying greater than or equal to its associated single largest post-accident load while paralleled ~~to offsite~~ with normal, reserve or backfeed power, or while solely supplying the bus; or

E Subsystem I

b. Tripping ~~its~~ associated single largest post-accident load with the ~~DG~~ solely supplying the bus.

DB8

~~As required by IEEE-308 (Ref. 14)~~ Consistent with Safety Guide 9 (Ref. 3), the load rejection test is acceptable if the ~~increase in~~ diesel speed does not exceed the nominal (synchronous) speed plus 75% of the difference between ~~synchronous~~ nominal speed and the overspeed trip setpoint, or ~~15% above synchronous~~ 115% of nominal speed, whichever is lower. ~~For DGs 2A, 2C, and 1B, this represents 65.5 Hz, equivalent to 75% of the difference between nominal speed and the overspeed trip setpoint.~~

PA1

CLB4

~~The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The [6] seconds specified is equal to 60% of the 10 second load sequence interval associated with sequencing the residual heat removal (RHR) pumps during an undervoltage on the bus concurrent with a LOCA. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection.~~ The ~~[18 month]~~ Frequency of 24 months ~~is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).~~ takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

X2

a PA1 CLB8

This SR is modified by ~~two~~ a Note~~s~~. ~~The reason for Note 1 is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems.~~ TA1 ~~Credit may be taken for unplanned events that satisfy this SR.~~ In order to ensure that the ~~DG~~ subsystem is tested under load conditions that are as close to design basis conditions as possible, the Note ~~2~~ requires that, if ~~synchronized to offsite~~ Paralleled with normal, reserve or backfeed power, testing must be performed using a power factor ≤ ~~[~~0.9~~]~~. This power factor is chosen to be representative of the actual design basis inductive loading that the ~~DG~~ subsystem would experience. However, if the grid conditions do not permit, the power factor limit is not required to be met. In this condition the test is performed with a power factor as close to the design rating of the machine as practicable. This is permitted since, with a high grid voltage it may not be possible to raise the EDG output voltage sufficiently to obtain the required power factor without creating an overvoltage condition on the emergency bus.

E E I X-4 I I

(continued)

Revision ~~G~~ I

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	------------------NOTE------------------ All EDG subsystem starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. -- Verify each EDG subsystem starts from standby conditions, force parallels, and achieves: a. in ≤ 10 seconds voltage ≥ 3900 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 3900 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

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(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	-----------------NOTE------------------- Only required to be met for each offsite circuit that is not energizing its respective 4.16 kV emergency bus. --- Verify automatic and manual transfer of plant power supply from the normal station service transformer to each offsite circuit.	24 months
SR 3.8.1.8	-----------------NOTE------------------- If performed with the EDG subsystem paralleled with normal, reserve or backfeed power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each EDG subsystem rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 66.75 Hz.	24 months

I

LCO

Two qualified circuits between the offsite transmission network and the plant Class 1E Distribution System and two separate and independent EDG subsystems each consisting of two EDGs ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated DBA.

Qualified offsite circuits are those that are described in the UFSAR, and are part of the licensing basis for the plant.

Each qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the emergency buses. Each qualified offsite circuit consists of the incoming switchyard breakers and disconnect devices to reserve station service transformer (RSST) 71T-2 or 71T-3, and the respective circuit path including feeder breakers to the 4.16 kV emergency bus 10500 or 10600. In addition, to ensure a fault on one qualified offsite circuit does not adversely impact the other qualified offsite circuit, the 115 kV North and South bus disconnect (10017) automatic opening feature must be OPERABLE if the disconnect is closed. If the automatic opening feature is inoperable, then one of the offsite circuits must be declared inoperable. In addition, due to the unique nature of this design, the automatic opening feature is periodically demonstrated in accordance with plant procedures. I

Each EDG subsystem must be capable of starting, accelerating to rated speed and voltage, force paralleling and connecting to its respective emergency bus on detection of bus undervoltage. This sequence must be accomplished within 11 seconds. Each EDG subsystem must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the emergency buses. These capabilities are required to be met with the EDGs in standby with the engines at ambient conditions. Additional EDG capabilities must be demonstrated to meet required Surveillances, e.g., capability of each EDG to reject a load greater than or equal to the load of a core spray pump. Proper sequencing of loads, including tripping of nonessential loads, is a required function for EDG OPERABILITY. I

The AC sources must be separate and independent (to the extent possible) of other AC sources. For the EDGs, the separation and independence are complete. For the offsite

(continued)

BASES

LCO (continued)

AC sources, the separation and independence are to the extent practical. A qualified offsite circuit that is not connected to an emergency bus is required to have OPERABLE automatic transfer interlock mechanisms to its associated emergency bus to support OPERABILITY of that circuit.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormal operational transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 4 and 5 are covered in LCO 3.8.2, "AC Sources – Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the availability of the remaining offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second offsite circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the division cannot be powered from an offsite source, is intended to provide assurance that an event with a coincident single active failure of the associated EDG subsystem does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has no power from an offsite circuit.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

simulated accident conditions). The SRs for demonstrating the OPERABILITY of the EDG subsystems are in general conformance with the recommendations of Safety Guide 9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10). Where the SRs discussed herein specify steady state voltage and frequency tolerances, the following summary is applicable. The minimum steady state output voltage of 3900 V is approximately 94% of the nominal 4160 V output voltage. This value, which is slightly greater than that specified in ANSI C84.1 (Ref. 11), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 4400 V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the EDG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations found in Safety Guide 9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the plant distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that emergency buses and loads can be or are connected to their offsite power source and that appropriate independence of offsite circuits is maintained. Offsite circuit alignment verification can be accomplished by verifying that a offsite circuit bus is energized and that the status of offsite circuit supply breakers and disconnects displayed in the control room is correct. Offsite source power availability can be verified by communication with Niagara Mohawk for the Nine Mile Point Unit One switchyard, South Oswego substation, and Light House Hill substation. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room. In addition, the Frequency is adequate since administrative controls are in place that require plant notification by Niagara Mohawk of distribution system problems that affect power availability.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2

This SR helps to ensure the availability of the onsite electrical power supply to mitigate DBAs and transients and maintain the plant in a safe shutdown condition.

To minimize the wear on moving parts, this SR has been modified by a Note to indicate that all EDG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup prior to loading.

For the purposes of this testing, the EDGs are started from standby conditions. Standby conditions for an EDG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

This SR requires that, at a 31 day Frequency, the EDG subsystem starts from standby conditions, force parallels, and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions in the design basis LOCA analysis of UFSAR, Section 6.5 (Ref. 12).

In addition to the SR requirements, the time for the EDG to reach steady state operation is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.108 (Ref. 9). This Frequency provides adequate assurance of EDG subsystem OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This SR verifies that the EDG subsystems are capable of synchronizing and accepting greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the EDG subsystem is paralleled with the normal, reserve or backfeed power source.

Although no power factor requirements are established by this SR, the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while 1.0 is an operational

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 31 day Frequency for this Surveillance is consistent with Safety Guide 9 (Ref. 3).

Note 1 modifies this SR to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this SR by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this SR should be conducted on only one EDG subsystem at a time in order to avoid common cause failures that might result from normal, reserve or backfeed power source perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful EDG subsystem start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which the low level alarm is annunciated. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1.8 hours of EDG operation at full load.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and plant operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.5 (continued)

fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequency is consistent with Regulatory Guide 1.137 (Ref. 10). This SR is for preventive maintenance. The presence of water does not necessarily represent a failure of this SR provided that accumulated water is removed during performance of this Surveillance.

SR 3.8.1.6

This SR demonstrates that at least one fuel oil transfer pump associated with each OPERABLE EDG operates and automatically transfers fuel oil from its associated storage tank to its associated day tank. It is required to support continuous operation of onsite power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE for each EDG.

The Frequency for this SR is consistent with the Frequency for testing the EDG subsystem in SR 3.8.1.3. EDG operation for SR 3.8.1.3 is normally long enough that fuel oil level in the day tank will be reduced to the point where the fuel oil transfer pump automatically starts to restore fuel oil level in the day tank.

SR 3.8.1.7

Automatic residual transfer of each 4.16 kV emergency bus power supply from the normal or backfeed source (NSST 71T-4) to each offsite circuit demonstrates the OPERABILITY of the offsite circuit distribution network to power the shutdown loads. As Noted, the SR is only required to be met for each offsite circuit that is not energizing its respective 4.16 kV emergency bus (i.e., the bus is being energized by the NSST), since the automatic transfer must be OPERABLE when the 4.16 kV emergency bus is being supplied by the main

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.7 (continued)

generator. The 24 month Frequency of the Surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed on the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

In lieu of an actual automatic residual transfer, testing that adequately demonstrates the automatic residual transfer capability is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire automatic residual transfer function and emergency bus energization is verified.

SR 3.8.1.8

Each EDG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the EDG subsystem capability to reject the largest single load without exceeding predetermined frequency and while maintaining a specified margin to the overspeed trip. The largest single load for each EDG subsystem is a core spray pump (1250 bhp). This Surveillance may be accomplished by:

a. Tripping the EDG output breakers with the EDG subsystem carrying greater than or equal to its associated single largest post-accident load while paralleled to normal, reserve or backfeed power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the EDG subsystem solely supplying the bus.

Consistent with Safety Guide 9 (Ref. 3), the load rejection test is acceptable if the diesel speed does not exceed the nominal (synchronous) speed plus 75% of the difference between nominal speed and the overspeed trip setpoint, or 115% of nominal speed, whichever is lower.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

The Frequency of 24 months, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. In order to ensure that the EDG subsystem is tested under load conditions that are as close to design basis conditions as possible, the Note requires that, if paralleled with normal, reserve or backfeed power, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the EDG subsystem would experience. However, if the grid conditions do not permit, the power factor limit is not required to be met. In this condition the test is performed with a power factor as close to the design rating of the machine as practicable. This is permitted since, with a high grid voltage it may not be possible to raise the EDG subsystem output voltage sufficiently to obtain the required power factor without creating an overvoltage condition on the emergency bus.

SR 3.8.1.9

Consistent with Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this SR demonstrates the as designed operation of the onsite power sources due to an emergency bus loss of power (LOP) signal. This test verifies all actions required following receipt of the LOP signal, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the EDG subsystem. It further demonstrates the capability of the EDG subsystem to automatically achieve the required voltage and frequency within the specified time.

The EDG auto-start time of 11 seconds is derived from requirements of the accident analysis for responding to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the EDG subsystem loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of the connection and loading of these loads, testing that adequately shows the capability of the EDG subsystem to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is to minimize the wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.10

This SR demonstrates that the EDG subsystem automatically starts, force parallels and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.10.d and SR 3.8.1.10.e ensure that permanently connected loads and emergency loads are energized from the reserve power system on a LOCA signal without a LOP signal.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the loading logic for loading onto reserve power. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

demonstration of the connection and loading of these loads, testing that adequately shows the capability of the EDG subsystem to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

In addition to the SR requirements, the time for the EDG to reach steady state operation is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is to minimize the wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.11

Consistent with IEEE-387 (Ref. 14), Section 7.5.9 and Table 3, this SR requires demonstration that the EDGs can run continuously at full load capability for an interval of not less than 8 hours – 6 hours of which is at a load equivalent to 90-100% of the continuous rating of the EDG, and 2 hours of which is at a load equivalent to 105% to 110% of the continuous duty rating of the EDG. The EDG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to ensure that the EDG subsystem is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the EDG subsystem could experience. A load band is provided to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

avoid routine overloading of the EDG subsystem. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 24 month Frequency is consistent with the recommendations of IEEE-387 (Ref. 14), Section 7.5.9 and Table 3 which takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 2 is provided in recognition that when grid conditions do not permit, the power factor limit is not required to be met. In this condition, the test is performed with a power factor as close to the design rating of the machine as practicable. This is permitted since, with a high grid voltage it may not be possible to raise the EDG output voltage sufficiently to obtain the required power factor without creating an overvoltage condition on the emergency bus.

SR 3.8.1.12

In the event of a DBA coincident with an emergency bus loss of power signal, the EDGs are required to supply the necessary power to Engineered Safeguards systems so that the fuel, RCS, and containment design limits are not exceeded.

This SR demonstrates EDG operation, as discussed in the Bases for SR 3.8.1.9, during an emergency bus LOP signal in conjunction with an ECCS initiation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG subsystem to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 24 months.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

This SR is modified by a Note. The reason for the Note is to minimize the wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.13

Under accident conditions loads are sequentially connected to the bus by the individual time delay relays. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the EDGs due to high motor starting currents. The minimum load sequence time interval ensures that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding emergency safeguards equipment time delays are not violated. There is no upper limit for the load sequence time interval since, for a single load interval (i.e., the time between two load blocks), the capability of the EDG to restore frequency and voltage prior to applying the second load is not negatively affected by a longer than designed load interval, and if there are additional load blocks (i.e., the design includes multiple load intervals), then the lower limit requirements will ensure that sufficient time exists for the EDG to restore frequency and voltage prior to applying the remaining load blocks (i.e., all load intervals must be greater than or equal to the minimum design interval).

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

REFERENCES

1. UFSAR Section 16.6.
2. UFSAR, Chapter 8.
3. Safety Guide 9, Selection Of Diesel Generator Set Capacity For Standby Power Supplies, March 1971.
4. UFSAR, Chapter 6.

(continued)

BASES

REFERENCES (continued)

5. UFSAR, Chapter 14.
6. 10 CFR 50.36(c)(2)(ii).
7. Generic Letter 84-15, Proposed Staff Actions To Improve And Maintain Diesel Generator Reliability, July 1984.
8. Regulatory Guide 1.93, Availability Of Electric Power Sources, December 1974.
9. Regulatory Guide 1.108, Revision 1, Periodic Testing of Diesel Generator Units Used As Onsite Electric Power Systems At Nuclear Power Plants, August 1977.
10. Regulatory Guide 1.137, Revision 1, Fuel-Oil Systems for Standby Diesel Generators, October 1979.
11. ANSI C84.1, Voltage Ratings for Electric Power Systems and Equipment, 1982.
12. UFSAR, Section 6.5.
13. ASME Boiler and Pressure Vessel Code, Section XI.
14. IEEE-387, IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations, 1995.

Specification 3.8.3

A1

See ITS 3.8.1

3.9 (cont'd)

B. Emergency A-C Power System

The availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3, and 3.9.B.4, except when the reactor is in the cold condition:

1. From and after the time that incoming power is available from only one line or through only one reserve station service transformer, continued reactor operation is permissible for a period not to exceed 7* days unless the line or reserve transformer is made operable earlier provided that during such 7* days both Emergency Diesel Generator Systems are operable. At the end of the 7*th day, if the condition still exists, the reactor shall be placed in a cold condition within 24 hours.

2. From and after the time that incoming power is not available from any line or through either reserve station transformer, continued reactor operation is permissible for a period not to exceed 7 days, provided that both redundant Emergency Diesel Generator Systems are operable, all core and containment cooling systems are operable and the shutdown cooling systems are operable. At the end of the seventh day, if the condition still exists, the Reactor shall be placed in a cold condition within 24 hours.

*From September 9, 2001 through September 23, 2001, with 115 kV line #3 and/or one reserve station service transformer inoperable, continued reactor operation under this condition is permissible for a period not to exceed 14 days, provided 115 kV line #4 is operable.

add proposed LCO Action E and F for starting air

M4

4.9 (cont'd)

B. Emergency A-C Power System

1. Once each month, each pair of diesel generators which forms a redundant Emergency Diesel Generator System shall be manually initiated to demonstrate its ability to start, accelerate, and force parallel; after connection to the bus, the paralleled pair will be loaded to 5,200 KW, this load will be maintained until both generators are at steady state temperature conditions. During this period the generators' load sharing capability will be checked.

[SR3.8.3.4]

2. Once per month the diesel starting air compressors shall be checked for proper operation and their ability to recharge air receivers.

M4

≥150 psig

L3

I

I

TECHNICAL CHANGES - MORE RESTRICTIVE

M4 (continued)

capacity) starting air receivers, an EDG support system, is within limits. ITS LCO 3.8.3 requires diesel starting air to be within limits for each EDG required to be OPERABLE. ITS SR 3.8.3.4 establishes and verifies required starting air receiver pressure is ≥ 150 psig (capacity for 5 starts). In addition, ITS 3.8.3 ACTION E, establishes the requirement to restore required starting air receiver pressure to ≥ 150 psig within 48 hours, for one or more EDGs with required starting air receiver pressure < 150 psig and ≥ 110 psig (minimum requirement for 1 start). Also ITS 3.8.3 ACTION F has been added to declare the affected EDG subsystem inoperable if the Required Action and associated Completion Time for ACTION E is not met or if the EDG starting air subsystem is not within limits for any other reason, consistent with the Applicability. Adding the diesel starting air receiver pressure LCO limitation, Surveillance Requirement, and associated ACTION is necessary to ensure EDG subsystem OPERABILITY, is consistent with NUREG-1433, Revision 1, imposes additional operational requirements, and is considered more restrictive. This change is considered to have no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The operational details of CTS 4.9.C, which require that the quantity of diesel fuel available in each storage tank be manually measured once per month and compared to the reading of the local level indicators to ensure the proper operation thereof, is being relocated to the Technical Requirements Manual. The requirements of ITS SR 3.8.3.1 to verify each fuel oil storage tank contains ≥ 32,000 gallons of fuel each 31 days is sufficient to ensure the required fuel is available to support EDG OPERABILITY. Therefore these details are not required to be in the ITS to provide adequate protection of the public health and safety. At ITS implementation, the relocated items will be incorporated by reference into the UFSAR. Changes to the relocated items in the Technical Requirements Manual will be controlled by the provisions of 10 CFR 50.59.

LA2 The details of CTS 4.9.C.1, which lists fuel oil properties are being relocated to the Technical Requirements Manual. The requirements of ITS SR 3.8.3.3 to verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits and Frequency of the Diesel Fuel Oil Testing Program (ITS 5.5.10) is sufficient to ensure the diesel fuel oil is acceptable to support EDG OPERABILITY. Therefore these details are not required to be in the ITS to provide adequate protection of the public health and safety. At ITS

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 (continued)

implementation, the relocated items will be incorporated by reference into the UFSAR. Changes to the relocated items in the Technical Requirements Manual will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.9.C.1 allowance, to continue reactor operation indefinitely, if the affected fuel oil storage tank level is manually measured at least once per day when the fuel oil storage tank level indicator is found to be inoperable for any reason, is not retained in the ITS. This change relaxes requirements, and is less restrictive. The associated CTS 4.9.C Surveillance Requirement, to compare the manually measured fuel oil storage tank level to the reading of the local level indicators to ensure proper local level indication, is being relocated to licensee controlled documents (see LA1), as an operational detail. This change is acceptable based on the fact that proposed ITS SR 3.8.3.1 retains the requirement to verify each oil storage tank contains the minimum amount of fuel required for EDG OPERABILITY each 31 days, consistent with the requirements of NUREG-1433, Revision 1.

L2 CTS 3.9.C.3 allows that, if the available diesel fuel decreases below the required quantity, "as a result of operation of the diesel generators 'to meet Technical Specification requirements', Specification 3.0.C does not apply", and 48 hours are allowed to restore the required diesel fuel supply. This allowance is not retained in the ITS. Because no other actions are specified in the CTS for the condition where the diesel fuel supply is less than required, LCO 3.0.C would be entered. ITS Specification 3.8.3 ACTION A requires the diesel fuel supply be restored to within limits in 48 hours, regardless of the reason it is not within limits. This change relaxes requirements, and is less restrictive. The minimum fuel supply required is sufficient for 7 days of operation of the EDG at continuous rating, and the condition is restricted to fuel oil level reductions that still maintain at least a 6 day supply (M1). This change is acceptable based on the 6 day supply remaining, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event that would require extended operation of the EDGs during this period.

L3 CTS 4.9.B.2 requires the EDG starting air compressor to be checked for operation and its ability to recharge air receivers. ITS SR 3.8.3.4 verifies that pressure in each required air start receiver is ≥ 150

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 (continued)

psig. The requirement to check the EDG air start compressor for operation is unnecessary and is proposed to be deleted. The requirement to verify pressure is ≥ 150 psig in the air receiver is sufficient to ensure proper operation of the EDG starting air compressor and its ability to recharge air receivers since if the EDG starting air compressor was inoperable, it would not be possible to maintain the required pressure in the associated air receiver. In addition, with the air receiver pressure ≥ 150 psig, sufficient air start capacity is available for each EDG without the aid of the EDG air start compressor. If the EDG air compressor could not operate to maintain the required air start receiver pressure, then ITS 3.8.3 ACTION E must be entered, and depending on the air start receiver pressure, the associated EDG subsystem may be required to be declared inoperable immediately and appropriate ACTIONS taken. As a result, the change has no impact on the ability to maintain the associated EDG subsystem Operable.

TECHNICAL SPECIFICATIONS - RELOCATIONS

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

The Licensee has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change proposes to delete the explicit requirement to check the EDG air start compressor for operation and its ability to recharge air receivers. This change does not result in any hardware or operating procedure changes. The EDG air start system is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. The role of the air start system is in supporting the Operability of the associated EDG to mitigate the consequences of accidents. The requirement to verify pressure is ≥ 150 psig in the air receiver is sufficient to ensure proper operation of the EDG starting air compressor and its ability to recharge air receivers since if the EDG starting air compressor was inoperable, it would not be possible to maintain the required pressure in the associated air receiver. In addition, with the air receiver pressure ≥ 150 psig, sufficient air start capacity is available for each EDG without the aid of the EDG air start compressor. If the EDG air compressor could not operate to maintain the required air start receiver pressure, then the ACTIONS of ITS 3.8.3 must be entered, and depending on the air start receiver pressure, the associated EDG subsystem may be required to be declared inoperable immediately and appropriate ACTIONS taken. The change has no impact on the ability to maintain the associated EDG subsystem Operable. As a result, accident consequences are unaffected by the deletion of the explicit requirements for checking the operation of the EDG air start compressor and its ability to recharge air receivers. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This change proposes to delete the explicit requirement to check the EDG air start compressor for operation and its ability to recharge air receivers. Since the EDG air compressor must still be capable of maintaining the associated air receiver pressurized to ≥ 150 psig (as

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L3 CHANGE

2. (continued)

required by ITS SR 3.8.3.4) to maintain EDG Operability, the possibility for a new or different kind of accident is not created. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

The proposed deletion of the requirement to check the EDG air start compressor for operation and its ability to recharge air receivers does not impact any margin of safety. The requirement to verify pressure is ≥ 150 psig in the air receiver is sufficient to ensure proper operation of the EDG starting air compressor and its ability to recharge air receivers since if the EDG starting air compressor was inoperable, it would not be possible to maintain the required pressure in the associated air receiver. In addition, with the air receiver pressure ≥ 150 psig, sufficient air start capacity is available for each EDG without the aid of the EDG air start compressor. Control of the availability of, and necessary compensatory activities, for the EDG air start compressor, are addressed by plant procedures and policies. If the EDG air compressor could not operate to maintain the required air start receiver pressure, then ITS 3.8.3 ACTION E must be entered, and depending on the air start receiver pressure, the associated EDG subsystem may be required to be declared inoperable immediately and appropriate ACTIONS taken. As a result, the change has no impact on the ability to maintain the associated EDG subsystem Operable. Therefore, this change does not involve a significant reduction in a margin of safety.