PSNH

Public Service of New Hampshire

May 29, 1985

SBN-806
T.F. B7.1.1

New Hampshire Yankee Division

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Reference: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444

Subject: Seabrook Station Technical Specification Improvement Program

Dear Sir:

As discussed in our meeting with you on April 9, 1985, we are developing a Seabrook Station Technical Specification Improvement Program. This Program, as described in the enclosure, is an effort to make the Seabrook Station Technical Specifications a more useful document for the control room operators. The Program involves both format changes and technical changes designed to improve the Technical Specifications administratively and enhance their safety impact. The format changes consist of the removal of several tables and programs which are not used by the operators and which are located in other licensee-controlled documents. The technical changes are proposed in order to improve the impact of current Technical Specifications on plant safety while at the same time providing greater flexibility in operating the plant.

All of these changes are proposed within the context of the Standard Technical Specifications, modified to be Seabrook specific. It is acknowledged that further improvements could be made but these would involve additional time in creation of a new Standard and possibly rulemaking. This Program is designed to provide a substantial improvement to Technical Specifications while maintaining the Standard format.

Several detailed supporting analyses referenced in the Program are not complete but will be provided at the time of our submittal of the marked-up Seabrook Technical Specifications. This Program of proposed changes is being sent prior to our Technical Specification submittal in order to not impact your review schedule. Please address any questions or comments to Mr. Warren J. Hall at (603) 474-9574, extension 4046.

Very truly yours,

John DeVincentis

John DeVincentis, Director
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ENCLOSURE

SEABROOK STATION TECHNICAL SPECIFICATION IMPROVEMENT PROGRAM

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SEABROOK STATION TECHNICAL SPECIFICATION IMPROVEMENT PROGRAM (TSIP)

I. Introduction

The Seabrook Station Technical Specification Improvement Program, described herein, is an effort to make the Seabrook Station Technical Specifications a more useful document for the control room operators. The Program involves both format changes and technical changes designed to improve the Technical Specifications administratively and enhance their safety impact. The format changes consist of the removal of several tables and programs which are not used by the operators and which are located in licensee-controlled documents. The technical changes are proposed in order to improve the impact of current Technical Specifications on plant safety while, at the same time, providing greater flexibility in operating the plant.

All of these changes are proposed within the context of the Standard Technical Specifications modified to be Seabrook specific. It is acknowledged that further improvements could be made but these would involve additional time in creation of a new Standard and possibly rulemaking. This Program is designed to provide a substantial improvement to Technical Specifications while maintaining the Standard format.

The basic philosophy used in proposing changes to Technical Specifications is that they should focus on those conditions important to the control room operator. A number of current Technical Specifications involve conditions that are not controlled by the operator nor used by the operators in day-to-day operation. Also, these same Specifications are covered by licensee-controlled procedures and programs, which are subject to NRC inspection. In addition, a number of current Technical Specifications can lead to unnecessary plant shutdowns, test-induced plant transients and challenges to safety systems.

A number of general safety improvements were considered for their applicability to the current Seabrook Technical Specifications. These potential safety improvements are briefly described below and are further referred to in Sections II and III:

1. Action Statements

   a) Action statements may be modified to prevent unnecessary shutdowns for conditions only slightly "out of spec." This would reduce the number of plant transients, unnecessary challenges to safety systems, and excessive equipment cycling.

b) Equipment allowed outage times may be lengthened to provide more likely repair/restoration times. This would reduce the number of unnecessary plant shutdowns and allow repair and testing to be done with more planning.

c) "One hour shutdown" action statements may be extended to allow sufficient time for an orderly plant shutdown. This would potentially reduce the number of plant transients and unnecessary challenges to safety systems.

2. Test/Surveillance Requirements

a) Test and surveillance frequencies may be reduced for equipment which is highly reliable. This would decrease the unnecessary diversion of operator attention.

b) Test frequencies may be optimized. This would assure that the equipment is not out of service too often due to test, does not wear out due to over test, or does not remain undetected in a failed state too long.

c) Test and surveillance frequencies may be reduced for equipment in high radiation areas. This would reduce excessive, unwarranted radiation exposure to plant personnel.

d) Test frequencies may be reduced to reduce the likelihood of an inadvertant, test-induced plant transient. This would reduce the number of unnecessary plant transients and unnecessary challenges to safety systems.

e) The test type may be modified if it has a potential to degrade the equipment or if it requires placing the plant into a less safe configuration to perform a test.

f) Testing required after a failure may be modified to reduce the likelihood of damaging the redundant system. This would ensure that testing provides assurance that the system is available without causing an additional failure.

g) Testing and surveillance frequencies may be modified to more appropriately reflect the importance of the test.

II. Format Changes

The programs and tables listed in Table 1 are proposed to be removed from Technical Specifications and replaced with references to licensee-controlled documents which contain the same information. The bases for these changes include the following:

(1) The programs and tables are not used by control room operators in day-to-day operation.

(2) They are not of immediate safety importance.

(3) They tend to clutter up the Technical Specifications, making them less useable for operators.

(4) They are contained in some other adminstratively controlled licensee program or procedure, which is available for NRC inspection.

These changes are proposed as format changes alone and it is intended that the activities that are indicated within the programs and tables will be continued.

The justifications indentified in Table 1 refer to either Seabrook-specific documents or regulatory documents as applicable.

TABLE 1 FORMAT CHANGES

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
4.2.1.1.a.2	Axial Flux Difference	1	Eliminate this subsection of the Specification	- Needless duplication of effort.
3/4.2.2	Heat Flux Hot Channel Factor - F_Q	1	Rename this Specification "Planar Radial Peaking Factor - F_{XY}"	- The current Specification calls for the LCO on F_Q to be met by performing a surveillance on F_{XY}. This change reflects actual practice and does not involve any technical requirements.
3/4.2.3	RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel	1	Eliminate RCS flow rate conditions	- The current Specification has lost its significance with the elimination of rod bow (R_2). Thus, this Specification becomes just the $F\Delta H$ spec and RCS flow is covered under 3/4.2.5.
3.2.4	Quadrant Power Tilt Ratio	1	Simplify action statement: With QPTR determined to exceed 1.02, within 24 hours and every 7 days thereafter, verify that F_{XY} and $F\Delta H$ are within their limits by performing Surveillance 4.2.2.2 and 4.2.3.2	- QPTR does not have a safety limit, it just indicates that something abnormal is happening in the core that warrants investigation. Thus the action is to verify that the peaking factors are within limits.

TABLE 1 FORMAT CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
4.2.4.2	Quadrant Power Tilt Ratio	1	Confirm indicateu QPTR at least once per 24 hours	- QPTR is not expected to change significantly over 24 hours.
3/4.2.5	DNB Parameters	1	Add RCS flow limits and Surveillance requirements formerly found in 3/4.2.3	- (See 3/4.2.3 Justification)
3.3.3.2	Movable Incore Detectors	-	Action: with less than the required detector thimbles but more than 50%, perform an evaluation of the reduced number of detector thimbles to determine what, if any, increased uncertainty shall be applied to incore measurements.	- Unnecessary shutdown. - Current Specification would require an eventual plant shutdown if the number of operable thimbles drop from 43 to 42.
4.3.3.2	Movable Incore Detectors	-	Eliminate Surveillance	- Current Surveillance has little to do with intent of the Specification; also, the operability section covers detector normalization and plateau determination.

TABLE 1 FORMAT CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
3.3.1	Reactor Trip System Instrumentation	Various	Remove Table 3.3-2 "RTS Instrumentation Response Times"	- Information contained in an I&C Procedure. - Engineer will measure response time and report channel operability to operators. Operators will act according to Table 3.3-1 of Tech Spec 3.3.1. - Response time measurement made only every 18 months.
3.3.2	ESFAS Instrumentation	Various	Remove Table 3.3-5 "Engineered Safety Features Response Times"	- Information contained in an I&C Procedure. - Engineer will measure response time and report channel operability to operators. Operators will act according to Table 3.3-3 of Tech Spe. 3.3.2. - Response time measurement made only every 18 months. - No action required for response times out of spec.
3/4.3.4	Turbine Overspeed Protection	1,2,3	Remove Specification; Surveillance Frequency once per 92 days*	- Information and action statements contained in Maintenance Procedure. - Action is inconsistent with low risk importance of system.

* Technical Change proposed to be made in addition to the format change.

TABLE 1 FORMAT CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
3/4.3.3.8	Loose Parts Detection System	1,2	Remove Specification	- Included in the Surveillance Program under the scope of the Test Control Manual.
3/4.4.5	Steam Generator Inspection	1,2,3,4	Remove Specification	- Action: Operability of S/G's is covered in Tech Specs 3.4.1.1, 3.4.1.2, and 3.4.1.3. - Surveillance: Included in the ISI/IST Program under the scope of the Test Control Manual.
4.4.9.1.2	Reactor Vessel Material Irradiation Surveillance Specimens	All	Remove Specification	- Surveillance: Included in the ISI Program Manual
3/4.4.10	RCS Structural Integrity	All	Remove Specification	- Included in the ISI/IST Program under the scope of the Test Control Manual.
3.6.1.2	Containment Leakage	5	Remove Table 3.6-1	- Information included in the Test Control Manual.
3/4.6.1.6	Containment Structural Integrity	1,2,3,4	Remove Specification	- Surveillance under the scope of the Test Control Manual.
3/4.7.9	Fire Suppression Water System	All	Remove Specification	- Surveillance under the scope of the Test Control Manual and the Fire Protection Manual.

TABLE 1 FORMAT CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
3/4.7.10	Fire Rated Assemblies	All	Remove Specification	- Surveillance under the scope of the Test Control Manual and the Fire Protection Manual.
3/4.7.7	Snubbers	1,2,3,4 5,6	Remove Specification, including Tables 3.7-3 a. and b.	- Included in the ISI/IST Program under the scope of the Test Control Manual. - Engineer will perform visual inspection and functional test of snubbers. If a snubber is inoperable and not repairable in 72 hours, operators will declare attached system inoperable and follow that action statement.
3.8.4.2	Containment Penetration Conductor Overcurrent Protective Devices	1,2,3,4	Remove Table 3.8-1	- Information contained in an Electrical Maintenance Procedure. - Engineer will determine status of the protective devices and inform the operators. The operators will perform the action based on this information.

TABLE 1

FORMAT CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
3.8.4.3	MOV Thermal Overload Devices	Various	Remove Specification, including Table 3.8-2	- This information and action statement is contained in an Electrical Maintenance Procedure. - Engineer will determine status of device and report to operators. Operators will take appropriate action based on valves declared inoperable.
4.10.2.2	Special Text Exceptions - Group Height, Insertion and Power Distribution Limits	1	Eliminate Surveillance	- Current Surveillance requires a full core flux map at least once per 12 hours, which is of little or no value during the Special Test which requires it.
3/4.11.1.1	Liquid Effluents Concentration	All	Remove Table 4.11-1	- Releases are controlled by the ODCM, which is an NRC-approved document based on 10 CFR 50 Appendix I and 10 CFR 20. - Surveillance frequencies are covered by Effluent Surveillance Program.
3/4.11.2.1	Gaseous Effluents Dose Rate	All	Remove Table 4.11-2	- Releases are controlled by the ODCM, which is an NRC-approved document based on 10 CFR 50 Appendix I and 10 CFR 20. - Surveillance frequencies are covered by Effluent Surveillance Program.

TABLE 1 FORMAT CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
3/4.12.1	Radiological Environmental Monitoring Program	All	Remove Specification	- Program is included in the HP procedures, Seabrook Station Environmental Report, and Test Control Manual.
3/4.12.2	Land Use Census	All	Remove Specification	- Program is included in the HP procedures, Seabrook Station Environmental Report, and Test Control Manual.
3/4.12.3	Interlaboratory Comparison Program	All	Remove Specification	- Program is included in the Yankee Atomic Environmental Laboratory Procedures.
6.2.3	Operational Engineering Section	-	Remove all this section except a summary paragraph	- Information will be in FSAR § 13.4.3.
6.5.1	Station Operation Review Committee	-	Remove all this section except a summary paragraph	- Information will be in FSAR § 13.4.1.
6.5.2	Nuclear Safety Audit and Review Committee	-	Remove all this section except a summary paragraph	- Information will be in FSAR § 13.4.2.
6.9	Reporting Requirements	All	Remove Specification	- Included in Station Reporting Manual.

III. Technical Changes

A. General

A number of technical changes to the Technical Specifications are proposed based on risk analysis and on engineering judgment. To justify these changes, the following steps were used and are documented herein:

1) The importance of systems and components addressed by Technical Specifications was determined by calculating risk importance measures and by use of engineering judgment. The risk importance measures were calculated by using methodology developed by Battelle Labs and applying it to the Seabrook Station Probabilistic Safety Assessment (SSPSA). The output of this analysis yielded an assignment of systems into HIGH and LOW categories of relative risk worth.*

2) For systems with a HIGH risk importance worth, the Technical Specifications were evaluated and modified as appropriate to assure that the Specification is optimized with regard to risk.**

3) For systems with a LOW risk importance worth, the Technical Specifications were examined for actions which consume operator time unnecessarily, have potential for causing plant trip, result in undue radiation exposure to plant personnel, and generally are inappropriate to the safety significance of the action. The Technical Specifications were modified where such a change will reduce the complexity of the Specifications, reduce the potential for plant trip, and generally ease the operating restrictions of the Technical Specifications while not drastically affecting the reliability of the system.

The evaluations described above were performed using insights from the SSPSA and the engineering judgment of operators and technical personnel. The SSPSA was used to determine the risk importance of systems in the Technical Specifications. The "importance" results were determined quantitatively; however, the results could have been determined for the most part from a qualitative inspection of the important sequences. The "importance" results are rigorous and do not depend on indvidual numerical calculations in the SSPSA.

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* The detailed calculations of system importance will be submitted along with the submittal of the marked-up Seabrook Technical Specifications. The determination of system importance which is summarized in Section III.B was based on engineering judgment and a qualitative consideration of the SSPSA. The detailed calculations will provide quantitative support for the system importance analysis but is not expected to affect the results.

** The detailed evaluation and optimization of the Technical Specifications for HIGH risk important systems will be submitted along with the submittal of the marked-up Seabrook Technical Specifications.

After determining risk importance, detailed analyses of the HIGH risk important systems were done with regard to Technical Specifications to ensure that the allowed outage times, surveillance/test types and frequencies, and maintenance activities are optimized. These analyses used the system analysis documented in the SSPSA as the basis for sensitivity calculations. The results of this study indicate the important Technical Specification parameters and potential changes that can be made to maintain or improve the reliability of HIGH risk important systems and components.

For the rest of the systems (the so-called LOW risk importance systems), engineering judgment was the primary tool to justify deviations from the Standard Technical Specification.

This assignment of systems into HIGH and LOW risk importance categories permitted the detailed, quantitative risk optimization of Technical Specifications to be targeted to those systems that are the most important with regard to plant risk. Technical changes to other systems were justified based on qualitative, engineering judgment. The results are described in the following sections.

B. System Importance*

The risk importance of systems was determined by calculating risk importance factors and by using engineering judgment to assess the importance of systems to initiating events, containment failure and offsite release, and external events. The results are displayed below. The systems/components of HIGH risk importance are listed with a basis for their inclusion.

The systems not listed below are designated "LOW risk importance" systems. This designation does not imply that these systems are unimportant to plant safety or that they could be eliminated without affecting risk. Also the LOW risk importance systems are not necessarily at the same level of importance. However, for LOW risk importance systems, it is assumed that a change in the related Technical Specifications would have less effect on plant risk than for HIGH risk importance systems. Thus, minor changes to Technical Specifications for LOW risk importance systems can be justified without a detailed risk-sensitivity analysis.

Systems designated "HIGH risk importance" are systems whose failures contribute to the most likely core melt or fission product release sequences. Thus, a system appears on the HIGH importance list because of its critical safety importance and because it appears in a relatively high frequency core melt/release sequence.

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* The detailed calculations of system importance will be submitted along with the marked-up Seabrook Technical Specifications.

The following systems are judged to be "HIGH risk importance systems" with a brief justification for that designation. The details with regard to the basis for including systems in this list will be submitted later.

1. A.C. Electric Power -

   Loss of all A.C. electric power (station blackout) is the most frequent category of core melt sequences. This involves loss of offsite power, failure of the diesels, and failure to recover electric power before core damage.

2. Service Water -

   This is important because of the dependency of diesels on service water. Failure of this system is a contributor to "station blackout" sequences.

3. Primary Component Cooling Water -

   This is important because of the dependency of high pressure injection pumps and RCP seal cooling on PCCW cooling. Failure of this system leads to small LOCA (leakage out the seals) with no makeup injection.

4. Emergency Feedwater -

   This is important because of the importance of transient events, which usually demand EFW to respond. This system is also important in delaying the time to core melt in "station blackout" sequences, which affects likelihood of recovery of emergency power.

5. Containment Isolation -

   This is important to offsite releases because of the analyzed failure modes of containment. If the containment is intact at time of core melt, the most likely failure would be late overpressure failure which occurs one to three days after the accident. For this late failure, settling of the fission products plus complete evacuation contribute to make this release and the resultant health effects relatively small. Thus, to get a significant source term release, the containment must be open at time of core melt, i.e., failure of containment isolation, and more specifically, failure of the purge valves to close if open.

C. High Risk Importance Systems

High Risk importance systems listed in Section III.B have been analyzed with respect to their Technical Specifications to ensure that the allowed outage times, test/surveillance times and types, and maintenance activities are optimized. The detailed analysis and the proposed changes coming out of the analysis will be submitted later along with the marked-up Seabrook Technical Specifications.

D. Low Risk Importance Systems (see Table 2)

The proposed changes to Tech Specs for low risk importance systems are given in Table 2. The justifications for changes listed are based on consideration of the potential safety improvements described in Section I.

TABLE 2 TECHNICAL CHANGES

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
2.1.1	Safety Limits - Reactor Core	1,2	Be in HOT STANDBY within 2 hours	- One hour shu.down greatly increased the likelihood of human error causing a plant trip or other less safe plant conditions.
2.1.2	Safety Limits - RCS pressure	1,2,3,4,5	Be in HOT STANDBY with RCS pressure within its limit within 2 hours	- One hour shutdown greatly increases the likelihood of human error causing a plant trip or other less safe plant conditions.
3.4.1.1	Reactor Coolant Loops and Coolant Circulation	1,2	Be in at least HOT STANDBY within 2 hours	- One hour shutdown greatly increases the likelihood of human error causing a plant trip or other less safe plant conditions.
4.4.1.1	Reactor Coolant Loops	1,2	Eliminate surveillance requirement	- Surveillance not necessary, condition is obvious to operators by alarms and indicators, unnecessary diversion of operators.
3.4.3a	Pressurizer Heaters	1,2,3	With either Groups A or B inoperable, restore inoperable group to Operable status within 7 days or...	- 72 hour allowed outage time too short based on low safety significance of heaters. - Only Groups A and B are used in Safe Shutdown analysis.

TABLE 2 TECHNICAL CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
4.4.3.2	Pressurizer Heaters	1,2,3	Tested once every 18 months	- 92 day surveillance interval too short, heaters don't change drastically in power over 18 months. - Test is time consuming, involves personnel hazard.
3.4.6.2 and 4.4.6.2.2	Reactor Coolant System Pressure Isolation Valves	1,2,3,4	Remove Specification subsections referring to RCS Pressure Isolation Valves, including Action Statement C and Table 3.4-1	- Specification not necessary; RCS pressure isolation valve leakage would be discovered when measuring identified or unidentified leakage and thus would require action if these limits were exceeded. - Testing of these valves will be performed in accordance with the ISI Program.
3.5.1.1a	Accumulators	1,2,3	Restore inoperable accumulator (except as a result of closed isolation valves) to operable within 8 hours	- 1 hour allowed outage time too short to accomplish most likely repairs. - Accumulators are needed in accident analysis for large LOCA which is a relatively unimportant risk contributor.

TABLE 2 TECHNICAL CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
4.5.1.1	Accumulators	1,2,3	24 hour surveillance interval	- 12 hour surveillance too frequent, occupies operator time and attention unnecessarily; operator will respond to alarms.
3.5.2	ECCS Subsystems	1,2,3	Restore inoperable subsystem to operable within 7 days	- 72 hours allowed outage time too short to accomplish most likely repairs, requiring unnecessary shutdowns. - Longer allowed outage times do not affect ECCS system reliability.
4.5.2a	ECCS Subsystems	1,2,3	24 hour surveillance interval for valve alignment	- 12 hour surveillance too frequent, occupies operator time and attention unnecessarily.
3.5.5	RWST	1,2,3,4	2nd Level Specification: if volume or boron concentration is out of spec by no more than 10%, restore to operable within 8 hours. .	- Most likely inoperable conditions are slightly out of spec level or concentration, which have minimal effect on ability of RWST to function, requiring unnecessary shutdowns.

TABLE 2

TECHNICAL CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
3.6.2.1	Containment Building Spray	1,2,3,4	Restore inoperable system to operable within 7 days	- 72 hours allowed outage time too short to accomplish most likely repairs, requiring unnecessary shutdowns. - Longer allowed outage time does not affect ECCS system reliability.
3.6.2.2	Spray Additive System	1,2,3,4	Restore inoperable system to operable within 31 days	- System of very low importance, compared to CBS, action statement not appropriate to risk significance of system outage. - Risk from plant shutdown is much greater than the risk from having SAS inoperable.
4.6.2.2a	Spray Additive System	1,2,3,4	Verify valve alignment at least once per 6 months	- System of very low risk importance. - Valve alignment very unlikely to change in 6 month period.
4.6.4.1	Hydrogen Monitors	1,2	Channel check once per 7 days	- 12 hour channel check is not consistent with importance of Hydrogen Monitors as reflected by the 30 day allowed outage time.

TABLE 2 TECHNICAL CHANGES (cont'd)

TECH SPEC	SYSTEM/COMPONENT	MODE	PROPOSED CHANGE	JUSTIFICATION FOR CHANGE
4.6.4.2a	Hydrogen Recombiners	1,2	System functional test every 18 months	- Unnecessary cycling of equipment.
3.6.5.1	Containment Enclosure Building Integrity	1,2,3,4	Restore integrity within 7 days	- 24 hour allowed outage time is too short, not appropriate to to the low risk significance of enclosure building. - Enclosure building not important in reducing offsite releases from a core damage/melt accident.
3.6.5.3	Containment Enclosure Building Structural Integrity	1,2,3,4	Restore integrity within 7 days	- (See 3.6.5.1 Justification - above)
4.7.6a	Control Room Makeup Air System	All	Eliminate verifying control room temperature is less than 120°F	- Unnecessary diversion of operators, such a condition will become obvious long before the temperature approaches 120°F.