# B 3.7 PLANT SYSTEMS

## B 3.7.1 Steam Generator Pressure / Temperature Limitations

BASES

### BACKGROUND

In order to meet regulatory and code requirements with respect to material toughness, certain limits on steam generator pressure and temperature are established. Material toughness varies with temperature and is lower at room temperature than at operating temperature. One indicator of the temperature effect on ductility is the nil-ductility temperature (NDT). Therefore, a nil-ductility reference temperature ($RT_{NDT}$) has been determined by experimental means. The $RT_{NDT}$ is that temperature below which brittle (non-ductile) fracture may occur. For the steam generators, the $RT_{NDT}$ has been determined to be 10°F for steam generators A, B, and C and 30°F for steam generator D (Ref. 1). Considering uncertainties and proper margins, the minimum operating temperature has been determined to be 70°F. The 70°F temperature must be established before the pressure is increased to 200 psig. This limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits.

The fracture mechanic methodology, which is used to determine the stresses and material toughness, follows the guidance given by 10 CFR 50, Appendix G (Ref. 2). Reference 1 mandates the use of ASME Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 4).

### APPLICABLE SAFETY ANALYSES

The $RT_{NDT}$ limit is not derived from the Design Basis Accident analyses. The $RT_{NDT}$ limit is imposed during normal operation to avert encountering pressure/temperature combinations which are not analyzed as part of the steam generator design. Unanalyzed pressure/temperature combinations could cause propagation of minor, undetected flaws, which could cause brittle failure of the pressure boundary. Because the $RT_{NDT}$ limit is related to normal operation, the $RT_{NDT}$ limit is not a consideration in designing the accident sequences for theoretical hazard evaluations (Ref. 5).

(continued)

BASES (continued)

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| TR | TR 3.7.1 requires that the pressures on the primary and the secondary sides in the steam generator are kept at or below 200 psig when the temperature is less than or equal to 70°F. The pressure induced stress from the 200 psig pressure is low enough to be insignificant, even at temperatures at or below $RT_{NDT}$. |
| APPLICABILITY | The operating requirements which must be observed to avoid a condition, which could lead to brittle failure, are not strictly limited to specific MODES. Hence, in general, Applicability should be At All Times. However, in practice it is unlikely that these limits will be violated in the lower numbered MODES, due to the high operating temperature on the primary as well as the secondary side in the steam generators. Accordingly, the limits are most easily violated at low temperature, during shutdown and startup of the plant. Applicability can therefore conveniently be limited to whenever the temperature on the primary or the secondary side is at or below 70°F. Since cooldown would make the steam generator more susceptible to brittle failure, a Note to the Applicability has been added to preclude cooldown in the primary or secondary to ≤ 70°F when the pressure is > 200 psig. |
| ACTIONS | A.1, A.2, and A.3<br><br>With the combination of pressure and temperature not within limits, a reduction in pressure to ≤ 200 psig is required within 30 minutes. An engineering evaluation must be performed to determine the effect on the structural integrity of the pressure boundary. The evaluation must be finished and the conclusion made that no hazard exists before the temperature is increased to more than 200°F. Condition A is modified by a Note which states that whenever Condition A is entered, all ACTIONS A.1 through A.3 must be completed. |

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.1.1

TSR 3.7.1.1 verifies that the pressures on the primary and the secondary sides in the steam generators are less than 200 psig (value does not account for instrument error). At temperatures below 70°F (value does not account for instrument error), the temperature margin to $RT_{NDT}$ is diminished. Hence, the pressure must be checked every hour to ensure that the material toughness criteria are not violated. The 1 hour Frequency is based on engineering judgment and is consistent with industry practice.

Note: Instrument uncertainty has been considered in establishing these values and is discussed in this Bases section under Background.

REFERENCES

1. WCAP-13146, “Technical Basis for Determination of Secondary Side Pressure Test Temperatures in Sequoyah and Watts Bar Units 1 and 2 Steam Generators.”
2. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."
3. Not used.
4. ASME Boiler and Pressure Code, Section III.
5. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.7 PLANT SYSTEMS

B 3.7.2 Flood Protection Plan

BASES

BACKGROUND

Nuclear power plants are designed to prevent the loss of capability for cold shutdown and maintenance thereof resulting from the most severe flood conditions that can reasonably be predicted to occur at the site as a result of severe hydrometeorological conditions, seismic activity, or both (Ref. 1). Assurance that safety-related facilities are capable of surviving all possible flood conditions is provided by the flood protection plan.

The elevations of plant features which could be affected by the submergence during floods vary from 714.5 ft Mean Sea Level (MSL) (access to electrical conduits) to 736.9 ft MSL (including wave runup). Plant grade is elevation 728 ft MSL which can be exceeded by extreme rainfall floods and closely approached by seismic-caused dam failure floods. A warning plan is needed to assure plant safety from floods.

The warning plan is divided into two stages. This two-stage plan is designed to allow adequate time for preparing the plant for operation in the flood mode and to avoid excessive economic loss in case a potential flood does not fully develop. Stage I warning, which is a minimum of 10 hours, allows preparation steps, causing some damage to be sustained, but will postpone major economic damage. Stage II warning, which is a minimum of 17 hours, is a warning that a forthcoming flood above grade is predicted.

Stage I procedures consist of a controlled reactor shutdown and other easily revokable steps, such as moving flood supplies above the probable maximum flood elevation and making temporary connections and load adjustments on the onsite power supply. After unit shutdown, the Reactor Coolant System will be cooled and the pressure will be reduced to less than 350 psig. Stage II procedures are the least easily revokable and more damaging steps necessary to have the plant in the flood mode when the flood exceeds plant grade. Heat removal from the steam generators will be accomplished by adding river water from the Fire Protection System, and relieving steam to the atmosphere through the steam generator power operated relief valves. Other essential plant cooling loads will be transferred from the Component Cooling Water System to the Essential Raw Cooling Water System (ERCW); the ERCW

(continued)

BASES

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| BACKGROUND (continued) | will also replace the Raw Cooling Water System to the ice condensers. The Radioactive Waste System will be secured by filling tanks below Design Bases Flood (DBF) level with enough water to prevent floatation; one exception is the waste gas decay tanks, which are sealed and anchored against floatation. Power and communication lines running beneath the DBF that are not required for submersed operation will be disconnected, and batteries below the DBF will be disconnected (Ref. 2). |
| APPLICABLE SAFETY ANALYSES | The flood protection plan specifies flood control measures to protect safety related equipment in the event that the maximum elevation for the ultimate heat sink or other body of water, as applicable, is exceeded. Because external flooding conditions present substantial warning time to achieve plant shutdown, this requirement is not a contributor to a dominant risk sequence (Ref. 3). |
| TR | TR 3.7.2 requires that the flood protection plan be ready for implementation to maintain the plant in a safe condition. This requirement ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. |
| APPLICABILITY | The flood protection plan TR is applicable when one or more of the following conditions exist:<br><br>a. Extreme flood producing rainfall conditions in the east Tennessee watershed, or<br><br>b. Receipt of notification from TVA River Operations (RO) that predicted flood levels based on rainfall on the ground or potential failure problems with one or more dams combined with critical headwater elevations and flood-producing rainfall may result in subsequent issuance of a Stage I flood warning. |

(continued)

ACTIONS

A.1, A.2, and A.3

If a Stage I flood warning is issued, several actions are required to be taken. The first requires the plant to be placed in MODE 3 in 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

Upon issuance of a Stage I flood warning, initiate and complete the Stage I flood protection plan, which involves preparatory steps. The required Completion Time for this Required Action is 10 hours. The plant is also required to be brought from full power operation to a safe shutdown. This is accomplished by Required Action A.3. This Required Action requires the establishment of a SHUTDOWN MARGIN of at least 5% $\Delta k/k$ and $T_{avg}$ less than or equal to 350°F. The Completion Time of 10 hours is reasonable to accomplish the required SHUTDOWN MARGIN and $T_{avg}$.

A.4.1 and A.4.2

Once a Stage I flood warning has been issued, it is necessary to maintain communications between the TVA RO Group and the Watts Bar Nuclear Plant. This is necessary because the TVA RO Group provides the flood forecasting for the Watts Bar Nuclear Plant. The Completion Time of 10 hours corresponds to the time specified to initiate and complete the Stage I flood protection plan.

If communications between the TVA RO Group and the Watts Bar Nuclear Plant have not been established within the required Completion Time, the Stage II flood protection procedure must be initiated and completed within 27 hours. The Completion Time of 27 hours corresponds to the minimum pre-flood preparation time. This is to ensure adequate warning time for safe plant shutdown.

B.1

If the Stage II flood warning has been issued, the Stage II flood protection plan must be initiated and completed within 17 hours or prior to flooding of the site. The Completion Time of 17 hours corresponds to the remaining hours of the 27 hour pre-flood preparation time after the Stage I flood warning consisting of 10 hours has expired, and is an adequate time period to complete Stage II preparations.

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BASES

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| ACTIONS | B.1 (continued)<br><br>At any time it is determined that the potential for flooding at the site does not exist, the Stage I and Stage II flood protection plans are to be terminated immediately. |
| TECHNICAL SURVEILLANCE REQUIREMENTS | TSR 3.7.2.1<br><br>This surveillance requires communications between Watts Bar Nuclear Plant and TVA RO Group be established and maintained every 3 hours. A Note for this surveillance states that this is required only when one of the applicability criteria is met. This communications requirement exists because the TVA RO Group provides the flood forecasting for Watts Bar Nuclear Plant. The 3 hour Frequency is adequate for early flood forecasting. |
| REFERENCES | 1. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."<br><br>2. Watts Bar FSAR, Section 2.4.14, "Flooding Protection Requirements."<br><br>3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989. |

B 3.7 PLANT SYSTEMS

B 3.7.3 Snubbers

BASES

BACKGROUND

Component standard supports, are those metal supports which are designed to transmit loads from the pressure-retaining boundary of the component to the building structure. Although classified as component standard supports, snubbers require special consideration due to their unique function. Snubbers are either operated hydraulically or mechanically, depending on the nature of the support needed. They are designed to provide no transmission of force during normal plant operations, but function as a rigid support when subjected to dynamic transient loadings. Therefore, snubbers are chosen in lieu of rigid supports where restricting thermal grow during normal operation would induce excessive stresses in the piping nozzles or other equipment. The location and size of the snubbers are determined by stress analysis. Depending on the design classification of the particular piping, different combinations of load conditions are established. These conditions combine loading during normal operation, seismic loading and loading due to plant accidents/transients to four different loading sets. These loading sets are denominated: normal, upset, emergency, and faulted condition. The actual loading included in each of the four conditions, depends on the design classification of the piping. The calculated stresses in the piping and other equipment, for each of the four conditions, must be in conformance with established design limits.

Supports for pressure-retaining components are designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 (Ref. 1). The combination of loadings for each support, including the appropriate stress levels, meet the criteria of Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports" (Ref. 2), and Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports" (Ref. 3).

(continued)

BASES (continued)

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| APPLICABLE SAFETY ANALYSES | Pipe and equipment supports, in general, are not directly considered in designing the accident sequences for theoretical hazard evaluations. Further, various Probabilistic Risk Assessment (PRA) studies have indicated that snubbers are not of prime importance in a risk significant sequence (Ref. 4 and 5). Therefore, the function of the snubbers is not essential in mitigating the consequences of a DBA or transient (Ref. 6). |
| TR | TR 3.7.3 requires that all snubbers utilized on safety related equipment shall be OPERABLE. Those snubbers that are utilized on non-safety related systems shall be OPERABLE if a failure could have adverse effect on a safety related system. Individual snubbers may be removed from service for functional testing within the limits established herein without violating these requirements, although Required Actions and Completion Times still apply. |
| APPLICABILITY | The OPERABILITY of the snubbers is required in MODES 1, 2, 3, and 4. For MODES 5 and 6, the OPERABILITY is limited to the snubbers located on those systems which need to be OPERABLE in MODES 5 and 6. |

ACTIONS

A.1.1, A.1.2, and A.2

If one or more snubbers have been declared inoperable, the snubber(s) must be restored to OPERABLE status in 72 hours. Alternatively, the snubber(s) must be replaced in the 72 hours. In either case, an engineering evaluation per Table 3.7.3-5 must be performed during the 72 hours to:

a) Determine the cause of the failure

As a result of this evaluation, the need for testing other snubbers will be considered. The results from the testing will be used to consider expanded functional testing and cause examination with consideration of manufacturing and design deficiency. It should be noted that the testing must be independent and not combined with TSR 3.7.3.3.

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ACTIONS

A.1.1, A.1.2, and A.2 (continued)

b) Determine the impact on the supported component

This evaluation shall determine if the inoperable snubber has adversely affected the attached component.

The 72 hours is based on engineering experiences and is reasonable, considering the time it will take to identify the problem and take the proper corrective actions. This requirement is considered met for those snubbers rendered inoperable by removal for functional testing by the generic engineering evaluation included in Reference 9.

A.3

Another alternative is to perform an engineering evaluation to demonstrate inoperable snubber(s) do not impact the OPERABILITY of the supported system for the existing plant condition (Reference 10).

B.1

If Required Actions under Condition A are not met within the 72 hours, the supported system or component is immediately declared inoperable.

TECHNICAL SURVEILLANCE REQUIREMENTS

The TSRs are preceded by three Notes. Note 1 states that the snubber inservice inspection program shall be carried out in accordance with the requirements in Tables 3.7.3-1, 3.7.3-2 and 3.7.3-3. This represents an enhanced snubber inservice inspection program compared to the Inservice Inspection Program which stipulates inservice inspection in accordance with ASME Section XI. The snubber inservice inspection program includes the requirements of Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." ASME Section XI, 1989 Edition, Subsections IWF-5300(a) and (b) require that inservice examinations of snubbers, using the VT-3 visual examination method described in IWA-2213, and inservice tests of snubbers be performed in accordance with the first Addenda to ASME/ANSI OM-1987, Part 4. Note 2 requires repair or replacement of snubbers which fail inspection, and testing of repaired snubbers before installation. Note 3 indicates that a "snubber type," as used in this TR, is determined by the design and manufacturer, but not by size.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.1

TSR 3.7.3.1 comprises a visual inspection of the snubbers. A pre-fuel load visual inspection and functional test has been performed on each snubber using the acceptance criteria listed in this TSR. The baseline considers that the snubbers have experienced thermal cycling and normal operating service as a result of previous hot functional testing. The initial inservice inspection must be performed on the snubbers prior to completion of the first refueling outage. The frequency of subsequent surveillances depends on the number of snubbers found inoperable from each previous inspection as provided in Table 3.7.3-2 and the Inservice Inspection Program. The acceptance criteria and the remedial ACTIONS are listed in Table 3.7.3-1.

The visual inspections are designed to detect obvious indications of inoperability of the snubbers. Removal of insulation or direct contact with the snubbers is not required initially. However, suspected causes of inoperability are to be investigated and all snubbers of the same type and all snubbers subjected to the same failure mode are to be inspected more frequently.

The visual inspection frequency is based upon the number of unacceptable snubbers found during the previous inspection. Therefore, the required inspection intervals vary inversely with the number of inoperable snubbers found during an inspection. If a snubber fails the visual acceptance criteria, the snubber is declared unacceptable and cannot be declared OPERABLE via functional testing. However, if the cause of rejection is understood and remedied for that type of snubber and for any other type of snubbers, that may be generically susceptible, and OPERABILITY verified by testing, that snubber may be reclassified acceptable for the purpose of establishing the next surveillance interval.

Snubbers may be categorized, according to accessibility, as noted in the Note to Table 3.7.3-2. The accessibility of each snubber is determined based on radiation level as well as other factors such as temperature, atmosphere, location, etc. The recommendations of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable," (Ref. 7) and Regulatory Guide 8.10, "Operation Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable," (Ref. 8), are considered in planning and implementing the visual inspection program.

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BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.2

TSR 3.7.3.2 comprises the inspection of all snubbers attached to systems that have experienced unexpected, potentially damaging transients. The potential impact of the transients is assessed by reviewing operating data and by visually inspecting the associated systems. The review and the inspection must be performed within six months of the event. In addition to the inspection, the freedom-of-motion of the mechanical snubber(s) is verified in accordance with Table 3.7.3-3.

TSR 3.7.3.3

TSR 3.7.3.3 comprises the functional testing of hydraulic and mechanical snubbers. The testing for these snubbers have been separated into two sample plans. Sample Plan A (10%) is used for the hydraulic steam generator snubbers based on the small population. Sample Plan B (37 snubbers) is used for mechanical snubbers based on the large population. The plans, when used in combination, are a conservative approach versus using only the Sample Plan B for the entire population.

Snubber functional testing is performed prior to completion of each refueling outage. This frequency is based on engineering experience and is reasonable for testing of a representative sample of snubbers. Credit may be taken toward meeting minimum outage testing requirements for mechanical snubbers functionally tested within the refueling cycle. Snubbers may be removed from service for functional testing in Modes 1 through 4 provided that the following administrative controls are implemented:

1. Required Actions and Completion Times must be met.

2. No more than one snubber may be removed from service at a time on any line and attached piping which is analyzed as a seismic subsystem. Multiple snubbers may be removed for testing simultaneously only if separated by two or more seismic anchors.

3. Snubbers on trained systems or portions of systems may be removed only on the train which is undergoing maintenance in that work week. Snubbers on non-trained systems or portions of systems may only be removed following a documented risk assessment. Snubbers may not be removed from service for testing on one train of a system while the other train has been declared inoperable for any reason.

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.3.3 (continued)

4. Snubbers adjacent to equipment nozzles may not be removed for testing except in Modes 5 and 6. In determining the applicability of this limitation, engineering judgment must be used regarding the placement of the snubber relative to the nozzle, the routing of the affected piping, and any other supports available to protect equipment function.

TSR 3.7.3.4

The TSR is preceded by three Notes which underline the need for considering service life of sub-components and to replace these sub-components before the end of the respective service lives. The replacement of sub-components must be documented and the documentation retained for further reference. TSR 3.7.3.4 addresses the monitoring of the service life of the snubbers. The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. The expected service life is established by the manufacturer and is based on operating experience with critical snubber parts such as seals and springs in a radiation environment. The every refueling outage Frequency is based on engineering experience and is reasonable for the verification service life.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III and XI.

2. Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports."

3. Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports."

4. "Zion Probabilistic Safety Study", Commonwealth Edison Company, September 1981.

5. "Millstone Unit 3 Probabilistic Safety Study," North-east Utilities Company, August 1983.

(continued)

BASES

REFERENCES (continued)

6. NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of The Commission's Interim Policy Statement Criteria to Standard Technical Specifications, Attachment to letter dated May, 1988 from T. E. Murley, NRC to W. S. Wilgus, Chairman The B&W Owners Group.

7. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable."

8. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable."

9. SA/SE WBPLCE-97-028-0, RIMS T28970829803.

10. Screening Review WBPLCE-02-003-0.

B 3.7 PLANT SYSTEMS

B 3.7.4 Sealed Source Contamination

BASES

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| BACKGROUND | A sealed source is any byproduct, source, or special nuclear material that is encased in a capsule designed to prevent leakage or escape of the material (Ref. 1). Sealed sources are classified into three groups according to their use (sources in use, not in use, and startup sources and fission detectors) and may contain alpha, beta, gamma, or neutron emitting material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on Reference 2. Those sources that are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring, excore fission detector assemblies or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism. |
| APPLICABLE SAFETY ANALYSES | The sealed source contamination requirement ensures that leakage from sealed sources will not exceed allowable intake values. This TR is important to the safety of plant personnel, however it is not required to mitigate the consequences of a DBA or transient (Ref. 3). |
| TR | TR 3.7.4 requires that the removable contamination shall be less than 0.005 microcuries for each sealed source containing the following radioactive material:<br><br>a. Greater than 100 microcuries of beta and/or gamma emitting material; or<br><br>b. Greater than 5 microcuries of alpha emitting material. |
| APPLICABILITY | Since the limits on the removable contamination for each sealed source containing radioactive material are not MODE dependent, this TR is applicable at all times. |

(continued)

BASES (continued)

ACTIONS

Since this TR is applicable at all times, the Required Actions have been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

A.1

With a sealed source having removable contamination in excess of the limits, the sealed source should be withdrawn from use immediately. The immediate Completion Time reflects the importance of preventing the contamination from spreading.

A.2.1 and A.2.2

If the sealed source contamination is not within the specified limit and the sealed source has been removed from use, the sealed source must be decontaminated and repaired, otherwise, disposal of the sealed source is required. If the sealed source is to be decontaminated and repaired, it must be done prior to returning the sealed source to use. If disposal of the sealed source is to be done, it must be completed in accordance with NRC regulations.

TECHNICAL SURVEILLANCE REQUIREMENTS

Notes have been added to this section stating that the licensee or other persons specifically authorized by the NRC shall perform the TSRs, and that the test methods used shall have a detection sensitivity of greater than or equal to 0.005 microcurie per test sample.

TSR 3.7.4.1

This surveillance determines every 6 months that the removable contamination is less than 0.005 microcuries for each sealed source. The 6 month Frequency is frequent enough to identify a leaking or contaminated sealed source without having extensive spreading of contamination.

This surveillance is modified by several Notes. The Notes state that this TSR is only applicable to sources in use, to sources with half-lives of more than 30 days, and to sources in any form other than gas. Also, this TSR is not applicable to startup sources and fission detectors previously subjected to core flux.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2

This surveillance determines within 6 months prior to use or transfer to another licensee that the removable contamination is less than 0.005 microcuries for each sealed source and fission detector. This Frequency is adequate to identify a leaking or contaminated sealed source or fission detector to avoid extensive contamination.

This surveillance is modified by two Notes. The first Note states that this TSR is only applicable to sealed sources not in use. The second Note states that sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed in use.

TSR 3.7.4.3

This surveillance determines that the removable contamination is less than 0.005 microcuries for each startup source and fission detector. This test should be performed on each startup source and incore fission detector within 31 days prior to being installed in the core or being subjected to core flux. It also should be performed following any repairs or maintenance to the source. This Frequency ensures that the startup source or fission detector is not leaking or contaminated over the specified limit.

This Surveillance is modified by a Note stating this TSR only applies to startup sources and incore fission detectors that are not in use.

TSR 3.7.4.4

This surveillance determines that the removable contamination is less than 0.005 microcuries for each excore fission detector. This test should be performed on each excore fission detector assembly within 31 days prior to the detector assembly being installed in its permanent configuration. It also should be performed following any repairs or maintenance to the detector assembly. This frequency ensures that the excore detectors are not leaking or contaminated over the specified limit.

This Surveillance is modified by a Note stating this TSR only applies to excore fission detectors.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 70.4 "Definitions."

2. 10 CFR 70.39 "Specific Licenses for the Manufacture or Initial Transfer of Calibration or Reference Sources."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.7 PLANT SYSTEMS

B 3.7.5 Area Temperature Monitoring

BASES

BACKGROUND

Thermal-life of various electrical and mechanical equipment is one of several important aging concerns in the qualification of hardware. The requirement is that the equipment remains functional during and after specified design basis events. Design basis events consist of loss of offsite power and design basis accidents (DBA). In general, the following three groups of hardware are subjected to qualification:

a. Safety related equipment,

b. Non-safety related equipment (failure of which could prevent safety related equipment to operate as designed), and

c. Specific post-accident monitoring equipment.

The normal service temperatures of concern are relatively low, hence, most of the equipment requiring consideration are components in the electrical power supply and the instrumentation systems. Some of these components are designed for relatively low temperature with very little margin to normal operating temperatures in cabinets and buildings. The procedure for thermal qualification is normally to subject prototypes from the production line to life tests by natural or artificial (accelerated) aging to its end-of-installed life condition. Analyses with justifications of methods and assumptions are used to qualify the prototypes to the actual service conditions, which may differ from the test conditions. Although the equipment is qualified for an environment expected after a DBA, the components are only subjected to normal operating conditions for most of the design life. Therefore, the thermal aging due to normal operating conditions is of major importance and is the parameter which is controlled by the Technical Requirements. Accordingly, this particular requirement establishes temperature limits during normal operation for specific locations in various buildings, except the containment. The temperature limits are related to the expected thermal-life for the hardware which operates in the areas where the temperatures are monitored and controlled.

(continued)

BASES

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| BACKGROUND (continued) | Due to valve design, ambient temperatures can affect the setpoints of the main steam safety valves (MSSVs), whereby a decrease in valve body temperature causes an increase in setpoint, resulting in non-conservative relief pressure. Ambient temperatures are monitored within the main steam valve vaults to ensure that the MSSVs minimum temperatures are maintained to meet the 1% code allowable variance on setpoints. Detailed BASES for the MSSVs is provided in Technical Specification B 3.7.1.<br><br>The general guidelines, which are followed for the qualification of electrical equipment, are provided in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 1). Detailed requirements for the implementation of the general guidelines are provided in various Regulatory Guides and IEEE Standards. Basic requirements for the qualification of mechanical equipment are outlined in General Design Criteria 4 (Ref. 2). |
| APPLICABLE SAFETY ANALYSES | Certain components, which have the service temperatures controlled by this requirement, are part of the primary success path and function to mitigate DBAs and transients. However, the integrity/OPERABILITY of these components is addressed in the relevant specifications that cover individual components. The service temperatures and the thermal aging, which are controlled by observing the requirements of this TR, are not inputs to the safety analysis. Further, Probabilistic Risk Assessment studies, performed to date, do not explicitly model the function of area temperature monitors. In addition, this particular requirement covers only service temperatures and thermal aging of these components, which are not considerations in designing the accident sequences for theoretical hazard evaluations (Ref. 3). |
| TR | TR 3.7.5 provides nominal temperature limits in the vicinity of major equipment. The TR allows for each area shown in Table 3.7.5-1 to be higher or lower than the normal limit for a maximum of eight hours. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error. |
| APPLICABILITY | The limits on temperature and time apply whenever the affected equipment in an affected area is required to be OPERABLE. |

(continued)

BASES (continued)

ACTIONS

A.1

Whenever the temperature in one or more areas has exceeded the normal temperature limits for more than eight hours, document the exceedance in accordance with the Corrective Action Program. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits.

Condition A has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

B.1.1, B.1.2, and B.2

Whenever the temperature in one or more areas has exceeded the abnormal temperature limits, the temperature must be restored to within the normal limits in 4 hours. The Completion Time of 4 hours is based on operator experience and is a reasonable time for restoring the temperature. Alternatively, the affected equipment must be declared inoperable and the inoperability documented in accordance with the Corrective Action Program along with the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be prepared which demonstrates OPERABILITY of the affected equipment.

C.1 and C.2

Whenever the temperature in the Intake Pumping Station mechanical or electrical equipment rooms exceeds the lower limit of 40°F, actions must be initiated within 24 hours to ensure the temperature does not decrease below 32 °F. The 24 hour Completion Time hours is based on temperature analysis. Within 7 days, restore normal temperatures within the areas affected. The 7 day Completion Time is based on a reasonable repair duration, and compensatory actions available during the interim period to maintain temperatures above 32°F.

(continued)

BASES

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| ACTIONS (continued) | D.1 and D.2<br><br>If the temperature in the Intake Pumping Station mechanical or electrical equipment rooms decreases to 32°F or lower, the affected equipment must be immediately declared inoperable. The Completion Time is based on potential freezing of safety-related components. The inoperability of the equipment must be documented in the Corrective Action Program along with the cumulative time and amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. |
| TECHNICAL SURVEILLANCE REQUIREMENTS | TSR 3.7.5.1<br><br>The temperatures for the areas listed in Table 3.7.5-1 must be determined every 12 hours to ensure compliance with the limits. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error. |
| REFERENCES | 1. 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."<br><br>2. 10 CFR 50, Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases."<br><br>3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989. |