DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
3.3-6	3.3-6
3.3-7	3.3-7
3.5-31	3.5-31
3.5-32	3.5-32
3.6-2	3.6-2

Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad.(1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling.(2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

The requirement for a flowpath from LPI discharge to HPI pump suction is provided to assure availability of long term core cooling following a small break LOCA in which the BWST is depleted and RCS pressure remains above the shutoff head of the LPI pumps.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core. The minimum boron concentration is specified in the Core Operating Limits Report.

It has been shown that the containment temperature response following a LOCA or main steam line break accident will be within the equipment qualification analysis conditions with one train of Reactor Building spray and two Reactor Building coolers operable.(4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Valve LPSW-108 is the LPSW isolation valve on the discharge side of each Unit's RBCUs. This valve is required to be locked open in order to assure the LPSW flowpath for the RBCUs is available.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation.

The Units 1 and 2 LPSW system requires two pumps to meet the single failure criterion provided that one of the Units has been defueled and the following LPSW system loads on the defueled Unit are isolated: RBCUs, Component Cooling, main turbine oil tank, RC pumps, and LPI coolers. In this configuration, if two of the three LPSW pumps are inoperable, 72 hours are permitted by TS 3.3.7.b to restore two of the three LPSW pumps to operable status. At all other times when the RCS of Unit 1 or 2 is ≥ 350 psig or ≥ 250°F, all three LPSW pumps are required to meet the single failure criterion. When all three LPSW pumps are required to be operable and one of the three pumps is inoperable, 72 hours are permitted by TS 3.3.7.b to restore the pump to operable status.

The operability of redundant equipment(s) is determined based on the results of inservice inspection and testing as required by Technical Specification 4.5 and ASME Section XI.

REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.3.3.2

(4) FSAR, Section 15.14.5

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure	Reactor Building Spray	≤15 psig
	High-Pressure Injection	≤4 psig
	Low-Pressure Injection	≤4 psig
	Start Reactor Building Cooling & Reactor Building Isolation (Essential and Non-essential Systems)	≤4 psig
	Penetration Room Ventilation	≤4 psig
Lower Reactor Coolant System Pressure	High Pressure Injection (1) & Reactor Building Isolation (Non-essential systems)	≥1500 psig
	Low Pressure Injection (2)	≥500 psig

(1) May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

(2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 15 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1500 psig low reactor coolant pressure setpoint for high pressure injection initiation and 500 psig for low pressure injection is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.(1)

REFERENCE

(1) FSAR, Section 15.14.

2. For plant conditions when the Reactor Coolant System temperature is above 250°F and pressure is above 350 psig but the reactor is at or below hot shutdown, one Reactor Building Purge isolation valve on each penetration may be open for testing and/or maintenance per Specification 4.4.4.1 and 3.6.6.

3. For plant conditions other than contained in Specification 3.6.3.b.1, .2 above, with one or more Reactor Building purge valves open, the open valves shall be closed within one hour, or the plant shall be in hot shutdown within 12 hours and within an additional 24 hours, Reactor Coolant System temperature below 250°F and pressure below 350 psig.

c. A containment isolation valve, other than a Reactor Building Purge isolation valve, may be inoperable provided either:

1. The inoperable valve is restored to operable status within four hours.

2. The affected penetration is isolated within four hours by the use of a deactivated automatic valve secured and locked in the isolated position.[1]

3. The affected penetration is isolated within four hours by the use of a closed manual valve or blind flange.[1]

4. The reactor is in the hot shutdown condition within 12 hours and cold shutdown within 24 hours.

3.6.4 The reactor building internal pressure shall not exceed 1.2 psig or a vacuum of -2.5 psig if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

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[1]Penetration flow paths (except for the Reactor Building Purge flow path) may be unisolated intermittently under admistrative controls.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

TECHNICAL SPECIFICATIONS
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Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad.—(1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling.(2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

The requirement for a flowpath from LPI discharge to HPI pump suction is provided to assure availability of long term core cooling following a small break LOCA in which the BWST is depleted and RCS pressure remains above the shutoff head of the LPI pumps.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent Δk/k subcritical at 70°F without any control rods in the core. The minimum boron concentration is specified in the Core Operating Limits Report.

It has been shown *that the containment temperature response following a LOCA or main steam line break accident will be within the equipment qualification analysis conditions*~~for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded~~ with one *train of Reactor Building* spray and two *Reactor Building* coolers operable.—(4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor

Building spray systems are operable ~~for seven days~~ or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Valve LPSW-108 is the LPSW isolation valve on the discharge side of each Unit's RBCUs. This valve is required to be locked open in order to assure the LPSW flowpath for the RBCUs is available.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation.

The Units 1 and 2 LPSW system requires two pumps to meet the single failure criterion provided that one of the Units has been defueled and the following LPSW system loads on the defueled Unit are isolated: RBCUs, Component Cooling, main turbine oil tank, RC pumps, and LPI coolers. In this configuration, if two of the three LPSW pumps are inoperable, 72 hours are permitted by TS 3.3.7.b to restore two of the three LPSW pumps to operable status. At all other times when the RCS of Unit 1 or 2 is ≥ 350 psig or ≥ 250°F, all three LPSW pumps are required to meet the single failure criterion. When all three LPSW pumps are required to be operable and one of the three pumps is inoperable, 72 hours are permitted by TS 3.3.7.b to restore the pump to operable status.

The operability of redundant equipment(s) is determined based on the results of inservice inspection and testing as required by Technical Specification 4.5 and ASME Section XI.

REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.3.3.2

(4) FSAR, Section 15.14.5

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure	Reactor Building Spray	≤*15*~~30~~ psig
	High-Pressure Injection	≤4 psig
	Low-Pressure Injection	≤4 psig
	Start Reactor Building Cooling & Reactor Building Isolation (Essential and Non-essential Systems)	≤4 psig
	Penetration Room Ventilation	≤4 psig
Lower Reactor Coolant System Pressure	High Pressure Injection (1) & Reactor Building Isolation (Non-essential systems)	≥1500 psig
	Low Pressure Injection (2)	≥500 psig

(1) May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

(2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the *1530* psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1500 psig low reactor coolant pressure setpoint for high pressure injection initiation and 500 psig for low pressure injection is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.(1)

REFERENCE

(1) FSAR, Section 15.14.

2. For plant conditions when the Reactor Coolant System temperature is above 250°F and pressure is above 350 psig but the reactor is at or below hot shutdown, one Reactor Building Purge isolation valve on each penetration may be open for testing and/or maintenance per Specification 4.4.4.1 and 3.6.6.

3. For plant conditions other than contained in Specification 3.6.3.b.1, .2 above, with one or more Reactor Building purge valves open, the open valves shall be closed within one hour, or the plant shall be in hot shutdown within 12 hours and within an additional 24 hours, Reactor Coolant System temperature below 250°F and pressure below 350 psig.

c. A containment isolation valve, other than a Reactor Building Purge isolation valve, may be inoperable provided either:

1. The inoperable valve is restored to operable status within four hours.

2. The affected penetration is isolated within four hours by the use of a deactivated automatic valve secured and locked in the isolated position.[1]

3. The affected penetration is isolated within four hours by the use of a closed manual valve or blind flange.[1]

4. The reactor is in the hot shutdown condition within 12 hours and cold shutdown within 24 hours.

3.6.4 The reactor building internal pressure shall not exceed 1.2~~5~~ psig or a vacuum of *-2.5 psig*~~of five inches of Hg~~ if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

[1]Penetration flow paths (except for the Reactor Building Purge flow path) may be unisolated intermittently under admistrative controls.

TECHNICAL JUSTIFICATION

Proposed Technical Specification Change

The proposed changes to Technical Specifications 3.5.3 and 3.6.4, in conjunction with changes to the Bases of Technical Specifications 3.3, modify requirements associated with limiting peak Reactor Building pressure. These modifications lower the allowable Reactor Building pressure and lower the actuation setpoint for the Reactor Building Spray System. These changes are proposed as a result of an NRC Safety Evaluation Report (SER) that approved a new containment pressure response methodology documented in Duke Power topical report DPC-NE-3003-P.

Background Information

Following a high energy line break, the Low Pressure Injection (LPI) coolers, Reactor Building Cooling Units (RBCUs) and Reactor Building Spray (RBS), in conjunction with a worst case single failure, must be capable of performing two functions. These two functions are to maintain the Reactor Building temperature less than the environmental qualification (EQ) envelope and to maintain Reactor Building internal pressure less than fifty-nine (59) psig. The heat removal capabilities of the LPI coolers and RBCUs are tested at least once per quarter to ensure that these systems are operable. The acceptance criteria for these tests are based, in part, on the containment pressure/temperature response to a mass and energy release following a high energy line break inside containment. The Duke Power Company methodology for simulating the mass and energy release from high energy line breaks and the resulting containment response for Oconee Nuclear Station is presented in Duke Power topical report DPC-NE-3003-P, "Mass and Energy Release and Containment Response Methodology", submitted to the NRC on August 11, 1993. The NRC approved the use of this methodology in an SER dated March 15, 1995. The results of the analyses presented in the topical report demonstrate that the EQ requirements and containment pressure acceptance criteria are met for a range of LPI cooler and RBCU performance. The analyses in the topical report will require less restrictive performance requirements for the LPI coolers and the RBCUs, and less frequent cleaning and testing of these systems. Several changes to the Oconee Technical Specifications are required based on the input assumptions that are used in these analyses.

Proposed Changes and Justifications

Technical Specification 3.5.3, Engineered Safety Features Protective System Actuation Setpoints

Proposed Change

The setpoint for the Reactor Building Spray System actuation on high Reactor Building pressure is proposed to be changed from ≤ 30 psig to ≤ 15 psig.

Justification

The small break LOCA containment pressure response analysis which is presented in Duke Power topical report DPC-NE-3003-P requires an actuation setpoint for the Reactor Building Spray system of 20 psig. The long term Reactor Building temperature response following a small break Loss of Coolant Accident (LOCA) is improved by the initiation of Reactor Building Spray at 20 psig relative to initiation at 30 psig. An allowance of 5 psig is applied to the analysis assumption to give the proposed setting of 15 psig. Additional margin exists in the implementation of this setpoint since it is administratively controlled via procedure to ≤ 10 psig. The setpoint has been administratively controlled at ≤ 10 psig since 1971, and has not caused any operational concerns. The proposed change is more restrictive than the existing setpoint and is consistent with the existing bases since it is low enough to cover the entire spectrum of break sizes and high enough to prevent spurious initiation during normal operation.

Technical Specification 3.6.4, Reactor Building Pressure

Proposed Change

The maximum allowable Reactor Building internal pressure is proposed to be changed from 1.5 psig to 1.2 psig when the reactor is critical.

Justification

The post-LOCA Reactor Building peak pressure analysis presented in Duke Power topical report DPC-NE-3003-P assumed an initial internal pressure of 1.2 psig. As described in Section 6.2 of the report, the peak Reactor Building pressure remains below 59 psig for all Reactor Building temperatures with an initial Reactor Building pressure of 1.2 psig. This proposed Technical Specification change to 1.2 psig is more restrictive than the existing Technical Specification, and is consistent with the existing bases since peak Reactor Building pressure remains below the design pressure for all analyses. Current operating procedures require Reactor Building depressurization if the Reactor Building pressure indications exceed 0.6 psig. Duke Power Company calculations indicate that the uncertainty of the average of the three Reactor Building pressure indications is less than 0.4 psig. Therefore, using the average of the three indications, in making the determination when Reactor Building depressurization is required, will ensure that the actual Reactor Building pressure remains below 1.2 psig at all times.

The lower Reactor Building pressure limit (a vacuum of 5 in. of Hg) is unaffected by this change, and is still applicable. However, this vacuum limit has been converted from inches of Hg to psig. This change allows the units specified in the Technical Specification to be consistent with the units displayed on Control Room instrumentation.

Technical Specification 3.3.5 - Reactor Building Cooling and Technical Specification 3.3.6 - Reactor Building Spray, Bases

Proposed Change

The first sentence in the Bases paragraph related to Reactor Building Cooling and Reactor Building Spray is proposed to be revised as follows:

> It has been shown that the containment temperature response following a LOCA or main steam line break accident will be within the equipment qualification analysis conditions with one spray and two coolers operable.

Justification

The current wording of the Bases implies that the operation of the Reactor Building sprays and coolers are required to meet the containment pressure acceptance criterion following a LOCA. This implication conflicts with the Bases for Technical Specification 4.5.2 and with the analyses in the Duke Power topical report DPC-NE-3003-P. The Reactor Building sprays and coolers are not required to mitigate the pressure response following a LOCA because the pressurization rate is too rapid and the peak containment pressure is achieved prior to the operation of these systems. The proposed change reflects the methodology presented in DPC-NE-3003-P, which credits the operation of the sprays and coolers to mitigate the containment temperature response following a LOCA or a steam line break.

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

No. The analysis of the post-LOCA Reactor Building response to high-energy line breaks, using the new methodology, uses assumptions different from the requirements currently delineated in Technical Specifications. The new assumptions used for initial Reactor Building pressure and Reactor Building Spray system actuation are 1.2 psig and 20 psig respectively. These values are lower, and hence more conservative, than the values currently specified in Technical Specifications.

Since the new values for Reactor Building pressure and Reactor Building Spray actuation are more conservative and the analysis methodology has received approval from the NRC via a SER, this change does not involve a significant increase in the probability or consequences of an accident previously identified.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

No. The methodology for Reactor Building high energy line break analysis is being revised. The revision of the method of analysis does not alter the manner by which plant systems and components function for accident mitigation.

(3) Involve a significant reduction in a margin of safety.

No. By letter dated March 15, 1995, the NRC stated that the new analyses described in the topical report, DPC-NE-3003-P, expand the scope of analyzed piping failures in containment for the Oconee facilities. The NRC further stated that this new analysis method has been used to reanalyze existing licensing basis pipe failure events in containment, and to examine the potential effects of previously unanalyzed assumptions and initial conditions which the NRC staff finds to be consistent with current NRC staff acceptance criteria or produce equally conservative results. In conclusion, the NRC confirmed that this methodology, with appropriate adjustments to reflect potential plant modifications, may be used by Duke Power to perform future analyses in support of licensing applications related to containment accident response. This proposed change to Technical Specifications reflects the use of this new methodology. Based on this new methodology, changes have been made to setpoint assumptions for initial Reactor Building pressure and Reactor Building Spray actuation. This proposed Technical Specification change reflects those assumption changes. This methodology has been accepted by the NRC. This proposed change to Technical Specifications does not involve a significant reduction in the margin of safety.

Duke has concluded, based on the above justification, that there are no significant hazards considerations involved in this amendment request.

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10 CFR 51.22 (b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the No Significant Hazards Consideration evaluation that is contained in Attachment 4.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   The proposed amendment will not change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

   The proposed will not increase the individual or cumulative occupational radiation exposure.

In summary, the proposed amendment request meets the criteria set forth in 10 CFR 51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.