Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

July 25, 2003
LIC-03-0050

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

Reference: Docket No. 50-285

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Reactor Coolant System Leakage Limits"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby requests the following amendment to Fort Calhoun Station (FCS) Technical Specification 2.1.4, "Reactor Coolant System (RCS) Leakage Limits." This proposed amendment will: (1) add a requirement for no RCS pressure boundary leakage, (2) combine the existing RCS leakage limits into a format similar to the Improved Standard Technical Specification (ISTS), and (3) replace the existing basis associated with this specification with a basis similar in format and content of the ISTS. The proposed changes will assure that design criteria of no RCS pressure boundary leakage is maintained and bring the FCS RCS leakage specifications into alignment with the Improved Standard Technical Specifications. This amendment is modeled after the Improved Standard Technical Specifications.

Attachment 1 provides the No Significant Hazards Evaluation and the technical bases for this requested change to the Technical Specifications. Attachment 2 contains a marked-up version reflecting the requested Technical Specification and Basis changes. Attachment 3 contains a clean version reflecting the proposed Technical Specification and Basis.

OPPD requests approval of the proposed amendment by March 1, 2004. Once approved, the amendment shall be implemented within 90 days.

A001

I declare under penalty of perjury that the foregoing is true and correct. (Executed on July 25, 2003)

If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833.

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/RRL/rrl

Attachments:
1. Fort Calhoun Station's Evaluation
2. Markup of Technical Specification Pages
3. Proposed Technical Specification Pages

c: T. P. Gwynn, Acting Regional Administrator, NRC Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska

ATTACHMENT 1

Fort Calhoun Station's Evaluation for Amendment of Operating License

Fort Calhoun Station's Evaluation for Amendment of Operating License

1.0 INTRODUCTION

This letter is a request to amend Operating License DPR-40 for Fort Calhoun Station Unit No. 1.

The proposed change will revise Fort Calhoun Station (FCS) Technical Specification 2.1.4, "Reactor Coolant System (RCS) Leakage Limits." This proposed amendment will: (1) add a requirement for no RCS pressure boundary leakage, (2) combine the existing RCS leakage limits into a format similar to the Improved Standard Technical Specification (ISTS), and (3) replace the existing basis associated with this specification with a basis similar in format and content of the ISTS. The proposed changes will assure that design criteria of no RCS pressure boundary leakage is maintained and bring the FCS RCS leakage specifications into alignment with the Improved Standard Technical Specifications. This amendment is modeled after the Improved Standard Technical Specifications.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed changes to Technical Specifications Section 2.1.4 will add a requirement for no RCS pressure boundary leakage. This proposed change will assure that design criteria of no RCS pressure boundary leakage is maintained as recommended by the NRC Davis-Besse Lessons-Learned Report (Reference 10.5). The remainder of the proposed changes will bring the FCS specifications into closer alignment with the Improved Standard Technical Specifications and are not technical in nature. This amendment implements the requirements in the Improved Standard Technical Specifications, Reference 10.1. The applicable limits are either equal to or less than present limits; modes and entry conditions are not being changed.

3.0 BACKGROUND

This proposal resulted from a FCS Condition Report corrective action written as a result of a review by FCS of the NRC Davis-Besse Lessons-Learned Report. This review identified: (1) that present RCS leakage requirements do not include a limit on pressure boundary leakage, and (2) RCS pressure boundary leakage is being monitored by the performance of other programs and surveillances even though there is no technical specification requirement. Replacing the entire RCS leakage specification with the specifications provided in ISTS will provide the necessary technical

specification requirements on RCS pressure boundary leakage and minimize the chance for incorrect interpretation of the existing RCS leakage specifications.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

The proposed Technical Specifications Section 2.1.4 satisfies the FCS Design Criterion 16 *Monitoring Reactor Coolant Pressure Boundary*, which is similar to 10 CFR 50, Appendix A, General Design Criteria (GDC) 30, *Quality of reactor coolant pressure boundary*. FCS was issued a construction permit prior to May 21, 1971, and therefore the GDC is based upon the plant-specific design criterion documented in Appendix G of the FCS Updated Safety Analysis Report (Reference 10.4). These changes will ensure that proper limiting conditions for operation are entered for equipment or functional inoperability.

5.0 TECHNICAL ANALYSIS

The proposed changes to Technical Specifications Section 2.1.4 will clarify the scope, allowed outage times, and required actions for reactor coolant system leakage. The proposed change will assure that FCS continues to satisfy FCS Design Criterion 16, *Monitoring Reactor Coolant Pressure Boundary*. These proposed changes have no affect upon design basis.

6.0 REGULATORY ANALYSIS

The present Technical Specification Section 2.1.4 has no provisions for monitoring RCS pressure boundary leakage, and as worded is subject to interpretation or confusion by some readers. The proposed changes to Technical Specifications Section 2.1.4 will clarify the scope, allowed outage times, and actions required for the reactor coolant system leakage. The changes will provide clearer identification of: (1) allowed leakage rates, (2) allowed outage time or time to restore operability, and (3) the required actions for various conditions of inoperability. These proposed changes have no affect upon design basis. The proposed clarification of the specifications and basis are consistent with those established in Improved Standard Technical Specifications (Reference 10.1 and 10.2).

The following changes are proposed:

1. A new limiting condition for operation, [2.1.4(1)a], for "No pressure boundary leakage" is being added. This is the exact statement from the ISTS 3.4.13. This specification will have the same completion time as provided in ISTS. This proposed change will assure that design criteria of no RCS pressure boundary leakage is maintained. This is considered an

acceptable change as this is a new and more restrictive requirement than presently contained in the FCS Technical Specifications.

2. The present limiting conditions for operation for unidentified, identified, and primary to secondary leakage limits are being combined and reformatted in a table, [2.1.4(1)b through d], as in ISTS 3.4.13. These values remain unchanged from present specification limits. This proposed change is considered an acceptable change as it is equivalent to the limits presently contained in the FCS Technical Specifications and is as recommended in the ISTS.

3. The allowed outage time for each of the existing separate specifications addressing RCS leakage are being combined into one allowed outage time of four (4) hours [2.1.4(2)]. The present specification does not specify the time to "identify" the leakage. The present specification allows up to twelve (12) hours to "identify" the leakage and then be in the hot shutdown condition for unidentified or identified leakage. The proposed specification is equal to the present requirement of four (4) hours for primary to secondary leakage. As recommended in the ISTS there is no allowed outage time for RCS pressure boundary leakage. This completion time allows time to verify leakage rates and either identify the leakage or reduce the leakage to within acceptable limits. In that this time (4 hours) is being taken from the completion time to place the plant in the hot shutdown condition, this proposed change is considered an acceptable change as it is a more restrictive requirement than presently contained in the FCS Technical Specifications. This specification will have the same completion time as provided in ISTS.

4. The required actions for each of the existing separate specifications are also being combined into one specification [2.1.4(3)]. This proposed specification reduces the time to be in hot shutdown (Mode 3 for FCS) from 12 hours to 6 hours. This completion time is based upon operating experience to reach the required conditions from full power in an orderly manner and without challenging plant systems and is as specified in ISTS. This proposed change is considered an acceptable change as it is a more restrictive requirement than presently contained in the FCS Technical Specifications. This specification will have the same completion time as provided in ISTS.

5. The time to be in the cold shutdown condition, [2.1.4(3)], is being changed to be consistent with the recommendations of ISTS and will have the same completion time as provided in ISTS. This completion time is based upon operating experience to reach the required conditions from full power in

an orderly manner and without challenging plant systems. This proposed change is considered an acceptable change as it will minimize incorrect interpretations, is as recommended in the ISTS, and has been previously accepted by the NRC as indicated in Section 9.0, Precedence, of this evaluation.

6. A definition for leakage is also being added to the definitions section of the Technical Specifications. This definition is identical to the definition for leakage provided in ISTS (Reference 10.1). The addition of the definition for leakage will assure consistent and correct interpretation of leakage and supports the above changes. This proposed change is considered an acceptable change as it is administrative and is as recommended in the ISTS.

7. The basis for this specification is also being changed to be consistent with ISTS. These changes replace the existing basis associated with this specification with a basis similar in format and content of the ISTS. These changes are being made, primarily, to help explain the new ISTS format and provide the operator with a better understanding of the limits, allowed outage time, completion time, and reason for the specifications. These proposed changes have no affect upon design basis. The proposed clarification of the basis is consistent with those established in the ISTS (Reference 10.1 and 10.2).

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Omaha Public Power District has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed changes to Technical Specifications 2.1.4 establish a limit on reactor coolant system pressure boundary leakage and provide an allowed outage time and actions required for restoring operability. The proposed Technical Specifications address the regulatory requirements for equipment required for FCS Design Criterion 16 (similar to 10 CFR 50 GDC 30). The change will ensure that proper Limiting Conditions for Operation are entered for equipment or functional inoperability. There are no physical alterations

being made to the reactor coolant system or related systems. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes will not result in any physical alterations to the reactor coolant system, any plant configuration, systems, equipment, or operational characteristics. There will be no changes in operating modes, or safety limits, or instrument limits. With the proposed changes in place, Technical Specifications will retain requirements for the reactor coolant system. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed changes clarify the regulatory requirements for the reactor coolant system as defined by FCS Design Criterion 16 (similar to 10 CFR 50 GDC 30). The times established are within those invoked by the present Technical Specifications or equal to those previously reviewed and approved for use by the NRC. The proposed changes will not alter any physical or operational characteristics of the reactor coolant system and associated systems and equipment. Therefore, the proposed changes do not involve a reduction in a margin of safety.

Based on the above, Omaha Public Power District concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

Based on the above considerations, the proposed amendment does not involve and will not result in a condition which significantly alters the impact of Fort Calhoun Station on the environment. Thus, the proposed changes meet the eligibility criteria

for categorical exclusion set forth in 10 CFR Part 51.22(c)(9), and, pursuant to 10 CFR Part 51.22(b), no environmental assessment need be prepared.

9.0 PRECEDENCE

The proposed Technical Specifications are patterned after the Improved Standard Technical Specifications as described in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Reference 10.1. The NRC has approved specifications very similar to these proposed changes for Palisades Nuclear Power Plant and Calvert Cliffs Unit 1 and 2. The only significant difference in these specifications and ISTS is that these proposed specifications are not in the ISTS recommended "tabular" format. This proposed amendment includes the limits as provided in the Improved Standard Technical Specifications. These proposed specifications are also similar to Standard Technical Specifications as described in NUREG-0212, "Standard Technical Specifications for Combustion Engineering Plants," Reference 10.2

10.0 REFERENCES

10.1 NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants"
10.2 NUREG-0212, Revision 2, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors"
10.3 Title 10, Code of Federal Regulations, Part 50, Appendix B, Criterion 30
10.4 FCS USAR Appendix G, Criterion 16 "Monitoring Reactor Coolant Pressure Boundary"
10.5 Memorandum from Arthur T. Howell (NRC to William F. Kane (NRC), Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head Lessons-Learned Report, Dated September 30, 2002, (ADAMS Accession Number ML022740211)

ATTACHMENT 2

Markup of Technical Specification Pages

DEFINITIONS

E - Average Disintegration Energy

E is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System,

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits

Applicability

Applies to the leakage rates of the reactor coolant system whenever the reactor coolant temperature (T_{cold}) is greater than 210 °F.

Objective

To specify limiting conditions of the reactor coolant system leakage rates.

Specifications

To assure safe reactor operation, the following limiting conditions of the reactor coolant system leakage rates must be met:

(1) RCS operational LEAKAGE shall be limited to:

a. No Pressure Boundary LEAKAGE,
b. 1 gpm unidentified LEAKAGE,
c. 10 gpm identified LEAKAGE,
d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

(2) If RCS LEAKAGE limits of (1), above, are not met for reasons other than Pressure Boundary LEAKAGE, then reduce LEAKAGE to meet limits within 4 hours.

(3) If the Required Action and associated completion time of (2), above, is not met, OR Pressure Boundary LEAKAGE exists, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

~~(1) If the reactor coolant system leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the hot shutdown condition. If the source leakage exceeds 1 gpm and is not identified within 24 hours, the reactor shall be placed in the cold shutdown condition.~~

~~(2) If leakage exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 12 hours. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the cold shutdown condition.~~

~~(3) Primary-to-secondary leakage through the steam generator tubes shall be limited to 150 gallons per day per steam generator and 300 gallons per day total for both steam generators. When primary-to-secondary leakage has been determined to be in excess of the limit, the leakage rate shall be reduced to within limits in 4 hours or the reactor shall be placed in the cold shutdown condition within the next 36 hours.~~

(4) To determine leakage to the containment, a containment atmosphere radiation monitor (gaseous or particulate) or dew point instrument, and a containment sump level instrument must be operable.

a. With no containment sump level instrument operable, verify that a containment atmosphere radiation monitor is operable, and restore the containment sump level instrument to operable status within 30 days.

b. With no containment atmosphere radiation monitor and no dewpoint instrument operable, restore either a radiation monitor or dewpoint instrument to operable status within 30 days.

c. With only the dewpoint instrument operable, or with no operable instruments, enter Specification 2.0.1 immediately.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

(5) To determine leakage to the secondary system one of the following must be operable:

a. Steam Generator Blow Down Radiation Sample Instrument
b. Condenser Off Gas Radiation Monitor
c. Periodic Secondary Samples Analyzed for Activity

Basis

Background

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

FCS Design Criteria (Ref. 2), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

Applicable Safety Analysis

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The safety analysis assumes a 1 gpm primary to secondary leak as the initial condition. The Technical Specification requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the initial condition for the safety analysis

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day limit on primary to secondary LEAKAGE through any one SG is based upon guidance in NEI 97-06, *Steam Generator Program Guidelines*.

APPLICABILITY

The potential for RCPB LEAKAGE is greatest when the RCS is pressurized, that is, when the reactor coolant temperature (T_{cold}) is greater than 210 °F.

In MODES 4 and 5, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

REQUIRED ACTIONS (2)

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to meet limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

REQUIRED ACTIONS (3)

If any pressure boundary LEAKAGE exists or if unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to meet limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3, Hot Shutdown, within 6 hours and to MODE 4, Cold Shutdown, within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

~~Leakage directly into the containment indicates the possibility of a breach in the reactor coolant envelope. The limit is held low to minimize the chance of a crack progressing to an unsafe condition without detection and proper evaluation.~~

~~When the source of leakage can be identified, the situation shall be evaluated to determine if operation can safely continue. This evaluation will be reviewed by the Plant Review Committee and will be documented in writing and approved by the Plant Manager. Under these conditions, a maximum allowable reactor coolant leakage rate of 10 gpm has been established. This does not include the reactor coolant pump seal leak off that is piped to the volume control tank, which is not considered "leakage" from the reactor coolant system. A reactor coolant leakage to the containment atmosphere greater than 10 gpm would be indicative of seal and packing failures of sufficient magnitude to warrant shutdown for repair.~~

~~The maximum reactor coolant leakage rate of 10 gpm is within the 40 gpm capacity of one charging pump which would be available even under a loss-of-off-site power condition. Leakage from the reactor coolant system can be detected by monitoring one or a combination of reactor coolant system inventory, containment building radiation level, condenser offgas, steam generator blowdown water, containment dewpoint, and containment sump level (LT-599 or LT-600).[1,2] The containment atmosphere gaseous and particulate monitors are capable of detecting a one gpm leak from the reactor coolant system to containment within four hours of leak initiation following Regulatory Guide 1.45 criteria. The capability to detect a one gpm RCS leak within 4 hours is required in order to credit leak-before-break methodology. If reactor coolant leakage is to another closed system, it can be detected by the plant radiation monitors or by inventory control.~~

~~Placing the reactor in hot shutdown within 12 hours provides adequate time to arrange for an orderly reduction of power on the plant. The hot shutdown condition allows personnel to enter the containment and to inspect the pressure boundary for leaks. The 24 hours allowed prior to going to a cold shutdown condition allows reasonable time to correct small deficiencies. If major repairs are needed, a cold shutdown condition would be in order.~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.4 Reactor Coolant System Leakage Limits (Continued)

~~Limiting primary to secondary leakage is important to ensure steam generator tube integrity. The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day through any one steam generator or 300 gallons per day total). The safety analysis assumes a 1 gpm primary to secondary leak as the initial condition. The Technical Specification requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the initial condition for the safety analysis. This limit is based on industry operating experience as an indication of one or more propagating tube leak mechanisms. This leakage rate provides reasonable assurance against tube burst at normal and faulted conditions and provides reasonable assurance that flaws will not propagate to burst prior to detection by leakage monitoring and commencement of plant shutdown. Operating plants have demonstrated that primary-to-secondary leakage of 150 gallons per day can readily be detected by radiation monitors. Leakage from any one steam generator in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.~~

References

(1) USAR, Section 11.2.3

(2) USAR, Page G.16-1

ATTACHMENT 3

Proposed

Technical Specification Pages

DEFINITIONS

E - Average Disintegration Energy

E is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System,

DEFINITIONS

b. Unidentified LEAKAGE
All LEAKAGE (except RCP seal leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE
LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.4 Reactor Coolant System Leakage Limits

Applicability

Applies to the leakage rates of the reactor coolant system whenever the reactor coolant temperature (T_{cold}) is greater than 210 °F.

Objective

To specify limiting conditions of the reactor coolant system leakage rates.

Specifications

To assure safe reactor operation, the following limiting conditions of the reactor coolant system leakage rates must be met:

(1) RCS operational LEAKAGE shall be limited to:

a. No Pressure Boundary LEAKAGE,
b. 1 gpm unidentified LEAKAGE,
c. 10 gpm identified LEAKAGE,
d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

(2) If RCS LEAKAGE limits of (1), above, are not met for reasons other than Pressure Boundary LEAKAGE, then reduce LEAKAGE to meeet limits within 4 hours.

(3) If the Required Action and associated completion time of (2), above, is not met, OR Pressure Boundary LEAKAGE exists, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

(4) To determine leakage to the containment, a containment atmosphere radiation monitor (gaseous or particulate) or dew point instrument, and a containment sump level instrument must be operable.

a. With no containment sump level instrument operable, verify that a containment atmosphere radiation monitor is operable, and restore the containment sump level instrument to operable status within 30 days.

b. With no containment atmosphere radiation monitor and no dewpoint instrument operable, restore either a radiation monitor or dewpoint instrument to operable status within 30 days.

c. With only the dewpoint instrument operable, or with no operable instruments, enter Specification 2.0.1 immediately.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

(5) To determine leakage to the secondary system one of the following must be operable:

a. Steam Generator Blow Down Radiation Sample Instrument
b. Condenser Off Gas Radiation Monitor
c. Periodic Secondary Samples Analyzed for Activity

Basis

Background

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

FCS Design Criteria (Ref. 2), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.4 Reactor Coolant System Leakage Limits (Continued)

Applicable Safety Analysis

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The safety analysis assumes a 1 gpm primary to secondary leak as the initial condition. The Technical Specification requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the initial condition for the safety analysis

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day limit on primary to secondary LEAKAGE through any one SG is based upon guidance in NEI 97-06, *Steam Generator Program Guidelines*.

APPLICABILITY

The potential for RCPB LEAKAGE is greatest when the RCS is pressurized, that is, when the reactor coolant temperature (T_{cold}) is greater than 210 °F.

In MODES 4 and 5, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

REQUIRED ACTIONS (2).

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to meet limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

REQUIRED ACTIONS (3)

If any pressure boundary LEAKAGE exists or if unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to meet limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3, Hot Shutdown, within 6 hours and to MODE 4, Cold Shutdown, within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

References

(1) USAR, Section 11.2.3

(2) USAR, Page G.16-1

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