Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

10 CFR 50.90

LIC-14-0041
April 30, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

SUBJECT: License Amendment Request (LAR) 14-03, Proposed Change to Technical Specification 3.2, Table 3-5

Pursuant to 10 CFR 50.90, the Omaha Public Power District (OPPD) hereby requests an amendment to the Renewed Facility Operating License No. DPR-40 for Fort Calhoun Station (FCS), Unit No. 1. The proposed amendment will change Technical Specifications (TS) to add a new surveillance requirement similar to standard TS to verify the correct position of the valves required to restrict flow in the high pressure safety injection system.

OPPD has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The enclosure contains a description of the proposed changes, the supporting technical analyses, and the significant hazards consideration determination. Attachment 1 of the enclosure provides the existing TS pages marked-up to show the proposed changes. Attachment 2 of the enclosure provides the retyped (clean) TS pages.

OPPD requests approval of the proposed amendment by September 30, 2015. Once approved, the amendment shall be implemented within 120 days of issuance.

There are no regulatory commitments associated with this proposed change.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Nebraska official.

If you should have any questions regarding this submittal or require additional information, please contact Mr. Bill R. Hansher at (402) 533-6894.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 30, 2014.

Michael J. Prospero
Plant Manager

MJP/brh

Enclosure: OPPD's Evaluation of the Proposed Change

c: M. L. Dapas, NRC Regional Administrator, Region IV
J. W. Sebrosky, NRC Senior Project Manager
J. C. Kirkland, NRC Senior Resident Inspector
Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska

OPPD's Evaluation of the Proposed Change

License Amendment Request (LAR) 14-03, Proposed Change to Technical Specification 3.2, Table 3-5

ATTACHMENTS:

1. Technical Specification Page Markups
2. Retyped ("Clean") Technical Specifications Pages

1.0 SUMMARY DESCRIPTION

License amendment request (LAR) 14-03 proposes a change to the Renewed Facility Operating License No. DPR-40 for Fort Calhoun Station (FCS), Unit No. 1. Specifically, the Omaha Public Power District (OPPD) proposes to change Technical Specification 3.2, Table 3-5 to add a new surveillance requirement to verify the position of valves required to be throttled in the high pressure safety injection system and will be performed on a refueling frequency.

2.0 DETAILED DESCRIPTION

The proposed TS change for LAR 14-03 is based on NUREG-1432 Revision 4, *Standard Technical Specifications, Combustion Engineering Plant,* Surveillance Requirement (SR) 3.5.2.9. This SR requires that each emergency core cooling system valve position stop be verified to be in the correct position on an 18 month frequency.

Item 25, a new SR is proposed for TS 3.2, Table 3-5:

Item	*Test*		*Frequency*
25. HPSI Throttle Valves	*Verify, for each HPSI throttle valve listed below, each position stop is in the correct position.*		*R*
	HCV-311	*HCV-312*	
	HCV-314	*HCV-315*	
	HCV-317	*HCV-318*	
	HCV-320	*HCV-321*	

The Bases of TS 3.2 will be revised as follows to reflect information from the Bases of NUREG-1432, Revision 4:

Table 3-5, Item 25 verifies the proper position of stops on high pressure safety injection system valves. The valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The refueling frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillance were performed with the reactor at power.

The proposed Bases change is provided for information only.

3.0 TECHNICAL EVALUATION

Section 6.2 of the Fort Calhoun Station, Unit No. 1 (FCS) Updated Safety Analysis Report (USAR) describes the design basis of the safety injection system, which includes three (3) high pressure safety injection (HPSI) pumps.

The safety injection (SI) system is designed to prevent fuel and cladding damage that could interfere with adequate emergency core cooling. The SI system limits the cladding-water reaction to less than approximately 1 percent for all break sizes in the primary system piping up to and including the double-ended rupture of the largest reactor coolant pipe, for any break location, and for the applicable break time.

The SI system also provides rapid injection of borated water for added shutdown capability during rapid cooldown of the reactor coolant system caused by a rupture of a main steam line. No fuel damage would result from this accident with SI system operation, even with the most reactive control element assembly stuck in its fully withdrawn position. The system requirements during a main steam line rupture are discussed in USAR, Section 14.12, "Main Steam Line Break Accident."

The system requirements during a design basis large break loss-of-coolant accident (LOCA) are met with the assumption of three (3) of the four (4) safety injection tanks delivering borated water to the core, and one (1) HPSI pump and one low pressure safety injection (LPSI) pump each delivering approximately 75 percent of their rated flow to the core.

HPSI pumps are credited in USAR, Section 6.2 to respond to design basis accidents, including a large break LOCA. The HPSI pumps must be able to operate reliably to supply sufficient water for core cooling.

As reported in Licensee Event Report (LER) 2013-003-01 (Reference 6.1), HPSI pump operation was identified as unreliable when operated in the extended flow region of the manufacturer's pump curve during an accident. The available net positive suction head was identified as insufficient to support pump operation in the extended flow region. During system benchmark testing, it was identified that the safety injection loop flows were imbalanced. Three (3) valves were identified as having higher flow coefficients (Cv) values than specified in the original design specifications. As a result, HPSI discharge header isolation valve position has been set to balance safety injection flow into the four reactor coolant system loops. Valves in the Low Pressure Safety Injection flowpath do not have to be positioned to less than full-open and therefore are not included in the proposed surveillance requirement.

In Reference 6.2, the NRC noted that many plants have valves in the lines to each injection point that have electrical or mechanical stops, which have been adjusted to ensure flow requirements are satisfied. Reference 6.2 states:

In view of the safety function associated with the proper setting of valves used to throttle flow in these systems, we consider it appropriate that periodic verification be made of these valve positions. Accordingly, we request that you determine if throttle valves are used to obtain the required flow distribution in the HPSI or LPSI systems. If throttle valves are used, we request that you propose changes to your technical specifications to incorporate the surveillance requirements given in the enclosure.

As such, the proposed change will add a new surveillance requirement similar to standard TS to verify the correct position of the valves required to restrict flow. The valves to be verified are HCV-311/HCV-312, HPSI to Reactor Coolant Loop 1B Isolation valves, HCV-314/HV-315, HPSI to Reactor Coolant Loop 1A Isolation valves, HCV-317/HCV-318, HPSI to Reactor Coolant Loop 2A Isolation valves, and HCV-320/HCV-321, HPSI to Reactor Coolant Loop 2B Isolation valves.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Regulations

General Design Criteria:

Fort Calhoun Station was licensed for construction prior to May 21, 1971, and at that time committed to the draft General Design Criteria (GDC). The draft GDC are contained in Appendix G of the FCS USAR and are similar to 10 CFR 50, Appendix A, *General Design Criteria for Nuclear Power Plants.*

CRITERION 44 – EMERGENCY CORE COOLING SYSTEMS CAPABILITY

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

This criterion is met. The emergency core cooling systems are covered in detail in USAR Section 6.

There are three (3) systems; the high pressure injection system, which uses pumps; the low pressure injection system, which also uses pumps, and the stored energy tanks, or accumulators. These systems are designed to meet the criteria stated above in reference to the prevention of fuel and clad damage (cf. USAR Section 6.2.1). The systems do not share active components other than the valves controlling the suction headers of the high and low pressure safety injection pumps. These valves are in no way associated with the function of the stored energy tanks. The partial loss of capacity due to failure of one of these valves to function automatically will not compromise system effectiveness.

CRITERION 45 – INSPECTION OF EMERGENCY CORE COOLING SYSTEMS

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles.

This criterion is met. USAR Section 6.2.3 describes the arrangement and location of the main components in the emergency core cooling system. All pumps, the shutdown heat exchangers, and valves and piping external to the containment building are accessible for physical inspection at any time. All safety injection valves and piping inside the containment building, and the stored energy tanks, can be inspected whenever access to the containment building is possible.

The reactor vessel internals, reactor coolant piping and items such as the water injection nozzles are accessible for physical inspection.

CRITERION 46 – TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

This criterion is met. All active components in the emergency core cooling systems can be tested periodically for operability and functional performance. The high pressure and low pressure safety injection pumps and the motor operated valves can be tested, with the reactor at operating pressure, by flow recirculation. Two recirculation paths are available. One path, through the pump recirculation lines, verifies the functional performance of the pumps. The second path, through the leakage coolers, verifies flow path continuity and provides a check on the operability of the motor operated valves and all check valves except the final check valve in each safety injection header.

During normal plant cooldown, the low pressure pumps function as shutdown cooling pumps and feed the reactor coolant system via the safety injection headers. This operation verifies that the final check valve in each header is operable.

Operation of the check valve in the stored energy tank discharge line can be verified by bleeding through the leakage cooler and the recirculation line. Cf. USAR Section 6.2.7.

CRITERION 47 – TESTING OF EMERGENCY CORE COOLING SYSTEMS

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as practical.

This criterion is met. The emergency core cooling systems are designed to permit periodic testing of their delivery capability up to the reactor coolant pipe.

The low pressure safety injection pumps will be used as shutdown cooling pumps during normal plant cooldown. The pumps will discharge into the safety injection header via the shutdown heat exchangers and the low pressure injection lines.

With the plant at operating pressure, the high pressure safety injection pumps and valves may be operated so as to discharge through the leakage coolers back to the safety injection and refueling water (SIRW) tank. This will verify flow path continuity in all high pressure injection lines.

Borated water from the stored energy tanks may be bled through the drain heat exchanger and the recirculation line to verify flow path continuity from each tank to its associated main safety injection header.

The proposed change does not impact the ability to comply with USAR Appendix G, Criteria 44, 45, or 46.

4.2 Precedent

4.2.1 Letter from NRC (G. Lear) to OPPD (T. E. Short) dated June 30,1977 (NRC-77-0060)

4.2.3 NUREG-1432 Revision 4, *Standard Technical Specifications, Combustion Engineering Plant,* Surveillance Requirement 3.5.2.9.

4.3 Significant Hazards Consideration

The proposed change would change Technical Specification (TS) 3.2, Table 3-5 to add a new surveillance requirement to verify the correct position of valves on a refueling frequency consistent with Standard TS.

The Omaha Public Power District (OPPD) 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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to Technical Specification (TS) 3.2 Table 3-5 would add a new surveillance requirement to verify the position of valves required to restrict flow in the high pressure safety injection system to ensure adequate flow is maintained following a design basis accident.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because: 1) the proposed amendment does not represent a change to the system design, 2) the proposed amendment does not alter, degrade, or prevent action described or assumed in any accident Updated Safety Analysis Report (USAR) from being performed, 3) the proposed amendment does not alter any assumptions previously made in evaluating radiological consequences, and 5) the proposed amendment does not affect the integrity of any fission product barrier. No other safety related equipment is affected by the proposed change.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change adds a surveillance requirement to verify the position of valves. The proposed change does not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Hence, the proposed change does not introduce any new accident initiators, nor does it reduce or adversely affect the capabilities of any plant structure or system in the performance of their safety function.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The proposed change to add a new surveillance requirement does not alter the manner in which safety limits or limiting safety system settings are determined. The safety analysis acceptance criteria are not affected by this proposed change. Further, the proposed change does not change the design function of any equipment assumed to operate in the event of an accident. The change only adds a requirement to periodically verify the position of valves.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, OPPD 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.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), *Licensee Event Report 2013-003, Revision 1, for the Fort Calhoun Station,* dated November 27, 2013 (LIC-13-0171)

6.2 Letter from NRC (G. Lear) to OPPD (T. E. Short) dated June 30,1977 (NRC-77-0060)

6.3 Letter from OPPD (T. E. Short) to NRC (G. E. Lear) dated August 22, 1977 (LIC-77-0090)

Technical Specification

And

Technical Specification Bases

Page Markups

[Word-processor mark-ups using "double underline/~~strikeout~~" feature for "new text/deleted text" respectively.]

3.0 SURVEILLANCE REQUIREMENTS

3.2 Equipment and Sampling Tests (continued)

Table 3-5, Item 8b verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this surveillance requirement is not met, compliance with LCO 3.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of daily is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

Table 3-5, Item 25 verifies the proper position of stops on high pressure safety injection system valves. The valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The refueling frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillance were performed with the reactor at power.

References

1) USAR, Section 9.10
2) ASTM D4057, ASTM D975, ASTM D4176, ASTM D2622, ASTM D287, ASTM 6217, ASTM D2709
3) ASTM D975, Table 1
4) Regulatory Guide 1.137
5) EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency
19.	Refueling Water Level	Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21.	Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.
24.	Spent Fuel Cask Loading	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11.	Prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool.
25.	HPSI Throttle Valves	Verify, for each HPSI throttle valve listed below, each position stop is in the correct position. HCV-311 HCV-312 HCV-314 HCV-315 HCV-317 HCV-318 HCV-320 HCV-321	R

Retyped ("Clean") Technical Specification

And

Technical Specification Bases Pages

3.0 SURVEILLANCE REQUIREMENTS

3.2 Equipment and Sampling Tests (continued)

Table 3-5, Item 8b verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this surveillance requirement is not met, compliance with LCO 3.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of daily is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

Table 3-5, Item 25 verifies the proper position of stops on high pressure safety injection system valves. The valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The refueling frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillance were performed with the reactor at power.

References

1) USAR, Section 9.10
2) ASTM D4057, ASTM D975, ASTM D4176, ASTM D2622, ASTM D287, ASTM 6217, ASTM D2709
3) ASTM D975, Table 1
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TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency
19.	Refueling Water Level	Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21.	Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.
24.	Spent Fuel Cask Loading	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11.	Prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool.
25.	HPSI Throttle Valves	Verify, for each HPSI throttle valve listed below, each position stop is in the correct position. HCV-311 HCV-312 HCV-314 HCV-315 HCV-317 HCV-318 HCV-320 HCV-321	R