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An Exelon Company

RS-06-079

10 CFR 50.90

June 30, 2006

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Request for Technical Specification Change to Revise Surveillance Requirement Note

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) hereby requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed change revises the Note preceding TS Surveillance Requirement (SR) 3.4.6.1 to be consistent with the wording in NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Revision 3. Specifically, the Note will be revised to read, "Not required to be performed in MODE 3." The current wording of this Note differs from the Standard TS wording and is not consistent with the required Modes for performance of this Surveillance Requirement. This proposed revision will support the performance of this SR during the required Modes and will provide the necessary clarification to ensure that the intent of the SR is met.

Attachment 1 to this letter provides the evaluation of the proposed change to the Note preceding TS SR 3.4.6.1. Attachment 2 provides a copy of the marked up TS page and Attachment 3 contains copies of the marked up TS Bases pages provided for information only.

AmerGen is requesting approval of this change by June 30, 2007 with implementation within 60 days of issuance of the amendment.

There are no regulatory commitments contained in this letter.

This proposed change has been reviewed by the CPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the Quality Assurance Program.

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AmerGen is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of June 2006.

Respectfully,

Keith R. Jury

Keith R. Jury
Director – Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specification Page
3. Markup of Proposed Technical Specification Bases Page (For Information Only)

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

Subject: Request for Technical Specification Change to Revise Surveillance Requirement Note

1.0 DESCRIPTION

This is a request from AmerGen Energy Company, LLC (AmerGen) to amend Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed change revises the Note preceding TS Surveillance Requirement (SR) 3.4.6.1 to be consistent with the wording of this Note in the Standard TS (Reference 1). The current wording of this Note differs from the Standard TS wording and is not consistent with the required Modes for performance of this Surveillance Requirement. This proposed revision will support the performance of this SR during the required Modes and will provide the necessary clarification to ensure that the intent of the SR is met.

2.0 PROPOSED CHANGES

The proposed change revises the Note to TS SR 3.4.6.1 to read as follows.

"Not required to be performed in MODE 3."

In addition, the TS Bases will be revised to document the basis for the proposed Note. The Bases changes will be implemented in accordance with the CPS TS Bases Control Program defined in TS 5.5.11.

3.0 BACKGROUND

TS Section 3.4.6, "Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage," requires the leakage from each RCS PIV to be within limit. RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB). The function of RCS PIVs is to separate the high pressure RCS pressure boundary from an attached low pressure system. This protects the RCS pressure boundary described in 10 CFR 50.2, "Definitions," 10 CFR 50.55a, "Codes and standards," paragraph (c), and Criterion 55 of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." This limiting condition for operation (LCO) allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. RCS PIV leakage is leakage into closed systems connected to the RCS. Although TS Section 3.4.6 provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of the connecting systems.

LCO 3.4.6 is applicable in Mode 1, "Power Operation," Mode 2, "Startup," and Mode 3, "Hot Shutdown," since the PIV leakage potential is greatest when the RCS is pressurized. In Mode 3, valves in the Residual Heat Removal (RHR) flowpath are not required to meet the requirements of this LCO when in, or during transition to or from, the RHR shutdown cooling mode of operation.

The purpose of TS SR 3.4.6.1 is to verify that leakage is below the specified limit and to identify each leaking valve. The specified leakage limit of 0.5 gallons per minute (gpm) per inch of nominal valve diameter up to 5 gpm maximum at an RCS pressure ≥ 1000 psig and ≤ 1025 psig applies to each valve. Leakage testing requires a stable pressure condition.

SR 3.4.6.1 is modified by a Note that states the leakage surveillance is only required to be performed in Modes 1 and 2. However, the CPS procedure that implements SR 3.4.6.1 indicates that the plant is required to be in Mode 4, "Cold Shutdown," or Mode 5, "Refueling," to conduct this surveillance. As a result, performance of this procedure could be considered to be in conflict with the Note to SR 3.4.6.1. While the Note indicates that performance of SR 3.4.6.1 is only required to performed in Modes 1 and 2, it does not preclude performance of this SR in Modes 4 or 5. However, to provide clarification and ensure this SR is performed in accordance with the intent of the surveillance, AmerGen proposes that this Note be revised to be consistent with the wording provided in the applicable Improved Standard TS (Reference 1).

4.0 TECHNICAL ANALYSIS

As noted above, the primary purpose of this TS is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection. Various PIV configurations, leakage testing of the valves, and operational changes have been evaluated (Reference 2) to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

As stated in the TS 3.4.6 Bases, this LCO applies in Modes 1, 2, and 3 because the PIV leakage potential is greatest when the RCS is pressurized. In Mode 3, valves in the RHR flowpath are not required to meet the requirements of this LCO when in, or during transition to or from, the RHR shutdown cooling mode of operation. In Modes 4 and 5, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for LOCA outside the containment. Therefore, the LCO is not applicable in these modes of operation.

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. Leakage testing requires a stable pressure condition and SR 3.4.6.1 requires verification that the equivalent leakage from each RCS PIV is within the specified limits at an RCS pressure ≥ 1000 psig and ≤ 1025 psig. The ASME Operations and Maintenance Code (Reference 3), however, permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (i.e., the maximum pressure differential). In accordance with Reference 3, the

observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

The leakage testing at CPS is performed every 24 months (i.e., in accordance with the Inservice Testing Program) and is performed in Modes 4 or 5 to accommodate required access to areas of the plant not accessible during plant operation. Testing is performed using a test rig capable of producing the high pressures required by this test (i.e., greater than 1000 psig). Since CPS is capable of generating the required pressure to perform this surveillance independent of the reactor, the testing is performed in Modes 4 or 5. Hence, it is not necessary to perform the surveillance during modes of operation when the RCS is at the pressures required to verify leakage is within the specified limits. Performing this SR in Modes 4 or 5 allows access to the valves located in the drywell that are being tested. This ensures any necessary repairs can be made prior to entering a Mode in which the LCO is applicable. Testing in Modes 4 or 5 also precludes the potential for an unplanned transient if the surveillance were performed with the reactor at power. The proposed change to the Note preceding SR 3.4.6.1 will not alter the method for performing this testing, the limits the test results are evaluated against or the frequency at which the testing is performed.

Therefore, to gain the benefits associated with performing SR 3.4.6.1 in Modes 4 or 5, AmerGen proposes to revise the Note associated with this SR. By revising the Note to be consistent with the Note provided in Reference 1, entry into Mode 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower Modes. It also ensures that the SR can be performed in Modes 4 or 5. The current Note implies that the SR must be performed in Mode 1 or 2, which is not consistent with the intent of the TS. Revising this Note will ensure there is no confusion in determining when this SR must be performed.

In summary, performance of SR 3.4.6.1 in Modes 4 or 5 will preclude the potential for unplanned transients associated with performing the SR at power. It will also ensure that access to the valves being tested is available in the event that repairs are required. The proposed revision to the Note associated with SR 3.4.6.1 will provide clarification as to when this SR can be performed and will ensure that the performance of SR 3.4.6.1 can take place in Modes 4 or 5.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

AmerGen Energy Company (AmerGen), LLC is requesting a revision to the Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed change revises the Note preceding TS Surveillance Requirement (SR) 3.4.6.1 to be consistent with the wording in NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Revision 3. The current wording of this Note differs from the Standard TS wording and is not consistent with the desired Modes for performance of this Surveillance

Requirement. This proposed revision will support the performance of this SR during the desired Modes and will provide the necessary clarification to ensure that the intent of the SR is met.

AmerGen 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 amendment revises the note associated with TS SR 3.4.6.1, which requires verification that the leakage past the Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs) is less than a specified limit. The proposed revision provides clarification that performance of this SR is allowed during plant shutdown (i.e., a Mode other than Modes 1 and 2).

The proposed change does not require modification to the facility. The proposed change does not affect the operation of any facility equipment, the interface between facility systems, or the reliability of any equipment. In addition, the proposed change does not alter the requirement to perform the leakage testing of the RCS PIVs and does not revise the leakage limits associated with this SR. The function of the RCS PIVs is to separate the high pressure RCS from an attached low pressure system. Periodic testing of PIVs can substantially reduce intersystem Loss of Coolant Accident (LOCA) probability. Since the proposed change does not alter the method or limits associated with the leak rate testing of the RCS PIVs there is no significant increase in the probability of a LOCA. Therefore, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a previously analyzed event are dependent on the initial conditions assumed in the analysis, the availability and successful functioning of equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The method for performing the leakage testing of the RCS PIVs and the specified leakage limit for this testing will not change as a result of the proposed revision and, therefore, there is no change in the consequences associated with the LOCA analysis. The radiological consequences remain within applicable regulatory limits. The proposed change does not alter any system's performance measures or the ability to perform its accident mitigation functions. The radiological consequences associated with any previously evaluated accident do not change as a result of the proposed revision. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

Based on the above, 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 to the wording of the Note to TS SR 3.4.6.1 clarifies the plant conditions for when the surveillance is required to be performed. The proposed change does not affect the design, functional performance or operation of the facility. No new equipment is being introduced and installed equipment is not being operated in a new or different manner. Similarly, the proposed change does not affect the design or operation of any structures, systems or components involved in the mitigation of any accidents, nor does it affect the design or operation of any component in the facility such that new equipment failure modes are created. There are no setpoints at which protective or mitigative actions are initiated that are affected by this proposed action. No change is being made to procedures relied upon to respond to an off-normal event.

As such the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms or actions. The proposed change revises a note associated with a surveillance requirement to clarify the plant conditions for when the surveillance needs to be performed. This change involves an administrative clarification to reflect the original intent of the TS. The equipment will continue to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its intended safety function. There is no change in the design of the affected systems, no alteration of the setpoints at which alarms or actions are initiated, and no change in plant configuration from original design. There is no impact on the plant safety analyses.

Therefore, operation of CPS in accordance with the proposed change will not involve a significant reduction in a margin of safety.

Conclusion

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

5.2 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. AmerGen has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the Technical Specifications, and does not affect conformance with any General Design Criteria (GDC) differently than described in the CPS Updated Safety Analysis Report (USAR). The proposed change revises the Note associated with TS SR 3.4.6.1 to provide clarification as to when the surveillance is to be performed and does not change the intent of the SR to ensure that equivalent leakage from each RCS PIV is within the specified limit at the specified pressure.

CPS continues to test the PIVs for leakage to verify the RCS pressure boundary is protected in compliance with the requirements of 10 CFR 50.2, "Definitions," 10 CFR 50.55a, ""Codes and standards," paragraph (c), and Criterion 55 of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The test frequency required by the Inservice Testing Program does not change as a result of the proposed revision and continues to be in accordance with the ASME Code, Section XI, frequency requirement.

In conclusion, based on the considerations described 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 NRC 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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," 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, "Criterion for categorical exclusion; identification of

licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Volumes 1 and 2, Revision 3.0, dated June 2004

2. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," dated May 1, 1980

3. ASME Operations and Maintenance Code, Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants," Section 4.2.2.3(b)(4)

ATTACHMENT 2

Markup of Proposed Technical Specification Page

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Not ----------NOTE---------- ~~Only~~ required to be performed in MODE~~s~~ 3 ~~and 2~~. ---------- Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ 1000 psig and ≤ 1025 psig.	In accordance with Inservice Testing Program

ATTACHMENT 3

Markup of Proposed Technical Specification Bases Page
(For Information Only)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition. For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

The Frequency required by the Inservice Testing Program is within the ASME Code, Section XI, Frequency requirement and is based on the need to perform this Surveillance under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

3 Therefore, this SR is not modified by a Note that states the leakage Surveillance is ~~only~~ required to be performed in MODE~~S 1 and 2~~. Entry into MODE 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

With regard to leakage values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 9).

(continued)