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

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

Subject: Request for Amendment to Technical Specifications for Main Steam Line Leakage Limits

Reference:
(1) Letter from L. Rossbach (USNRC) to O. D. Kingsley (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. MA5754 and MA5755)," dated October 1, 1999
(2) Letter from T. Kim (USNRC) to R. O. Anderson (Northern States Power Company), "Monticello Nuclear Generating Plant – Issuance of Amendment Re: Main Steam Isolation Valve and 10 CFR Part 50, Appendix J, Leak Test Requirement (TAC No. M93332)," dated April 3, 1996

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (i.e., AmerGen), proposes a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed change is to TS Section 3.6.1.3, "Primary Containment Isolation Valves." Specifically, AmerGen proposes a change to Surveillance Requirement (SR) 3.6.1.3.9 to replace the existing leakage limit for any one main steam line as measured by main steam isolation valve (MSIV) leakage of ≤ 28 standard cubic feet per hour (scfh) with an aggregate leakage limit of ≤ 112 scfh for all main steam lines. The approach of using an aggregate value for main steam line leakage instead of an individual line leakage limit is consistent with NRC approved changes for the Dresden Nuclear Power Station, and the Monticello Nuclear Generating Plant, documented in the referenced letters.

The repair of an MSIV can represent significant worker dose and cost during an outage. An MSIV has historically cost about $80,000 to repair, workers can receive approximately 3.2 Rem of radiation dose and the work can take up to a day of outage time to complete. The accident analysis assumptions allow the change to be based on an aggregate leakage rate instead of the more restrictive limit on each main steam line with no reduction in safety, thereby allowing more efficient outage scheduling.

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Therefore, AmerGen respectfully requests approval of this change prior to February 15, 2002, in order to support preparation for the next refueling outage. This request is subdivided as follows.

1. Attachment A gives a description and safety analysis of the proposed change.

2. Attachment B includes the marked-up TS pages with the proposed change indicated. A marked-up copy of the affected TS Bases pages is also provided for information only.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (a)(1) which provides information supporting a finding of no significant hazards consideration in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (c).

4. Attachment D provides information supporting an Environmental Assessment.

This proposed change has been reviewed by the CPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board.

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

Should you have any questions concerning this letter, please contact Mr. J. L. Peterson at (217) 937-3418.

Respectfully,

J. M. Heffley
Vice President
Clinton Power Station

JLP/blf

Attachments:
Affidavit
Attachment A: Description and Safety Analysis for Proposed Change
Attachment B: Marked-up TS Pages for Proposed Change
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting An Environmental Assessment

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DEWITT	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	**Docket Number**
CLINTON POWER STATION, UNIT 1	)	**50-461**

SUBJECT: Request for Amendment to Appendix A, Technical Specifications for Main Steam Line Leakage Limits

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

J. M. Heffley
Vice President
Clinton Power Station

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 21 day of June, 2001.

" OFFICIAL SEAL "
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Jacqueline S. Matthias
Notary Public

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGES

A. SUMMARY OF THE PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (i.e., AmerGen), proposes a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed changes are to TS Section 3.6.1.3, "Primary Containment Isolation Valves." Specifically, AmerGen proposes a change to Surveillance Requirement (SR) 3.6.1.3.9 to eliminate the leakage limit for any one main steam line as measured by main steam isolation valve (MSIV) leakage of ≤ 28 standard cubic feet per hour (scfh) and to replace that with an aggregate leakage limit of ≤ 112 scfh for all main steam lines.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

SR 3.6.1.3.9 requires verification that the leakage rate through each main steam line is ≤ 28 scfh when tested at greater than or equal to the postulated peak accident pressure inside the primary containment (i.e., P_a). This is to be performed at a frequency in accordance with the Primary Containment Leakage Rate Testing Program.

C. BASES FOR THE CURRENT REQUIREMENTS

Two 24-inch Main Steam Isolation Valves (MSIVs) are welded in a horizontal run of each of the four main steam lines. The inboard MSIV is located inside the drywell adjacent to the main steam line guard pipe penetration. The outboard MSIV is located in the main steam tunnel of the Auxiliary Building as close as practical to the Containment. The radiological consequences analysis in the Updated Safety Analysis Report (USAR) Section 15.6.5, "Loss-of-Coolant Accidents (Resulting from Spectrum of Postulated Piping Break Within the Reactor Coolant Pressure Boundary) – Inside Containment," is based on a 28 scfh leakage rate per main steam line. Performance of TS SR 3.6.1.3.9 ensures that the leakage remains within the assumptions for the analysis. Compliance with this SR is only required in Modes 1 (i.e., "Power Operation"), 2 (i.e., "Startup"), and 3 (i.e., "Hot Shutdown"), since in the other Modes the reactor coolant system is not pressurized and primary containment leakage limits are not required.

D. NEED FOR REVISION OF THE REQUIREMENTS

A Boiling Water Reactor Owners' Group (BWROG) Study (Reference 1) found that disassembling and refurbishing MSIVs to meet the low leakage rate limits (i.e., ≤ 28 scfh), frequently contributes to repeated test failures. By not having to disassemble the valves and refurbish them for leakage, the licensee may avoid introducing a maintenance-induced failure that may cause recurring valve leakage problems that may lead to leakage rate test failures.

The repair of an MSIV can represent significant worker dose and cost during an outage. An MSIV has historically cost about $80,000 to repair and workers can receive approximately 3.2 Rem of radiation dose. The accident analysis assumptions allow the change to be based on an aggregate leakage rate instead of the more restrictive limit on

each main steam line with no reduction in safety, thereby allowing more efficient outage scheduling. Overhaul of an MSIV is not a normal activity on the outage critical path schedule. However, if both MSIVs in the same steam line failed to meet the leakage requirements for the steam line (i.e., $\leq$ 28 scfh), under the current requirements, both valves would be required to be repaired and retested, which could extend the critical path duration for the outage.

A review of maintenance history for the past three refueling outages has not shown any MSIV leak rate failures; however, the leak rate of the inboard MSIV on the "A" main steam line is approaching the current TS limit of 28 scfh. The leak rates of the other MSIVs in the other three main steam lines remain well within this limit.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed change to the maximum allowable main steam line leakage rate specified in SR 3.6.1.3.9 is from $\leq$ 28 scfh per line to a maximum pathway leakage of $\leq$ 112 scfh total for all four main steam lines, with no single valve or line limit.

The proposed change to SR 3.6.1.3.9 reads:

"Verify total leakage rate through all four main steam lines is $\leq$ 112 scfh when tested at $\geq P_a$."

The TS Bases for SR 3.6.1.3.9 will be revised as follows and is provided for information only.

"The analyses in References 1, 2, and 3 are based on leakage that is less than the specified leakage rate. Leakage through all four main steamlines must be $\leq$ 112 scfh when tested at Pa (9.0 psig). The MSIV leakage rate must be verified to be in accordance with the assumptions of References 1, 2, and 3. A Note is added to this SR which states that these valves are only required to meet this leakage limit in Modes 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and primary containment leakage limits are not required. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

With regard to the leakage rate value 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 (Reference 11)."

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The safety function of the MSIVs is to isolate the reactor coolant system in the event of a Loss of Coolant Accident (LOCA), or other events requiring containment or reactor coolant system isolation. Specifically, the MSIVs are assumed to isolate containment in the event of a large-break LOCA, which is a double-ended guillotine break of a recirculation system pipe, or a Main Steam Line Break. The large-break LOCA is the bounding accident based on dose consequence and assumes a total leakage through all the MSIVs of $\leq$ 112 scfh. This proposed change maintains the maximum allowed leakage from the Main Steam Lines assumed in the large-break LOCA event.

Therefore, the total release rate remains the same, and the radiological consequences to the public and the control room operators remain unchanged and within the guideline values contained in 10 CFR 100, "Reactor Site Criteria", and 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion 19, "Control Room."

G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed change regarding impact on any previous submittals, and have determined that there is no impact on any outstanding license amendment requests.

H. SCHEDULE REQUIREMENTS

We request approval of the proposed change prior to February 15, 2002, to support preparation for the next refueling outage.

I. REFERENCES

(1) NEDC-31858P, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 2, September 1993.
(2) Letter from L. Rossbach (USNRC) to O. D. Kingsley (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. MA5754 and MA5755)," dated October 1, 1999
(3) Letter from T. Kim (USNRC) to R. O. Anderson (Northern States Power Company), "Monticello Nuclear Generating Plant – Issuance of Amendment Re: Main Steam Isolation Valve and 10 CFR Part 50, Appendix J, Leak Test Requirement (TAC No. M93332)," dated April 3, 1996

Attachment B
Proposed Technical Specification Changes
Clinton Power Station, Unit 1

MARKED-UP TS PAGES FOR PROPOSED CHANGE

REVISED TS PAGES

3.6-19

REVISED BASES PAGES
(PROVIDED FOR INFORMATION ONLY)

B 3.6.27
B 3.6.28

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.08 L_a when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.9	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Verify **total** leakage rate through **all four** ~~each~~ main steam line**s** is ≤ **112** ~~28~~ scfh when tested at ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.10	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Verify combined leakage rate through hydrostatically tested lines that penetrated the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.8 (continued)

leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR.

The Frequency is consistent with the Primary Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria. Secondary containment bypass leakage is considered part of L_a.

A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

With regard to leakage rate 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).

SR 3.6.1.3.9

The analyses in References 1, 2, and 3 are based on leakage that is less than the specified leakage rate. Leakage through **all four** ~~each~~ main steamline**s** must be ≤ **112** ~~28~~ scfh when tested at P_a (9.0 psig). The MSIV leakage rate must be verified to be in accordance with the assumptions of References 1, 2, and 3. A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and primary containment leakage limits are not required. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

With regard to **the** leakage rate 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. 11).

SR 3.6.1.3.10

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 4 are met. The combined leakage rates (of 1 gpm times the total number of PCIVs when tested at $\geq$ 1.1 P_a) must be demonstrated at the frequency of the leakage test requirements of the Primary Containment Leakage Rate Testing Program.

This SR is modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

- With regard to leakage rate 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. 12).

(continued)

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92, "Issuance of Amendment," paragraph (c) a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

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

(2) Create the possibility of a new or different kind of accident from any previously analyzed; or,

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

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (i.e., AmerGen), proposes a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed change is to TS Section 3.6.1.3, "Primary Containment Isolation Valves." Specifically, AmerGen proposes a change to Surveillance Requirement (SR) 3.6.1.3.9 to eliminate the limit for any one main steam line as measured by main steam isolation valve (MSIV) leakage of $\leq$ 28 standard cubic feet per hour (scfh) and to replace that with an aggregate leakage limit of $\leq$ 112 scfh for all four main steam lines.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

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

The MSIVs are not initiators of or precursors to any of the accident scenarios presented in the Updated Safety Analysis Report. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The proposed change to the TS modifies the allowed main steam line leakage limit to an aggregate value (i.e., leakage for all four main steam lines combined) with no change to the currently allowed total leakage rate. This is the value currently used for calculation of dose consequences for the bounding accident for which MSIV closure is credited, the large-break loss of coolant accident (LOCA). This proposed change does not impact or increase the assumed radionuclide source term therefore; this change does not involve an increase in consequences of any accident previously evaluated.

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

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

The proposed change does not involve a change to the plant design or operation. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. The safety function of the MSIVs is to provide timely steam line isolation to mitigate the release of radioactive steam and limit reactor inventory loss under certain accident and transient conditions. Changing the leakage limits to include an aggregate value does not affect the isolation function performed by the MSIVs. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Does the change involve a significant reduction in a margin of safety?

The total allowed leakage rate for all four main steam lines remains unchanged at $\leq$ 112 scfh. The proposed change does not challenge the integrity of the fuel cladding, reactor coolant pressure boundary, or the primary containment.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage. As stated above, the total allowed leakage rate for all four main steam lines remains unchanged. In addition, there will not be a change in the types or amounts of any effluents released offsite. The radiological analyses remain unchanged and within the guidelines of 10 CFR 100, "Reactor Site Criteria," and 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion 19, "Control Room."

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

Therefore, based on the above evaluation, we have concluded that the proposed changes do not involve a significant hazards consideration.

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

AmerGen Energy Company, LLC (i.e., AmerGen) has evaluated this proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." AmerGen has determined that this proposed change meets the criteria for a 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), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or surveillance requirement, and the amendment meets the following specific criteria.

(i) The proposed changes involve no significant hazards consideration.

As demonstrated in Attachment C, this proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change allowing a total leakage rate limit for all Main Steam Lines instead of a single leakage rate limit for each line, is consistent with the design basis of the plant. As documented in Attachment A, there will be no significant increase in the amounts of any effluents released offsite. This proposed change does not result in an increase in power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed change will not affect the types or increase the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in changes in the configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from these changes.