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10CFR50.55a
10CFR50.90

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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 License Amendment and Relief Related to Excess Flow Check Valve Testing

References:
(1) General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000, for the Boiling Water Reactors Owners' Group
(2) NRC Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" (TAC Nos. MA7884 and M84809), dated March 14, 2000
(3) Browns Ferry Nuclear Plant, Units 2 and 3, Issuance of Amendment Regarding Revised Excess Flow Check Valve Surveillance Intervals (TAC Nos. MA6407 and MA6409), dated January 29, 2001
(4) Limerick Generating Station, Units 1 and 2, Issuance of Amendment Regarding Revised Excess Flow Check Valve Surveillance Requirements (TAC Nos. MA9927 and MA9928), dated February 23, 2001

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company LLC (i.e., AmerGen), is requesting a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed change relaxes the requirements of TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.12 to allow "a representative sample" of Excess Flow Check Valves (EFCVs) to be tested every refueling outage (i.e., 18 months), such that each EFCV will be tested nominally at least once every 10 years. Currently, TS SR 3.6.1.3.12 requires testing of each EFCV on an 18-month frequency. In addition, a request for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel

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from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a) (3), is also included that reflects the proposed change in EFCV testing.
Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.

As noted in Reference (1), earlier correspondence referred to Topical Report B21-00658-01. Both numbers (i.e., NEDO-32977-A and B21-000658-01) refer to the same Licensing Topical Report. The NRC approved the generic applicability of the Licensing Topical Report in March 2000 (Reference 2), as part of a plant-specific TS amendment request proposed by the Duane Arnold Energy Center. The proposed change to the CPS TS also requires a corresponding change to the applicable sections of the CPS Inservice Testing (IST) Program. The IST Program would then require that each refueling outage, a representative sample of EFCVs will be tested to satisfy the requirements of the ASME (B&PV) Code, Section XI, "Rules For Inservice Inspection Of Nuclear Power Plant Components."

The proposed TS change and relief are being requested to minimize personnel radiation exposure during refueling outages, reduce outage critical path time without significantly impacting the risk to the general public, and increase the availability of instrumentation during outages. The proposed change is similar to recent license amendment requests approved by the NRC for the Browns Ferry Nuclear Plant (Reference 3) and the Limerick Generating Station (Reference 4). We request approval of this proposed change prior to December 31, 2001, 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 and a marked-up copy of the affected TS Bases included for informational purposes.

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 using the standards in 10 CFR 50.92, "Issuance of amendment," paragraph (c).

4. Attachment D provides information supporting an Environmental Assessment.

5. Attachment E provides Relief Request 2203 involving the testing requirements for EFCVs.

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

AmerGen is notifying the State of Illinois of this request 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/krk

Attachments: Affidavit
Attachment A: Description and Safety Analysis for Proposed Change
Attachment B: Marked-Up Pages for Proposed Change
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E: Relief Request 2203 Involving Excess Flow Check Valve Testing Requirements

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) 50-461

SUBJECT: Request for License Amendment and Relief Related to Excess Flow Check Valve Testing

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 17th day of

April, 2001.

Notary Public

" OFFICIAL SEAL "
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

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). Specifically, AmerGen proposes to revise TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.12 to require testing of "a representative sample" of excess flow check valves (EFCVs) such that each EFCV will be tested nominally at least once every 10 years. Currently, TS SR 3.6.1.3.12 requires testing of each EFCV on an 18-month frequency. This proposed change is similar to previous changes that resulted in performance-based testing programs, such as Inservice Testing of snubbers and Option B to 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The basis for this change is consistent with the General Electric Nuclear Energy Topical Report, NEDO-32977-A, "Excess Flow Check Valves Testing Relaxation," (Reference 1) prepared for the Boiling Water Reactor Owners' Group (BWROG). The generic applicability of this Topical Report was approved by the NRC in a Safety Evaluation (SE) dated March 14, 2000 (Reference 2). This proposed change is also consistent with Standard Technical Specification Change Traveler TSTF-334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," proposed by the industry Technical Specification Task Force (TSTF). TSTF-334, Revision 2, was approved by the NRC on October 31, 2000. The Topical Report provides justification for a relaxation in the SR frequency. The report demonstrates through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

CPS TS require performance of surveillance tests on each EFCV every refueling outage (i.e., 18 months). TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," SR 3.6.1.3.12, requires a demonstration that each instrumentation line excess flow check valve is operable by verifying that the valve activates within the required flow range. This SR provides assurance that the instrumentation line EFCVs will perform as required to provide a primary containment barrier. This barrier functions, in certain multiple failure scenarios, to minimize the radiological consequences of an instrument line break outside primary containment. For instrument lines connected to the reactor coolant pressure boundary, the EFCVs serve as an additional flow restrictor to the orifices that are installed in the instrument lines that are inside the primary containment (i.e., drywell).

C. BASES FOR THE CURRENT REQUIREMENTS

The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within analyzed values. Primary containment isolation within the time limits specified for those PCIVs designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The operability requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the operability requirements provide assurance that the primary containment function assumed in the safety analysis will be maintained. Typically two isolation barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or in leakage that exceeds limits assumed in the safety analysis. One of these barriers may be other than a PCIV, such as a closed system, while other penetrations may be designed with only one barrier such as a welded closed spare penetration. The isolation devices addressed by the TS Limiting Condition for Operation (LCO) consist of either passive devices or active (i.e., automatic) devices. Manual valves that are closed or open in accordance with appropriate administrative controls, automatic valves that are de-activated and secured in their closed position, check valves with flow secured through the valve, and blind flanges that are in place, are considered passive devices. Check valves, such as EFCVs, and automatic valves, designed to close without operator action following an accident, are considered active devices. EFCVs are used in instrument lines to isolate a ruptured instrument line. The EFCV closes as a result of high flow in the instrument line.

The current 18-month surveillance frequency is based on the need to perform this testing under the conditions that apply during a plant outage and the potential for an unplanned transient if the testing were performed with the reactor at power.

D. NEED FOR REVISION OF THE REQUIREMENTS

The BWROG has developed a basis for relaxing the requirement to test each EFCV during each refueling outage. The change being requested is consistent with the NEDO-32977-A (Reference 1) and approved generically by the NRC by a letter dated March 14, 2000 (Reference 2) as part of a plant-specific TS amendment request submitted by the Duane Arnold Energy Center. In addition, other plants have recently received approval to implement this change. Specifically, on January 29, 2001, and February 23, 2001, the NRC approved similar requests for the Browns Ferry Nuclear Plant and the Limerick Generating Station, respectively (References 3, 4). The change proposed herein to the CPS TS is consistent with these submittals. The reduced testing associated with the proposed change will result in an increase in the availability of the instrumentation during outages, a savings in outage critical-path time, outage cost, and personnel dose, without significantly impacting plant operating safety.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed TS change is as follows.

1. TS SR 3.6.1.3.12 will be revised to verify that a representative sample of instrumentation line excess flow check primary containment isolation valves actuate with the required range every 18 months.

2. Changes to the TS Bases will be made consistent with the above change to TS Section 3.6.1.3. The representative sample will consist of an approximately equal number of EFCVs such that each EFCV will be tested nominally at least once every 10 years, consistent with Reference (1).

The proposed TS change is reflected on a marked-up copy of the affected page from the CPS TS contained in Attachment B. A marked-up copy of the affected pages from the current TS Bases is also provided in Attachment B. Following NRC approval of this request, we will revise the CPS TS Bases, in accordance with the TS Bases Control Program of TS Section 5.5.11, to incorporate the changes identified in Attachment B.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The proposed change will increase the test interval of the EFCVs. Reference (1) compares this situation to Option B of Appendix J to 10 CFR 50. The NRC revised Appendix J in 1995 by adding Option B which provides a risk-informed, performance-based approach to leakage rate testing of containment isolation valves. As discussed in Reference (2), the NRC accepted the test interval extension, which may be as great as 10 years, based on the EFCVs historically high reliability, their low risk significance, and the low radiological consequences should they fail.

CPS has a total of 22 EFCVs which serve as PCIVs, two of which are installed in instrumentation lines connected to the reactor coolant pressure boundary (RCPB). These EFCVs limit the release of fluid from the reactor coolant system in the event of an instrument line break. Instrument lines connected to the RCPB are equipped with a 3/8-inch flow-restricting orifice. The orifice size is selected by optimizing the minimum coolant release consistent with minimum effect on instrument response. The remaining 20 EFCVs are installed in instrument lines connected to the primary containment atmosphere or suppression pool, such as those that measure drywell pressure, or monitor the containment atmosphere, suppression pool water level, or ventilation system pressure. These EFCVs and the associated instrument lines are considered extensions of the primary containment. The EFCVs are used to provide primary containment isolation provisions consistent with the guidelines of Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11)."

Reference (1) states that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break outside of primary containment coincident with a design basis Loss of Coolant Accident (LOCA) would be of sufficiently low probability to be outside of the design basis. Reference (1) also provides detailed information about the results of EFCV testing at 12 Boiling Water Reactor (BWR) plants. Reference (1) determined an upper limit EFCV failure rate based on 12,424.5 valve operating years (i.e., 1.09E08 valve operating hours) with a plant average of 1035 valve operating years per plant. Considering the total number of EFCV failures (i.e., 11) out of 1.09E08 valve operating hours for the 12 plants, Reference (1) concluded that EFCVs had a low failure rate (i.e., 1.01E-07 failures per operating hour). In a similar representative time sample at CPS, (i.e., 220 valve operating years, or 1.93E06 valve operating hours for the 22 EFCVs at CPS), the only test failures to date have been due to testing methods. There have been no EFCV test failures resulting from a component failure at CPS.

The EFCV manufacturer type installed for all of the valves at CPS is Dragon. Table 4-2 of Reference (1) lists the best-estimate failure rate for the three plants using Dragon valves to be 9.2E-08 failures per operating hour. This data is based on two recorded failures out of 2.18E07 operating hours at the three BWR plants. It is concluded that Reference (1) bounds the reliability of CPS EFCVs.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in the CPS Updated Safety Analysis Report (USAR), Section 15.6.2, "Failure of Small Lines Carrying Primary Coolant Outside Containment." Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint, consequently, instrument line breaks are considered to be bounded specifically by the main steam line break. The failure of an EFCV, though not expected as a result of this proposed change, is bounded by the evaluation of an instrument line break.

Self-actuating EFCVs that are installed in low pressure instrument sensing lines (i.e., lines that sense drywell pressure, primary containment pressure, suppression pool water level, and ventilation system pressure) are designed to remain open if the instrument line integrity outside primary containment is lost during normal reactor operation. During normal reactor operation, there exists a small difference in atmospheric pressure between the drywell or primary containment and the secondary containment, the building where all EFCVs are located. If an instrument line outside primary containment ruptures during normal reactor operation, there will be insufficient differential pressure to actuate the EFCV. However, since there is negligible radiological source term available for release from inside the drywell or primary containment during normal reactor operation, the radiological consequences of a low pressure instrument sensing line failure is considered to be insignificant. Under accident conditions, the offsite radiological consequences from an EFCV failing to close will remain substantially below the guidelines of 10 CFR 100, "Reactor Site Criteria," will be bounded by the main steam line break analysis, and the integrity and functional performance of the secondary containment and its associated standby gas treatment system (SGTS) will be maintained.

In estimating the release frequency initiated by an instrument line break concurrent with an EFCV failure to close, two factors are considered: (1) the instrument line break frequency and (2) the probability of an EFCV failing to close. As discussed in the methodology provided in the Reference (2) NRC SE the BWROG assumed a single instrument line break frequency of 3.52E-05 per year. Thus, the product of this single instrument line break frequency and the total number of instrument lines with EFCVs (i.e., 22) results in a total instrument line break frequency of 7.74E-04 per year for CPS. Using a total plant EFCV failure frequency of 5.53E-03 per year as provided in the NRC SE results in a release frequency of 4.28E-06 per year for CPS. For the maximum surveillance testing interval of 10 years, a release frequency of about 2.57E-05 per year was estimated, which is an increase of 2.14E-05 per year from the current 18-month testing interval. This release frequency represents the increase in the total plant release frequency for a random break of any of the 22 instrument lines with EFCVs at CPS and a concurrent failure of the line's EFCV to isolate the break by closing. This release frequency is lower than the BWROG assumed instrument line break frequency of 3.52E-05 and is therefore bounded by the results of the Reference (1) Licensing Topical Report.

The reduced testing that would result from this proposed change will result in cost savings during outages, and personnel dose savings during outages without significantly impacting plant operating safety.

G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes 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 these proposed changes prior to December 31, 2001, to support preparation for the next refueling outage.

I. REFERENCES

(1) General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000, for the Boiling Water Reactor Owners' Group.
(2) NRC Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" (TAC Nos. MA7884 and M84809), dated March 14, 2000.
(3) Browns Ferry Nuclear Plant, Units 2 and 3, Issuance of Amendment Regarding Revised Excess Flow Check Valve Surveillance Intervals (TAC Nos. MA6407 and MA6409), dated January 29, 2001
(4) Limerick Generating Station, Units 1 and 2, Issuance of Amendment Regarding Revised Excess Flow Check Valve Surveillance Requirements (TAC Nos. MA9927 and MA9928), dated February 23, 2001

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

MARKED-UP TS PAGES FOR PROPOSED CHANGES

REVISED TS PAGES

3.6-19a

REVISED BASES PAGES
(PROVIDED FOR INFORMATION ONLY)

B 3.6-28a
B 3.6-28b

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	----------------NOTE---------------- Only required to be met in MODES 1, 2, and 3. -- Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ 3 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.3.12	Verify ~~each~~ **a representative sample of** instrumentation line excess flow check primary containment isolation valves actuates within the required range.	18 months

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11

This SR ensures that the combined leakage rate of the primary containment feedwater penetrations is less than the specified leakage rate. The leakage rate is based on water as the test medium since these penetrations are designed to be sealed by the FWLCS. The 3 gpm leakage limit has been shown by testing and analysis to bound the condition following a DBA LOCA where, for a limited time, both air and water are postulated to leak through this pathway. During the first hour following a DBA LOCA, the leakage is conservatively assumed to be entirely containment atmosphere. The feedwater check valves, 1B21-F010A(B) and 1B21-F032A(B), limit this leakage to the air equivalent of 3 gpm. During the remainder of the event, motor-operated valve(s) 1B21-F065A(B) assist to limit leakage in conjunction with the FWLCS.

The leakage rate of each primary containment feedwater penetration is assumed to be the maximum pathway leakage, i.e., the leakage through the worst of the three isolation valves [either 1B21-F010A(B), 1B21-F032A(B) or 1B21-F065A(B)] in each penetration. This provides assurance that the assumptions in the radiological evaluations of References 1 and 2 are met. Dose associated with leakage (both air and water) through the primary containment feedwater penetrations is considered to be in addition to the dose associated with all other secondary containment bypass leakage paths.

The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

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

SR 3.6.1.3.12

This SR requires a demonstration that ~~each~~ **a representative sample of** instrumentation line excess flow check valves (EFCV) is OPERABLE by verifying that the valve activates within the required flow range. **The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal).** This SR provides assurance that the instrumentation line EFCVs will perform as required to provide a second containment barrier. This second barrier functions in certain multiple failure scenarios to minimize the radiological consequences of an instrument line break (Ref. 7). **The nominal 10-year interval is based on other performance based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. EFCV test failures will be evaluated to determine if**

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.12 (continued)

additional testing in that test interval is warranted to ensure overall reliability is maintained. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 15).

For instrument lines connected to reactor coolant pressure boundary, the EFCVs serve as an additional flow restrictor to the orifices that are installed inside the drywell (Ref. 14). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

The operating limit or process parameter value associated with this SR, as read from plant indication instrumentation, is considered nominal. Instrument indications that are considered nominal do not require compensation for instrument indication uncertainties (Ref. 13).

REFERENCES

1. USAR, Chapter 15.6.5.
2. USAR, Section 15.6.4.
3. USAR, Section 15.7.4.
4. USAR, Section 6.2.
5. USAR, Table 6.2-47.
6. 10 CFR 50, Appendix J, Option B.
7. Regulatory Guide 1.11.
8. Calculation IP-0-0059.
9. Calculation IP-0-0056.
10. Calculation IP-0-0028.
11. Calculation IP-0-0063.
12. Calculation IP-0-0064.
13. Calculation IP-0-0065.
14. Calculation IP-M-0506
15. **NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.**

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.

AmerGen Energy Company, LLC (i.e., AmerGen), is requesting a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed change relaxes TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.12 to allow "a representative sample" of Excess Flow Check Valves (EFCVs) to be tested every refueling outage (i.e., 18 months), such that each EFCV will be tested nominally at least once every 10 years.

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 current Technical Specification (TS) Surveillance Requirement (SR) frequency requires each excess flow check valve (EFCV) to be tested every 18 months. The EFCVs at Clinton Power Station (CPS) are designed to remain open during normal operation, but will close automatically in the event of an instrument line break downstream of the valve following an accident. The proposed change allows a reduced number of EFCVs to be tested during each refueling outage. Industry operating experience demonstrates a high level of reliability for these EFCVs. A failure of an EFCV to isolate cannot initiate previously evaluated accidents. Therefore, there is no increase in the probability of occurrence of an accident as a result of this proposed change.

The EFCVs provide primary containment isolation provisions consistent with the guidelines of Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11)." The NRC-approved General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," discusses through operating experience that there is a high degree of reliability with the EFCVs and that there are little radiological consequences resulting from an EFCV failure.

The radiological consequences for an instrument line break do not credit the EFCVs for isolating the break. Therefore, the consequences of an instrument line break are not impacted by the proposed reduced testing. Based on the above, the proposed TS change does not involve a significant increase in the consequences of an 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 allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience as documented in NEDO-32977-A, provides supporting evidence that the reduced testing will not affect the high reliability of these valves. The potential failure of an EFCV to isolate as a result of the proposed reduced testing is bounded by the evaluation of an instrument line pipe break described in USAR Section 15.6.2 is bounded by a break in the main steam line. The proposed changes do not physically alter the plant and will not alter the operation of the structures, systems and components as described in the USAR. Therefore, a new or different kind of accident from any accident previously evaluated will not be created.

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

Industry experience with EFCVs indicates that they have very low failure rates. Postulated failures of an EFCV to isolate as a result of reduced testing is bounded by the limiting analysis in the USAR, that being the main steam line break analysis. The proposed change does not alter the instrument line design in any manner, and the integrity and functional performance of the secondary containment and SGTS are not affected by this proposed change. 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 change does 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, which allows testing of a representative sample of Excess Flow Check Valves (EFCVs) each refueling outage 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. These changes do not result in an increase in power level, do 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. The proposed change only affects the frequency of testing EFCVs each refueling outage. 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 this change.

ATTACHMENT E
RELIEF REQUEST FOR EXCESS FLOW CHECK VALVE TESTING REQUIREMENTS AT CLINTON POWER STATION

RELIEF REQUEST 2203

COMPONENT IDENTIFICATION

Code Class:	2
References:	OMa-1988, Part 10
Category:	A/C
Description:	Excess Flow Check Valve
Component Numbers:	1CM002A, 1CM002B, 1CM003A, 1CM003B, 1CM051, 1CM053, 1CM066, 1CM067, 1SM008, 1SM009, 1SM010, 1SM011, 1VR016A, 1VR016B, 1VR018A, 1VR018B, 1VG056B, 1VG057B, 1E22F330, 1E22F332, 1E51F377A, 1E51F377B

CODE REQUIREMENTS

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Operations and Maintenance (OM)-10, "Inservice Testing of Valves in Light Water Reactor Power Plants," Section 4.2, "Inservice Tests for Category A and B Valves," and Section 4.3, "Inservice Tests for Category C Valves," requires exercising valves nominally every three months to the positions in which they perform their safety functions. Paragraphs 4.2.1.2(e) and 4.3.2.2(e) allow deferral of this requirement to at least every reactor refueling outage. Paragraph 4.1, "Valve Position Verification," requires valve position verification at least once every two years to verify that valve operation is accurately indicated. At Clinton Power Station (CPS), excess flow check valves (EFCVs) are ASME Category A/C and are also containment isolation valves. These valves are exercise-tested and remote position indication tested every refueling outage.

CODE REQUIREMENT FROM WHICH RELIEF IS REQUESTED

Testing frequency for valve exercising and position indication verification.

BASIS FOR RELIEF

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3), we are requesting that the requirement for valve exercising and position indication verification be revised to be consistent with the proposed change to CPS Technical Specifications (TS) Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.12 to allow a representative sample of EFCVs to be tested every refueling outage (i.e., 18 months). This change is consistent with General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000. This report was approved in an NRC Safety Evaluation issued by NRC letter dated March 14, 2000. This report demonstrates the high degree of EFCV reliability based on extremely low industry failure rates.

PROPOSED ALTERNATIVE PROVISIONS

Exercising and position indication verification will be performed on a frequency to coincide with revised CPS TS SR 3.6.1.3.12.