# AmerGen

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10 CFR 50.90

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

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: License Amendment Request
Increase Safety Valve As-Found Setpoint Tolerance from ±1% to ±3%

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen), hereby requests the following amendment to the Technical Specifications (TS), Appendix A of Operating License No. DPR-16 for Oyster Creek Generating Station (OCGS).

The proposed changes will revise the OCGS TS Section 2.3.F, "Reactor High Pressure Safety Valve Initiation" and TS Section 4.3, "Reactor Coolant," and the respective TS Bases Sections, to increase the allowable as-found main steam Safety Valve code safety function lift setpoint tolerance from ±1% to ±3%. This change keeps the as-left Safety Valve code safety function lift setting within ±1% of the specified nominal lift setpoint prior to reinstallation in the plant. This change does not alter the TS requirements on the number of Safety Valves in each lift pressure grouping, the Safety Valve nominal lift setpoints, or the Safety Valve lift setpoint test frequency.

The proposed TS changes are consistent with guidance specified in Boiling Water Reactor Owners' Group (BWROG) document NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," which was developed to support the use of a ±3% safety lift setpoint tolerance for Safety Valves. NEDC-31753P has been reviewed and approved by NRC as documented in its Safety Evaluation Report (SER) issued by letter dated March 8, 1993.

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An additional administrative change is proposed for TS Section 4.3.E, to delete a reference to a subsection of the ASME Boiler & Pressure Vessel Code which no longer exists. Information supporting this License Amendment Request is contained in Attachment 1 to this letter, and the proposed marked-up TS pages are contained in Attachment 2. Attachment 3 contains the proposed marked-up TS Bases pages. The Attachment 4 analysis contains information proprietary to General Electric. General Electric requests that the document be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). An affidavit supporting this request is also contained in Attachment 4. Attachment 5 contains a non-proprietary version of the General Electric Analysis.

AmerGen has concluded that the proposed changes present No Significant Hazards consideration under the standards set forth in 10 CFR 50.92. No new regulatory commitments are established by this submittal.

AmerGen requests approval of the proposed amendment by October 1, 2006. Approval by this date is requested to support planned Safety Valve testing during Oyster Creek's Fall, 2006 refueling outage. Once approved, the amendment shall be implemented within 60 days of issuance.

These proposed changes have been reviewed by the OCGS Plant Operations Review Committee, and approved by the Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91(b)(1), "Notice for public comment; State consultation," paragraph (b), AmerGen Energy Company, LLC is notifying the State of New Jersey of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Doug Walker at (610) 765-5726.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

12/2/05
Executed on

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

Attachments:
1) Evaluation of Proposed Changes
2) Proposed Technical Specifications Marked-Up Pages
3) Proposed Technical Specifications Bases Marked-Up Pages
4) GE-NE-0000-0046-3343-R0, Oyster Creek SSV Set-point Tolerance Change Effects on Anticipated Operational Occurrences, and General Electric Company Affidavit, Proprietary
5) GE-NE-0000-0046-3343-R0, Oyster Creek SSV Set-point Tolerance Change Effects on Anticipated Operational Occurrences, Non-Proprietary

cc: S. J. Collins, USNRC, Administrator, Region I
G. E. Miller, USNRC, Project Manager
M. S. Ferdas, USNRC Senior Resident Inspector, OCGS
K. Tosch, Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection
File No. 05053

ATTACHMENT 1

**Oyster Creek Generating Station**

**Docket No. 50-219**

**License No. DPR-16**

**License Amendment Request**

**"Increase Safety Valve As-Found Setpoint Tolerance from ±1% to ±3%"**

**Evaluation of Proposed Changes**

## 1.0 DESCRIPTION

AmerGen Energy Company, LLC, licensee under Facility Operating License No. DPR-16 for Oyster Creek Generating Station (OCGS), requests that the Technical Specifications (TS) contained in Appendix A to the Operating License be amended as proposed herein to permit changes in the Safety Relief Valve (SRV) setpoint tolerance. It should be noted that Oyster Creek utilizes spring safety valves (Main Steam Safety Valves) to perform the safety function of the SRVs as discussed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report." The use of the term Main Steam Safety Valves (MSSV) is used throughout this report and is considered equivalent to the SRV's referenced in NEDC-31753P for the purposes of this submittal.

This proposed License Amendment Request involves revising TS Section 2.3.F, "Reactor High Pressure Safety Valve Initiation," and TS Section 4.3, "Reactor Coolant", and the respective associated Bases sections, to increase the allowable as-found MSSV code safety function lift setpoint tolerance from ±1% to ±3%. The as-left MSSV code safety function lift setting will still be required to be set within ±1% of the specified nominal lift setpoint prior to reinstallation in the plant. An additional administrative change is proposed for TS Section 4.3.E, to delete a reference to a subsection of the ASME Boiler & Pressure Vessel Code which no longer exists.

These proposed changes do not alter the MSSV nominal lift setpoints or the MSSV lift setpoint test frequency. The proposed changes do not change the MSSV testing commitment specified in the OCGS, TS Section 4.3 "Reactor Coolant."

This TS Change Request provides a discussion and description of the proposed TS changes, a safety assessment of the proposed TS changes, information supporting a finding of No Significant Hazards Consideration, and information supporting an Environmental Assessment.

## 2.0 PROPOSED CHANGE

The proposed LAR revises Oyster Creek TS Section 2.3.F, "Reactor High Pressure Safety Valve Initiation," and TS Section 4.3, "Reactor Coolant" and the associated Bases. The proposed revision implements a higher MSSV as-found setpoint tolerance to better match the TS performance requirements with the installed valve capabilities. This will align OCGS with Section XI of the ASME Boiler and Pressure Vessel Code. Additionally, this will reduce the number of non-safety significant Licensee Event Reports written due to MSSV as-found setpoints being outside technical specification limits.

The change increases the allowable MSSV safety function spring setpoint tolerance from ±1% to ±3%. The proposed change does not alter the TS requirements on the nominal MSSV safety function lift setpoints, the required frequency for the MSSV lift setpoint testing, or the number of MSSVs currently required to be operable. This proposed revision does not change the requirement that the MSSVs be adjusted to within ±1% of their nominal lift setpoints following testing as required in the MSSV In-service Inspection Procedures. An additional administrative change is proposed for TS Section 4.3.E, to delete a reference to a subsection of the ASME Boiler & Pressure Vessel Code which no longer exists.

<u>Proposed Change 1:</u> In TS section 2.3.F, revise the Reactor High Pressure, Safety Valve Initiation tolerance from ± 12 psi to ± 36 psi.

<u>Proposed Change 2:</u> In TS section 4.3, Reactor Coolant, Section E, revise the Safety Valve Initiation tolerance from ± 12 psi to ± 36 psi. In addition, the reference to "subsection IWV-3510 of section XI of the ASME Boiler and Pressure Vessel Code" is deleted and a reference is made to testing requirements of technical specification 4.3.C.

<u>Proposed Change 3:</u> Revise applicable TS Bases for section 2.3 to read as follows:

*"The ASME B&PV Code allows an as-found ±3% of setpoint pressure variation in the lift point of the valves. The as-left MSSV setpoint tolerance requirement will remain ±1% per GE NEDC-31753P recommendation."*

<u>Proposed Change 4:</u> Revise applicable TS Bases for section 4.3 to read as follows:

*"Experience in safety valve operation shows testing in accordance with Section XI of the ASME Boiler and Pressure Vessel Code is adequate to detect failures or deterioration. The as-found setpoint tolerance value is specified in Section XI of the ASME Code at ±3% of design pressure. An analysis has been performed which shows that with all safety valves set 36 psig higher, the safety limit of 1375 psig is not exceeded."*

## 3.0 <u>BACKGROUND</u>

There are 14 safety and relief valves on the 24-inch main steam headers inside the drywell. Nine are spring-loaded safety valves and discharge directly into the drywell. The remaining five are the Electromatic Relief Valves, controlled automatically by reactor pressure switches, Automatic Depressurization System (ADS) signals, or manually from the Control Room, and discharge to the suppression pool. These Electromatic Relief Valves are the main components of the ADS. The spring loaded safety valves typically do not open on any pressure transient except those resulting from main steamline isolation with scram failure, failure of the Electromatic Relief Valves, or failure of the turbine bypass valves.

The nine MSSVs provide reactor vessel overpressure protection for plant operations at the licensed core thermal power level of 1930 MWt. The MSSVs are designed to limit the reactor vessel pressure to 110% of the design pressure during a Main Steam Isolation Valve (MSIV) closure with reactor scram on high neutron flux.

The MSSVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. There are a total of 9 MSSVs. All 9 MSSVs are required to be operable.

The use of ±1% allowable as-found MSSV safety function lift setpoint tolerance in plant TSs was a generic issue in the industry. Nuclear power plant licensees have experienced difficulty in meeting the typical 1% setpoint tolerance for MSSVs. As a result, the BWR Owners' Group (BWROG) developed NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," to support the use of ±3% setpoint tolerance, which is consistent with American Society of Mechanical Engineers (ASME) OM Code requirements (formally Section XI requirements). On March 8, 1993, the NRC Staff issued their Safety Evaluation (SER) of Licensing Topical Report NEDC-31753P.

In the SER, the NRC stated that a generic change of setpoint tolerance to ±3% is acceptable provided that it is evaluated in the analytical bases. Specific analyses required to be provided are transient analysis, design basis overpressurization event, re-evaluation of high pressure systems (Motor Operated Valves, Reactor Vessel instrumentation and piping), alternate operating modes, containment response during LOCA, and hydrodynamic loads on MSSV discharge lines.

The results of these plant specific analyses are discussed in Section 4.0.

## 4.0 TECHNICAL ANALYSIS

The ±1% allowable as-found MSSV code safety function lift setpoint tolerance currently specified in TS Section 2.3.F and TS Section 4.3 for OCGS is based on the acceptance criteria originally defined by the ASME, Section I. ASME Section I was used for the design of OCGS's reactor pressure vessel (note that for newer plants, ASME Section III was used).

The use of the ±1% allowable as-found MSSV code safety lift setpoint tolerance in plant TSs is generic in the industry. Nuclear power plant licensees have experienced difficulty in meeting the typical 1% setpoint tolerance for MSSVs. As a result, the Boiling Water Reactor Owners' Group (BWROG) developed NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," to support the use of the ±3% MSSV code safety lift setpoint tolerance consistent with that specified in ASME Section XI requirements.

Additionally, an administrative change is being proposed to correct the reference to "IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code" in TS 4.3.E because the stated section no longer exists. The reference to IWV-3510 was deleted with no replacement provided in the ASME Boiler and Pressure Vessel Code. The TS is being changed to state MSSV testing will be in accordance with specification 4.3.C.

NEDC-31753P was reviewed and approved by the NRC as documented in a Safety Evaluation Report (SER) issued by letter dated March 8, 1993. The NRC determined that it is acceptable for licensees to submit TS amendment requests to revise the SRV (MSSV for OCGS) code safety function lift setpoint tolerance to ±3%, provided that the setpoints for those SRVs tested are restored to ±1% prior to reinstallation. The NRC also indicated in its SER that licensees planning to implement TS changes to increase the SRV setpoint tolerances should provide the following plant specific analyses:

1. Transient analysis, using NRC approved methods, of abnormal (anticipated) operational occurrences (AOOs) as described in NEDC-31753P utilizing a ±3% setpoint tolerance for the safety mode of the SRVs. (Note: OCGS UFSAR designates these events as abnormal operational transients (AOTs)).
2. Analysis of the design basis overpressure event using the ±3% tolerance limit for the SRV setpoints to confirm that the vessel pressure does not exceed ASME pressure vessel code upset limits.
3. Plant specific analyses described in Items 1 and 2 should assure that the number of SRVs included in the analyses corresponds to the number of valves required to be operable in the Technical Specifications.
4. Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping considering the ±3% tolerance limit.
5. Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.).

6. Evaluation of the effects of the ±3% tolerance limit on the containment response during loss-of-coolant accidents (LOCA) and the hydrodynamic loads on the SRV discharge lines and containment.

**Analysis of Anticipated Operational Transients (AOTs)**

An evaluation of the effects of pressurization transients on the fuel thermal limits was performed to determine whether the increase in MSSV code safety function lift setpoint tolerance from ±1% to ±3% would be acceptable. The Minimum Critical Power Ratio (MCPR) and fuel thermal-mechanical limits were considered in this evaluation. A review of the transient analysis identified the Turbine Trip No Bypass (TTNBP) as the limiting event. This event was evaluated to determine the impact of the MSSV code safety function lift setpoint tolerance change on thermal limits. The results are based on the reload analysis for OCGS Cycle 20. However, because the general trends and characteristics (e.g., time of MCPR and peak nodal power vs. time of MSSV opening) of the TTNBP and other AOTs do not change from cycle to cycle, the conclusions are generic and apply to future operating cycles.

The AOT analyses for OCGS conservatively do not credit the opening of the MSSVs. Furthermore, the results of the analyses demonstrate that the MCPR and peak nodal power both occur prior to the time that any MSSV would be expected to open. Therefore, increasing the MSSV setpoint to +3% above nominal will not affect the calculated thermal limit results.

For a decrease in MSSV opening pressure of 3% below the nominal setpoint, the results of the analysis continue to demonstrate that the MCPR and peak nodal power both occur prior to the time that any MSSV would be expected to open. Therefore, decreasing the setpoint to 3% below the nominal will not affect the calculated thermal limits results if the MSSVs are assumed to open.

Based on this evaluation, it is concluded that the increase in MSSV code safety function lift setpoint tolerance from ±1% to ±3% is acceptable for AOT events.

**Reactor Vessel Overpressure Protection**

The ASME Boiler & Pressure Vessel Code requires that the peak vessel pressure remains less than 110% of the vessel design pressure. The design pressure of the OCGS reactor vessel is 1250 psig. 110% of the design pressure is 1375 psig. TS Safety Limit 2.2, "Reactor Coolant System Pressure," requires that the reactor coolant system pressure not exceed 1375 psig. The limiting overpressure event is the Main Steam Isolation Valve (MSIV) Closure with Flux Scram event. This event assumes the failure of the MSIV limit switch (more than 10% closed) Scram function. The reactor Scram is initiated by the high neutron flux caused by the reactor vessel pressurization and the resultant collapse of moderator voids within the reactor core.

As part of the standard reload licensing analysis for Cycle 20, General Electric (GE) has performed an analysis of this event assuming 102% core thermal power, maximum licensed core flow (limiting condition), and conservative End-of-Cycle 20 nuclear dynamic parameters. This analysis also assumes that all the Electromatic Relief Valves fail. The setpoint of each of the MSSVs was assumed to be 3% above its nominal setpoint. The NRC-approved GE thermal hydraulic and nuclear kinetics coupled transient code (ODYN) was used to obtain the system response and peak calculated vessel pressure. The peak calculated reactor vessel (bottom head) pressure is 1335 psig, which is below the ASME limit of 1375 psig. The reactor vessel overpressure protection event is reanalyzed each reload to verify that the ASME overpressure protection criterion continues to be met.

**Number of MSSVs**

OCGS has a total of nine (9) MSSVs. All MSSVs are required to be operational per Technical Specification 4.3.E. The results of the previously discussed analyses are based on nine (9) operational MSSVs and no change to this number is required per this LAR.

**High Pressure System Performance**

The OCGS design does not include a high pressure inventory makeup system. OCGS relies on its low pressure Core Spray system for inventory makeup, which is actuated at reactor pressures below approximately 285 psig. Therefore, changing the upper MSSV setpoint tolerance from +1% to +3% above the nominal setpoint does not impact performance of the low pressure inventory makeup system.

Changing the upper MSSV setpoint tolerance will increase the maximum reactor pressure in which high pressure systems must operate, which could potentially cause a reduction in the performance capability of these systems. This section documents the evaluation of the impact of the proposed MSSV opening setpoint tolerance changes on the performance of the following high-pressure systems:

- Standby Liquid Control System (SLCS)
- Emergency Isolation Condensers

This evaluation is based on the current design basis requirements for these systems. Increasing MSSV opening pressure is limiting for system performance evaluations.

SLC System

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in an event in which all or some of the control rods cannot be inserted during a postulated Anticipated Transient Without Scram (ATWS) event. The SLCS performs this function by injecting a liquid solution of sodium pentaborate into the reactor vessel. The rate of reactivity compensation provided by the SLCS is designed to exceed the rate of reactivity gain associated with the reactor cool-down from the full power condition. The system is not provided as a backup for reactor trip functions since most transient conditions that require reactor trip occur too rapidly to be controlled by the SLCS.

The SLCS consists of two redundant, parallel, reciprocating piston, positive displacement pumps. The system's maximum pressure is limited by the setpoint for the system relief valves located on the discharge line of each pump. Oyster Creek's SLCS pump relief valves have a setpoint of 1400 psig. The limiting overpressure event (MSIV closure with high flux Scram) results in a maximum reactor vessel pressure of 1335 psig (as compared to 1320 psig for the previous cycle analysis with a 1% setpoint tolerance). This maximum overpressure value provides a 65 psi operating margin for the SLCS compared with the previous cycle (Cycle 19) operating margin of 80 psi. This 15 psi reduction in SLCS operating margin does not significantly impact the operability of the SLC system under overpressure conditions.

Furthermore, the analysis basis for ATWS credits the actuation of the EMRVs, which would further reduce maximum reactor vessel pressure. In addition, the ATWS event is a low probability event for which the use of nominal system operating parameters for event analysis has been accepted by the NRC. Since the nominal MSSV setpoints are not being changed, this proposed change does not affect the capability of the SLC system to mitigate the consequences of an ATWS event.

Therefore, the performance of the SLCS is not impacted by increasing the MSSV setpoint tolerance to ±3%.

Emergency Isolation Condensers

The emergency isolation condenser system is a standby high pressure system for removal of fission product decay heat if the main condenser is not available as a heat sink following a reactor isolation scram. The system is not intended to be activated quickly enough to have any effect on the initial pressure peaks resulting from transients such as main steam isolation valve closure. The system is activated manually or automatically upon sustained high pressure.

The emergency isolation condenser system operates by natural circulation, with steam flowing from the reactor vessel through heat exchanger tubes and returning back to the reactor vessel as condensate. The heat exchanger steam inlet valves from the reactor are normally open. The condensate return valves are normally closed and are opened to place the system in operation and allow condensate discharge back to the associated reactor recirculation pump suction line.

The emergency isolation condenser system is automatically actuated by a reactor vessel pressure at or above 1060 psig for approximately 1.5 seconds. The relaxation of the MSSV setpoint tolerance may increase the peak reactor pressure at which the emergency isolation condensers may be required to function. However, the emergency isolation condensers are a passive system and the function of the system is not affected by the increased system pressure. In addition, the peak reactor pressure continues to be within the system design.

Opening the motor-operated condensate return valve actuates each emergency isolation condenser. This MOV opens against the pressure differential (dP) between the vessel steam pressure and the vessel condensate pressure. This dP is not impacted by increasing the MSSV setpoint tolerance to ±3%. Isolation of the emergency isolation condensers in the event of a line break inside the condenser is achieved through closure of the steam supply valves and the condensate return valves. The GL 89-10, "Safety Related Motor Operated Valve Testing and Surveillance," requirements for closure of these valves are based on a line break at nominal reactor operating pressure. Therefore the performance of the emergency isolation condensers is not affected by this proposed change.

**Alternate Operating Modes Evaluation**

OCGS has evaluated the impact of the ±3 percent tolerance on plant-specific alternate operating modes. The alternate operating modes, including Extended Load Line Limit (ELLL), Increased Core Flow, and Feedwater Temperature Reduction were considered in determining the most restrictive analytical conditions (i.e., most limiting operating mode) for performing the analysis associated with the proposed TS change. Therefore, the impact of the ±3 percent tolerance on the plant-specific alternate operating modes has been explicitly addressed and determined to be acceptable.

**Containment Response During LOCA and the Hydrodynamic Loads on MSSV Discharge Lines and Containment**

The increase in the MSSV setpoint tolerance to ±3% was assessed to determine the potential impact on the containment design limits. The two primary areas of concern for the containment structures are (1) the pressure and temperature response as well as (2) the containment hydrodynamic loads from the MSSV discharge lines.

MSSV actuation exerts pressure and drag loads on containment structures and these discharge loads are potentially affected by an increase in discharge flow associated with an increased MSSV setpoint tolerance.

Containment Pressure and Temperature Response

The most limiting event in terms of peak containment pressure is the design basis accident (DBA) LOCA, a recirculation line discharge break (UFSAR Section 6.2.1.1.3). An increase in MSSV setpoint tolerance has no effect on this event because the vessel depressurizes without MSSV actuation. Therefore there is no effect on the DBA-LOCA containment peak pressure due to MSSV setpoint tolerance relaxation.

The most limiting event in terms of peak containment temperature is the main steam line break accident. Large steam line breaks result in a rapid depressurization that does not induce MSSV actuation. Smaller steam line breaks can result in high containment temperatures that can last for relatively long time periods because the vessel remains at high pressure for a longer period of time than the DBA LOCA. Closure of the MSIVs during a small steam line break event may pressurize the vessel enough to actuate the MSSVs.

For small steam line breaks with MSSV actuation the peak containment temperature occurs relatively late in the event following many MSSV actuations. The increased MSSV setpoint tolerance will result in a slight delay in initial MSSV actuation and a higher discharge flow rate due to the higher vessel pressure. The total inventory loss from the vessel into the containment during MSSV actuations at higher pressures will be similar in magnitude because the MSSVs will be cycling at a different rate due to the higher opening pressure and increased flow rate at that pressure. The containment temperature is primarily dependent on reactor decay heat, which is not affected by an increase in the MSSV setpoint. Therefore there is no effect on steam line break peak containment temperature due to MSSV setpoint tolerance relaxation.

Hydrodynamic Loads on MSSV Discharge Lines and Containment

The MSSVs discharge into the branch of a full size tee (the inside diameter of the tee matches the size of the safety valve discharge). There is no discharge pipe and the MSSVs do not discharge directly into the torus. The EMRVs, which are part of the Automatic Depressurization System, have discharge piping to the torus. Torus hydrodynamic loads are not changed by the MSSV discharge.

The MSSV discharge tees are oriented such that steam jet impingement impact on surrounding structures and components is avoided. The small increase in MSSV discharge flow rate resulting from an increase in setpoint tolerance will have negligible impact on containment structures and components.

Oyster Creek's MSSVs are designed to operate at 94% of their nominal setpoint pressure without simmer. With a 1% setpoint tolerance, the MSSVs are operating at no lower than 99% of their nominal setpoint pressure. With a 3% setpoint tolerance, the MSSVs will be operating at no lower than 97% of their nominal setpoint pressure. The 94% designed value continues to be bounding for the purposes of simmer margin.

Conclusions of NRC Specified Evaluations

As previously discussed in the items above, increasing the allowable MSSV as-found setpoint tolerance from ±1% to ±3% is considered acceptable based on the plant specific analyses, as required by the NRC in its SER accepting NEDC-31753P, dated March 8, 1993. In addition to the requirements specified in the NRC's SER, the following items were also considered:

Control Rod Drive System

The Control Rod Drive (CRD) system performs two functions. The first function is the movement of the control rod blades to control reactor power during normal plant operations (e.g., startup, power, and shutdown). The second function is the rapid insertion of all control rods (Scram) due to abnormal plant conditions. Increasing the allowable as-found MSSV code safety function lift setpoint tolerance from ±1% to ±3% does not change the normal reactor vessel operating pressure or the high reactor vessel pressure SCRAM setpoint; therefore, the proposed increase in MSSV as-found setpoint tolerance will not impact the CRD system capability of controlling reactor power during normal plant operations.

Increasing the allowable as-found MSSV setpoint tolerance will not change the reactor vessel pressure at which a Scram is initiated but does have the potential for increasing the reactor pressure the CRD system is subjected to during the Scram process. The CRD system is designed such that any increased reactor vessel pressure above atmospheric pressure aids in the insertion of the control rod blades during the Scram process. The nitrogen accumulators are capable of executing a full scram at lower reactor pressures and elevated reactor pressure may act as a supplementary force in driving the control rods. The potential for a higher MSSV setpoint drift may increase the rapidity of the Scram function by increasing the maximum reactor pressure, while a potentially lower maximum pressure will not impair the ability of the CRD system to perform a Scram. Therefore, the proposed change will not negatively impact CRD Scram performance.

Reactor Vessel Instrumentation System

The components of the Reactor Vessel Instrumentation System are designed to the same or greater pressure/temperature requirements as the reactor pressure vessel that have been evaluated as being acceptable for this proposed change. Normal plant operating parameters, which are unchanged by this proposed change, are used as inputs for instrument calibration.

The lowest potential MSSV setpoint of 1176 psig (1212 psig - 3%) was evaluated and maintains significant margin over the highest EMRV setpoint of 1105 psig, hence sufficient margin exists between the actuation of the EMRVs and the lifting of the MSSVs.

Therefore, increasing the allowable MSSV as-found setpoint tolerance from ±1% to ±3% will not affect the Reactor Vessel Instrumentation System.

Emergency Procedure Guidelines

The Emergency Procedure Guidelines (i.e., TRIP/SAMP Procedures) use nominal, realistic, and best estimate plant parameters for determining action levels. The nominal setpoints of the MSSVs are not impacted by the proposed TS changes. Therefore, increasing the allowable MSSV as-found setpoint tolerance from ±1% to ±3% will not affect the Emergency Procedure Guidelines.

## 5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

AmerGen Energy Company, LLC, 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. **Will operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed TS changes allow for an increase in the as-found Main Steam Safety Valve (MSSV) setpoint tolerance from ±1% to ±3%. The proposed changes do not alter the MSSV nominal lift setpoints or MSSV lift setpoint test frequency.

The proposed TS changes have been evaluated on both a generic and plant specific basis. The NRC has approved the general approach of this change; however, implementation is contingent on several plant specific evaluations. The required plant specific analyses and evaluations included transient analysis of the anticipated operational transients (AOTs); analysis of the design basis overpressurization event; evaluation of the performance of high pressure systems, and evaluation of the containment response during Loss-of-Coolant Accident (LOCA) and hydrodynamic loads on the MSSV discharge lines and containment. These analyses and evaluations demonstrate that there is adequate margin to the design core thermal limits and reactor vessel pressure limits using the ±3% MSSV as-found setpoint tolerance. The analyses and evaluations also demonstrate that the operation of high-pressure safety systems will not be adversely affected and that the containment response during a LOCA will be acceptable.

Evaluations of the impact of the proposed change on the equipment important to safety have been performed and no adverse conditions were identified. The reactor pressure vessel and attached systems and piping have been evaluated for the impact of this proposed TS change. A plant specific analysis has been performed which indicates that the ASME Code upset limits for the reactor pressure vessel will not be exceeded for the limiting event, i.e., Main Steam Isolation Valve (MSIV) closure with flux Scram. The reactor pressure vessel and attached piping design values will not be exceeded. Therefore, the probability of a malfunction of the reactor pressure vessel and attached systems and piping is not increased and the consequences of such an accident remain acceptable.

The nuclear fuel has been evaluated for the impact of the proposed change. Plant specific analyses were performed which indicate that for all abnormal operational transients adequate margin to the fuel thermal limit parameters, i.e., Minimum Critical Power Ratio (MCPR) and thermal-mechanical limits, is maintained. Emergency Core Cooling System (ECCS)/LOCA performance is maintained adequate to meet the requirements of 10CFR50.46. Therefore, the consequences of these accidents remain acceptable and the probability of the malfunction of the nuclear fuel is not increased.

The Containment response during a LOCA has been evaluated for the impact of the proposed change. The major factor in the Containment pressure response to a LOCA is the rate of reactor vessel water inventory loss due to a DBA LOCA. The rate of reactor vessel water inventory loss is mainly dependent on the initial reactor pressure, which is not affected by the proposed setpoint tolerance change. The major factor in the Containment temperature response to a LOCA is the integrated steam inventory loss due to Main Steamline Break. The rate of reactor vessel steam inventory loss is mainly dependent on the reactor decay heat, which is not affected by the proposed setpoint tolerance change. Therefore, the consequences of these accidents remain acceptable and the probability of the malfunction of the Containment is not increased.

The Control Rod Drive (CRD) system has been evaluated for the impact of the proposed change. The CRD system capability of controlling reactor power during normal plant operation and rapidly inserting control rod blades (Scram) during abnormal plant conditions is not impacted by the proposed change. Therefore, the probability of a malfunction of the CRD system is not increased.

The Reactor Vessel Instrumentation System has been evaluated for the impact of the proposed change. The Reactor Vessel Instrumentation System will continue to be operated within the current design pressure/temperature requirements; therefore, the probability of a malfunction of the Reactor Vessel Instrumentation System is not increased.

An administrative change is also being proposed to correct the reference to "IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code" in TS 4.3.E because the stated ASME section no longer exists. The TS is being changed to reference specification 4.3.C for MSSV testing. This is an administrative change and does not affect previously evaluated accidents.

Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

2. **Will operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed TS changes allow for an increase in the as-found MSSV setpoint tolerance from ±1% to ±3%. Generic and plant specific analyses and evaluations indicate that the plant response to any previously evaluated event will remain acceptable. All plant systems, structures, and components will continue to be capable of performing their required safety function as required by event analysis guidance.

The proposed TS changes do not alter the MSSV nominal lift setpoints or MSSV lift setpoint test frequency. The operation and response of the affected equipment important to safety is unchanged. All systems, structures, and components will continue to be operated within acceptable operating and/or design parameters. No system, structure, or component will be subjected to a condition that has not been evaluated and determined to be acceptable using the guidance required for specific event analysis.

The change to correct the reference to "IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code" in TS 4.3.E is an administrative change and does not affect the possibility of a new or different kind of accident.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Will operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

The proposed TS changes allow for an increase in the as-found MSSV setpoint tolerance from ±1% to ±3%. The proposed TS changes do not alter the MSSV nominal lift setpoints or MSSV lift setpoint test frequency. The operation and response of the affected equipment important to safety is unchanged. All systems, structures, and components will continue to be operated within acceptable operating and/or design parameters. While the calculated peak reactor vessel pressure for the ASME overpressure event is higher than that calculated without the increase in setpoint tolerance, it is still within the respective licensing acceptance limits associated with this event. These licensing acceptance limits have been determined by the NRC to provide a sufficient margin of safety.

The increase in MSSV steam flow and reactor vessel pressure does not reduce the margin of safety associated with the MSSVs and associated components and structures since the increased MSSV steam flow rate and reactor vessel pressure are bounded by the current design analysis.

The margin of safety for fuel thermal limits and 10CFR50.46 limits are unaffected by the proposed change.

The margin of safety for the Containment is unaffected by the proposed change.

The capability of the SLC system and the CRD system to perform their safety functions during all required events, using the required guidance for event analysis, is maintained. Therefore, the proposed changes do not reduce the margin of safety provided by the SLC and CRD systems.

The change to correct the reference to "IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code" in TS 4.3.E is an administrative change and does not affect the margin of safety.

Therefore, these proposed TS changes do not involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements/Criteria

The ±1% allowable as-found MSSV code safety function lift setpoint tolerance currently specified in OCGS TS is based on the acceptance criteria originally defined by the American Society of Mechanical Engineers (ASME), Section I; the code to which OCGS was built. The existing MSSVs are tested in accordance with ASME OM Code. "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," for the testing of MSSVs. ASME OM-1-1981 specifies a 3% acceptance criteria whereas the current OCGS TS specifies a more restrictive 1% tolerance.

The use of the ±1% allowable as-found MSSV code safety lift setpoint tolerance in plant TS is generic in the industry. Nuclear power plant licensees have experienced difficulty in meeting the typical 1% setpoint tolerance for MSSVs. As a result, the Boiling Water Reactor Owners' Group (BWROG) developed NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," to support the use of the ±3% MSSV code safety lift setpoint tolerance consistent with that specified in ASME Section XI requirements.

NEDC-31753P was reviewed and approved by the NRC as documented in a Safety Evaluation Report (SER) issued by letter dated March 8, 1993. The NRC determined that it is acceptable for licensees to submit TS amendment requests to revise the MSSV code safety function lift setpoint tolerance to ±3%, provided that the setpoints for those MSSVs tested are restored to ±1% prior to reinstallation. The NRC also indicated in its SER that licensees planning to implement TS changes to increase the MSSV setpoint tolerances should provide a plant specific analysis, which has been described in Section 4.0 above.

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.

## 6.0 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 10CFR20, 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 the 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.

## 7.0 PRECEDENT

The proposed change to the MSSV Setpoint tolerance has been implemented at several BWRs including Fermi, Vermont Yankee, Perry, LaSalle, Limerick, Susquehanna, River Bend, Grand Gulf, and Hope Creek.

## 8.0 REFERENCES

8.1 Letter to BWROG, "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated March 8, 1993

8.2 GE-NE-0000-0046-3343-R0, "Oyster Creek SSV Set-point Tolerance Change Effects on Anticipated Operational Occurrences," October 2005

8.3 0000-0029-3151-SRLR, Rev. 1, "Supplemental Reload Licensing Report for Oyster Creek Reload 20, Cycle 20," September 2004

8.4 ES-027, Rev. 4, "Environmental Parameters - Oyster Creek NGS."

8.5 TDR No. 180, Rev. 1, "Oyster Creek Containment Temperature Profile for Environmental Qualification of Equipment," June 1988.

8.6 M-NC112, Rev. 1, "Instructions for Installation and Maintenance of Consolidated Main Steam Safety Valves."

8.7 0000-0005-5032-SRLR, Rev. 0, "Supplemental Reload Licensing Report for Oyster Creek Reload 18, Cycle 19," September 2002.

**ATTACHMENT 2**

**Oyster Creek Generating Station**

**Docket No. 50-219**

**License No. DPR-16**

**License Amendment Request**

**"Increase Safety Valve As-Found Setpoint Tolerance from ±1% to ±3%"**

**Proposed Technical Specifications Marked-Up Pages**

**The pages included in this attachment are:**

**PAGES**

2.3-2
4.3-1

| FUNCTION | LIMITING SAFETY SYSTEM SETTINGS |
|---|---|

B. Neutron Flux, Control Rod Block

The Rod Block setting shall be the minimum of:

For $W \geq 0.0 \times 10^6 \text{lb/hr}$:

$$S \leq \left[(0.90 \times 10^{-6})\, W + 60.1\right] \frac{FRP}{MFLPD} \quad ; \text{or}$$

The applicable stability protection settings, as defined in the COLR,

with a maximum setpoint of 115.0% for core flow equal to $61 \times 10^6$ lb/hr and greater.

The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used.

This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.

| | | |
|---|---|---|
| C. | Reactor High Pressure, Scram | ≤1060 psig |
| D. | Reactor High Pressure, Relief Valves Initiation | 2 @ ≤ 1085 psig<br>3 @ ≤ 1105 psig |
| E. | Reactor High Pressure, Isolation Condenser Initiation | ≤ 1060 psig with time delay<br>≤ 3 seconds |
| F. | Reactor High Pressure, Safety Valve Initiation | 36<br>4 @ 1212 psig ±12 psi<br>5 @ 1221 psig ±12 psi |
| G. | Low Pressure Main Steam MSIV Closure | ≥ 825 psig (initiated in IRM Line, range 10) |
| H. | Main Steam Line Isolation Valve Closure, Scram | ≤10% Valve Closure from full open |

## 4.3 REACTOR COOLANT

**Applicability:** Applies to the surveillance requirements for the reactor coolant system.

**Objective:** To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

**Specification:**

A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. The results of these evaluations shall be used to assess the adequacy of the P-T curves A, B, and C in Figure 3.3.1, 3.3.2 and 3.3.3. New curves shall be generated as required.

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

C. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(f), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(f)(6)(i).

D. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with ~~subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code~~ Specification C above. Setpoints shall be as follows:

| Number of Valves | Set Points (psig) |
|---|---|
| 4 | 1212 ± ~~12~~ 36 |
| 5 | 1221 ± ~~12~~ 36 |

F. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

**ATTACHMENT 3**

**Oyster Creek Generating Station**

**Docket No. 50-219**

**License No. DPR-16**

**License Amendment Request**

**"Increase Safety Valve As-Found Setpoint Tolerance from ±1% to ±3%"**

**Proposed Technical Specifications Bases Marked-Up Pages**

**The pages included in this attachment are:**

**PAGES**

2.3-6
4.3-3

The Rod Worth Minimizer is not required beyond 10% of rated power. The ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed, by maintaining a minimum recirculation flow of $39.65x10^6$ lb/hr in range 10 a complete rod withdrawal initiated at 35% of rated power or less would not result in violating the fuel cladding safety limit. Therefore, a rod block on the IRMs at less than 35% of rated power would be adequate protection against a rod withdrawal transient.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship. Therefore, the worst-case MCPR, which could occur during steady-state operation, is at 115% of the rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of the rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gains.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (5). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of main steam isolation valve closure while operating at 1930 MWt, followed by (1) a reactor scram on high neutron flux, (2) failure of the recirculation pump trip on high pressure, (3) failure of the turbine bypass valves to open, and (4) failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 9 safety valves are required to turn the pressure transient. The ASME B&PV Code allows ~~a ±1% of working pressure (1250 psig) variation in the lift point of the valves.~~ This variation is recognized in Specification 4.3.

an as-found ±3% of setpoint pressure variation in the lift point of the valves. The as-left safety valve setpoint tolerance requirement will remain ±1% per GE NEDC-31753P (approval letter dated March 8, 1993) recommendation.

the existence of ASME Section XI. For this reason, the degree of access required by ASME Section XI is not generally available and will be addressed as "requests for relief" in accordance with 10 CFR 50.55a(g).

Experience in safety valve operation shows testing in accordance with Section XI of the ASME-Boiler and Pressure Vessel Code is adequate to detect failures or deterioration. ~~The tolerance value is specified in Section I of the ASME Code at ±1% of design pressure.~~ An analysis has been performed which shows that with all safety valves set ~~12 psig higher~~ 36 psig higher, the safety limit of 1375 psig is not exceeded.

Conductivity instruments continuously monitor the reactor coolant. Experience indicates that a check of the conductivity instrumentation at least every 72 hours is adequate to ensure accurate readings. The reactor water sample will also be used to determine the chloride ion content to assure that the limits of 3.3.E are not exceeded. The chloride ion content will not change rapidly over a period of several days; therefore, the sampling frequency is adequate.

. The as-found setpoint tolerance value is specified in Section XI of the ASME Code at ±3% of design pressure.