

WOLF CREEK

NUCLEAR OPERATING CORPORATION

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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Subject: Docket No. 50-482: Application For Amendment To Technical Specification Section 3.4.10, Pressurizer Safety Valves

Gentlemen:

This letter transmits an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS). This request proposes to revise Technical Specification 3.4.10, Pressurizer Safety Valves, by reducing the pressurizer safety valve set point and increasing the set point tolerance.

A Safety Evaluation for the proposed license amendment request is provided in Attachment I; a No Significant Hazards Consideration Determination, in Attachment II. Attachment III is the related Environmental Impact Determination. Marked up technical specification pages are provided in Attachment IV. Attachment V provides a list of licensing commitments made in this submittal.

It is requested that the proposed changes be reviewed prior to July 2000 to support the next WCGS refueling outage, and be issued after the implementation of Amendment No. 123, Improved Technical Specifications. The requested amendment will be implemented within 60 days of NRC approval.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State Official. If you should have any questions regarding this submittal, please contact me at (316) 364-4034, or Mr. Michael J. Angus at (316) 364-4077.

Very truly yours,


Richard A. Muench

RAM/rlr

Attachments

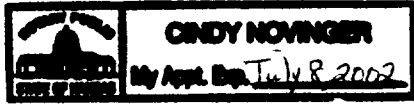
cc: V. L. Cooper (KDHE), w/a
J. N. Donohew (NRC), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Richard A. Muench, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By *Richard A. Muench*
Richard A. Muench
Vice President Engineering

SUBSCRIBED and sworn to before me this 21st day of October, 1999.



Cindy Novinger
Notary Public

Expiration Date July 8, 2002

ATTACHMENT I
SAFETY EVALUATION

Safety Evaluation

Description of the Proposed Change

This request for a license amendment proposes to revise Wolf Creek Generating Station (WCGS) Technical Specification 3.4.10, Pressurizer Safety Valves, by reducing the pressurizer safety valve set pressure and increasing the set point tolerance. WCGS Technical Specification Limiting Condition for Operation (LCO) 3.4.10 requires all three pressurizer safety valves (PSVs) to be operable with lift settings ≥ 2461 psig and ≤ 2509 psig (2485 psig $\pm 1\%$). The proposed change would reduce the limit for the nominal lift setting to 2460 psig $\pm 2\%$ and allow opening pressures of ≥ 2411 psig and ≤ 2509 psig. Following testing, the required lift setting will be within $\pm 1\%$ of the nominal value (2460 psig). No change is proposed to the maximum allowed lift setting of 2509 psig.

Reason for the Proposed Change

The difficulty of PSVs to meet the Technical Specification as-found set point tolerance of $\pm 1\%$ has been an industry problem for many years, not specific to WCGS. As a result, several licensees have requested and been granted license amendments to relax (i.e., increase) the set point tolerance for these valves, similar to the one proposed herein by Wolf Creek Nuclear Operating Corporation (WCNOC). The change being proposed by WCNOC is essentially the same change approved in Amendments 76 and 98 for the Vogtle plants, in which the Vogtle units PSV set points were lowered 1% and the set point tolerance increased from $\pm 1\%$ to $\pm 2\%$.

Basis for Proposed Change

The proposed change to the WCGS Technical Specification requirements for the pressurizer safety valve (PSV) lift setting proposes a reduced nominal set point and an increased set point tolerance. The impact of the proposed change on the safety analyses has been evaluated and the results are discussed in the following paragraphs.

Loss of Coolant Accident (LOCA) and LOCA-Related Evaluations

The WCGS Updated Safety Analysis Report (USAR) Loss of Coolant Accident (LOCA) related analyses result in a decreasing Reactor Coolant System (RCS) pressure and therefore, do not challenge the PSV opening pressure. Actuation of the PSVs is not modeled in these analyses. The change in the PSV set point and set point tolerance has no effect on the LOCA peak clad temperature or LOCA hydraulic forces analyses.

Steam Generator Tube Rupture Evaluation

The Steam Generator Tube Rupture (SGTR) event results in a decreasing RCS pressure, and does not challenge the PSV opening pressure. Actuation of the PSVs does not occur in these analyses. The change in the PSV set point and set point tolerance has no effect on the SGTR analysis.

Non-LOCA Related Evaluations

Several of the non-LOCA safety analyses conservatively model PSV actuation. The analysis models consider both the valve set point and set point tolerance, as defined in the WCGS Technical Specifications (currently equal to $\pm 1\%$ with a nominal pressure set point of 2485 psig), and the valve performance (e.g., valve accumulation and relief capacity). Depending on the transient being analyzed, it may be conservative to model either a maximum or minimum valve

opening pressure. The proposed change in the PSV tolerance from $\pm 1\%$ to $\pm 2\%$ with a reduction in the set point from 2485 psig to 2460 psig allows a decrease in the valve minimum opening pressure and, thereby, provides earlier pressurizer relief and a reduced RCS pressure. The proposed change does not affect the maximum opening pressure assumed in the non-LOCA analyses since the maximum PSV opening pressure is not impacted. Therefore, only those transients for which it is conservative to minimize the RCS pressure (i.e., Departure from Nucleate Boiling (DNB) and pressurizer overfill concerns) are potentially impacted by the proposed tolerance change. The events from Chapter 15 of the WCGS USAR which are potentially impacted are listed below:

- Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (USAR Section 15.1.2)
- Excessive Increase in Secondary Steam Flow (USAR Section 15.1.3)
- Inadvertent Opening of a Steam Generator Relief or Safety Valve (USAR Section 15.1.4)
- Steam System Piping Failure (USAR Section 15.1.5)
- Turbine Trip (USAR Section 15.2.3)
- Loss of Non-Emergency AC Power to the Station Auxiliaries (USAR Section 15.2.6)
- Loss of Normal Feedwater Flow (USAR Section 15.2.7)
- Feedwater System Pipe Break (USAR Section 15.2.8)
- Partial Loss of Forced Reactor Coolant Flow (USAR Section 15.3.1)
- Complete Loss of Forced Reactor Coolant Flow (USAR Section 15.3.2)
- Reactor Coolant Pump Shaft Seizure (USAR Section 15.3.3)
- Uncontrolled RCCA Bank Withdrawal at Power (USAR Section 15.4.2)
- RCCA Misoperation [dropped RCCA events only] (USAR Section 15.4.3)
- Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (USAR Section 15.4.4)
- Inadvertent Operation of the ECCS During Power Operation (USAR Section 15.5.1)
- Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (USAR Section 15.5.2)
- Inadvertent Opening of a Pressurizer Safety or Relief Valve (USAR Section 15.6.1)

For each of the above events, with the exception of the inadvertent operation of the emergency core cooling system (ECCS) during power operation transient, automatic actuation of pressurizer power-operated relief valves (PORVs) and/or pressurizer spray are modeled to conservatively minimize RCS pressure. The effect of the pressurizer spray and the PORV relief is sufficient in these analyses, including the turbine trip DNB analysis, to prevent PSV actuation during the applicable portions of these transients. The analysis results and conclusions for these events are not impacted. Therefore, the conclusions in the USAR remain valid.

Although not presented in USAR Chapter 15, the steamline break with coincidental rod cluster control assembly (RCCA) withdrawal at power analysis was also evaluated for the proposed PSV set point and set point tolerance change. The pressurizer pressure does not reach the revised PSV opening pressure (2411 psig), and therefore, the analysis is not impacted.

The current USAR analysis of the inadvertent operation of the ECCS during power operation transient credits operator action at 10 minutes to terminate ECCS injection in order to preclude a pressurizer water-solid condition. A revised pressurizer overfill analysis was performed to address concerns relative to the timing of operator action to terminate ECCS injection, and to demonstrate the pressurizer would not overfill conservatively assuming a reactor trip occurs at event initiation. The calculations performed to

determine the minimum departure from nucleate boiling ratio (DNBR) were not impacted.

The revised inadvertent operation of the ECCS analysis also bounds plant operation with a reduced minimum PSV opening pressure of 2411 psig. The analysis models pressurizer spray actuation which conservatively minimizes the pressurizer pressure. The analysis also credits operator action to terminate ECCS injection prior to 8 minutes in order to preclude a pressurizer water-solid condition. This is a change from the current analysis which assumes operator action to occur within 10 minutes. The reduced operator action time is supported by plant operations through a plant procedure. ECCS isolation can be accomplished, by isolating the boron injection tank, in 8 minutes or less from event initiation following an inadvertent ECCS actuation event. The peak pressurizer pressure for this analysis remains less than the minimum PSV opening pressure of 2411 psig. Therefore, no steam or water relief would occur through the pressurizer safety valves and the analysis is not impacted by the proposed change.

In summary, the non-LOCA safety analyses have been reviewed. For all non-LOCA events the analyses support the change in PSV set point and set point tolerance from 2485 psig $\pm 1\%$ to 2460 psig $\pm 2\%$.

Containment Integrity Evaluation (Short Term/Long Term LOCA Release, Steam Line Break)

Changing the PSV set point and set point tolerance has no effect on the short term/long term LOCA or main steam line break containment integrity analyses. These events result in reduced RCS pressure, and therefore do not challenge the PSV set point. The calculated mass and energy releases are not impacted, therefore, the resulting containment pressure and temperature conditions are not impacted.

Radiological Consequences and Post-LOCA Hydrogen Generation Evaluation

Changing the PSV set point and set point tolerance has no effect on the radiological analyses consequences or post-LOCA hydrogen generation.

Probabilistic Safety Assessment (PSA) Evaluation

The Probabilistic Safety Assessment (PSA) and its success criteria are not sensitive to the specific PSV set point setting or tolerance; the success or failure of the valve to open is more important. The same events currently assumed (such as station blackout) to result in valve actuation will continue to result in valve actuation for this relatively small change. Additionally, the lower valve setting does not increase the probability that an event will occur which will result in the valve opening. Therefore, the expected frequency of valve openings considered in the PSA will not increase and the lower valve lift setting and increased tolerance has no significant affect on the PSA.

Safety Systems Set Points Evaluation

The change in the PSV set point and set point tolerance has no effect on the Reactor Protection or Engineered Safety Features Systems trip set points.

Control Systems Evaluation

The proposed change has been evaluated for its impact on the NSSS control systems and the NSSS components and transients. With a $\pm 2\%$ tolerance and a set point of 2460 psig, the PSV minimum actuation pressure could potentially be as low as 2411 psig. The pressurizer PORV actuation set point is 2335 psig. Therefore, the margin between the PORV and PSV actuation set points could be as low as 76 psi, which is a reduction of 49 psi from the current 125

psi margin. Even with the 30 psi pressure control uncertainty, the actuation set point margin of 76 psi is considered adequate and the PORVs are expected to continue to actuate before the PSVs during Condition 1 transients. As such, the proposed change will not have any adverse effect on the control systems.

The PSVs are credited in the NSSS components design transients and actuate at the maximum opening pressure for the loss of load upset condition design transients. Since no change to the maximum opening pressure is proposed, these transients are not impacted. Note that a reduction in the PSV set point is a benefit to the loss of load design transient. The PSVs are not modeled in any other design transients.

Mechanical Components and Systems Evaluation

The PSVs provide overpressure protection for the RCS. The upper limit of over pressure protection is based on the surge of reactor coolant produced as a result of a transient from full load that produces the largest surge. The PSVs are sized on the basis of steam flow from the pressurizer to accommodate this surge at the set pressure and the accumulation defined by the ASME Code. No credit is taken for the relief capability provided by the power operated relief valves during this surge. Operability of the PSVs ensures that the RCS pressure will be limited to 110 percent of pressurizer design pressure.

The change in the lower limit of the PSV tolerance from -1% to -2% with a reduction in the nominal set pressure from 2485 psig to 2460 psig does not challenge the upper limit of the overpressure protection. Any evaluations performed on an overpressure transient conservatively assume the upper limit of the PSV tolerance as the pressure to which the RCS is subjected. The proposed change to the lower tolerance limit of the set pressure means that an overpressure transient may be terminated at a pressure that is lower than assumed in the analysis. It has also been determined that the design transients are not adversely affected because the limiting transients are not sensitive to the pressure tolerance decrease. Therefore, the primary system pressure boundary is not challenged by the PSV lower tolerance limit change.

The change in the upper limit of the PSV tolerance from +1% to +2% with a reduction in the nominal set pressure from 2485 psig to 2460 psig does not challenge the upper limit of the overpressure protection. The maximum opening set pressure is not changed, and therefore, does not impact analyses performed for overpressure transients. Although the lower valve set pressure would result in a lower qualified valve flow rate, the slightly lower valve flow rate would be more than compensated for by the reduced valve opening pressure.

The change in PSV set point and set point tolerance was also reviewed to determine the effect on the thermal and hydraulic analysis described in USAR Section 3.9(B).3.3. The change to the set point and set point tolerance does not change the conclusions of the existing thermal hydraulic analysis for the pressurizer safety and relief system.

Instrumentation and Control (I&C) Systems Evaluation

Since the change in PSV set point and set point tolerance does not increase the mass and energy releases, the proposed set point and set point tolerance change will not adversely affect the instrumentation and control systems or the I&C equipment environmental qualification.

Conclusion

Based on the above, the proposed change can be implemented without adverse impact on the safety analyses, mechanical systems and components, and instrumentation and control systems.

ATTACHMENT II

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

Proposed Change

This request for a license amendment proposes to revise Wolf Creek Generating Station (WCGS) Technical Specification 3.4.10, Pressurizer Safety Valves, by reducing the pressurizer safety valve set point and increasing the set point tolerance. WCGS Technical Specification Limiting Condition for Operation (LCO) 3.4.10 requires all three pressurizer safety valves to be operable with lift settings ≥ 2461 psig and ≤ 2509 psig (2485 psig $\pm 1\%$). The proposed change would reduce the limit for the nominal lift setting to 2460 psig $\pm 2\%$ and allow opening pressures of ≥ 2411 psig and ≤ 2509 psig. Following testing, the required lift setting will be within $\pm 1\%$ of the nominal value (2460 psig). No change is proposed to the maximum allowed lift setting of 2509 psig.

The following sections discuss the proposed change under the three standards of 10 CFR 50.92.

Standard I - Involves a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

Any evaluations performed on an overpressure transient conservatively assume the upper limit of the pressurizer safety valve (PSV) tolerance as the pressure to which the reactor coolant system (RCS) is subjected. The proposed change to the lower tolerance limit of the pressure set point means that an overpressure transient may be terminated at a pressure that is lower than assumed in the analysis. It has also been determined that the design transients are not adversely affected because the limiting transients are not sensitive to the pressure tolerance decrease. Therefore, the primary system pressure boundary is not challenged by the PSV lower tolerance limit change. The change in the upper limit of the PSV tolerance does not challenge the upper limit of the overpressure protection. The maximum opening set pressure is not changed, and therefore, does not impact analyses performed for overpressure transients. Although the lower PSV set point would result in a lower qualified valve flow rate, the slightly lower valve flow rate would be more than compensated for by the reduced valve opening pressure. The change to the PSV set point and set point tolerance does not change the conclusions of the existing thermal hydraulic analysis for the pressurizer safety and relief system. The design function of the valves is not being changed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the USAR.

Standard II - Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The proposed change would allow the PSV minimum actuation pressure to be as low as 2411 psig. The pressurizer power-operated relief valve (PORV) actuation set point is 2335 psig. Therefore, the margin between the PORV and PSV actuation set points could be as low as 76 psi, which is a reduction of 49 psi from the current 125 psi margin. Even with the 30 psi pressure control uncertainty, the actuation set point margin of 76 psi is considered adequate and the PORVs are expected to continue to actuate before the PSVs during Condition 1 transients. As such, the proposed change will not have any adverse effect on the control systems. Except for the reduced lower set point, the design and operation of the PSVs are not being changed. The maximum opening pressure is not being changed. The only effect of this change would be that the PSVs could open at a lower pressure, but still above the PORV actuation set point. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

Standard III - Involve a Significant Reduction in the Margin of Safety

The PSVs provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The PSVs are designed to prevent the system pressure from exceeding the system safety limit, 2735 psig, which is 110% of the design pressure. The change in the upper limit of the PSV tolerance from +1% to +2% with a reduction in the nominal set point from 2485 psig to 2460 psig does not challenge the upper limit of the overpressure protection. The maximum opening pressure set point is not changed, and therefore, does not impact analyses performed for overpressure transients. The change to PSV set point and set point tolerance does not change the conclusions of the existing thermal hydraulic analysis for the pressurizer safety and relief system. For all non-LOCA events the analyses support the change in PSV set point and set point tolerance from 2485 psig $\pm 1\%$ to 2460 psig $\pm 2\%$. The change in the PSV set point and set point tolerance also has no effect on the Reactor Protection or Engineered Safety Features Systems trip set points. Thus, the proposed change does not involve a significant reduction in any margin of safety.

Based on the above discussions, it has been determined that the requested technical specification revision does not involve a significant increase in the probability of consequences of an accident or other adverse conditions over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

ATTACHMENT III
ENVIRONMENTAL IMPACT DETERMINATION

Environmental Impact Determination

10 CFR 51.22(b) specifies the criteria for categorical exclusions from the requirement for a specific environmental assessment per 10 CFR 51.21. This amendment request meets the criteria specified in 10 CFR 51.22(c)(9) as specified below:

(i) the amendment involves no significant hazards consideration

As demonstrated in Attachment II, the proposed changes do not involve any significant hazards considerations.

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

The proposed change does not involve a change to the facility or operating procedures that would cause an increase in the amounts of effluents or create new types of effluents.

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

This change has no relation to occupational radiation exposure, either individual or cumulative.

Based on the above, it is concluded that there will be no impact on the environment resulting from this change and the change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to requiring a specific environmental assessment by the Commission.