



**UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001**

July 23, 2008

The Honorable Dale E. Klein
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: STRETCH POWER UPRATE APPLICATION FOR THE MILLSTONE POWER STATION, UNIT 3

Dear Chairman Klein:

During the 554th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 2008, we discussed the Stretch Power Uprate (SPU) application for Millstone Power Station, Unit 3, (MPS3) and the associated NRC staff's Safety Evaluation. Our Subcommittee on Power Uprates discussed this application on July 8, 2008. During these reviews, we had the benefit of discussions with the staff, representatives of the Dominion Nuclear Connecticut, Inc. (DNC) (the licensee), and members of the public. We also had the benefit of the documents referenced.

RECOMMENDATION

The application for power uprate at MPS3 should be approved.

BACKGROUND

By letter dated July 13, 2007, DNC submitted a license amendment request to the NRC to increase the maximum authorized power level of MPS3 from the current licensed thermal power level of 3,411 MWt to 3,650 MWt, and change the technical specifications, as necessary, to support operation at the SPU level, which is an increase of approximately 7 percent.

DISCUSSION

Plant Modifications

The licensee identified several modifications to be made in support of the MPS3 power uprate. One significant modification is the addition of new actuation logic (permissive - 19) that monitors the Reactor Coolant System (RCS) pressure and permits automatic high-pressure cold leg injection only when a Safety Injection (SI) signal is received in conjunction with low RCS pressure. This modification extends the time available for operators to recover from an inadvertent SI and improves overall safety margin. Another modification of note is the

elimination of the automatic rod withdrawal signal. Under some accident conditions, automatic rod withdrawal exacerbates the transient. Elimination of this signal improves safety margin. The staff has evaluated all of the modifications proposed by the licensee and found them satisfactory. The staff has also evaluated all piping and components in the power block and concluded that the components and systems are of sufficient capacity to operate safely at the stretch power level.

Fuel System and Nuclear Design

The staff reviewed the impact of the proposed SPU on:

- Fuel system and nuclear design
- Thermal-hydraulic design
- Overpressure protection
- Accident and transient analyses
- Loss-Of-Coolant Accident (LOCA)
- Anticipated Transient Without Scram (ATWS)

The scope of steady state and transient analyses for MPS3 is consistent with the NRC generic SPU guidelines. Analyses and evaluations are based on NRC-approved methodologies, analytical methods, and codes.

The mechanical analysis of the fuel is based on multiple fuel types. The nuclear and thermal hydraulic analysis is based on Robust Fuel Assembly (RFA/RFA2) fuel. These analyses, performed by Westinghouse for DNC, show the SPU will result in a slight increase in linear heat rate and a slightly less peaked core design. The staff's review included those transients specified in the NRC Review Standard for Extended Power Uprates (RS-001). Based on its review, the staff has concluded that the results of these analyses were acceptable as noted in the staff's Safety Evaluation.

DNC's evaluations demonstrate that an acceptable core design may be achieved at the uprated power level. In accordance with the NRC's reload licensing process, cycle-specific analyses will be performed using an NRC-approved methodology to demonstrate compliance with fuel safety limits.

Several accidents/transients, namely, overpressure protection, inadvertent Emergency Core Cooling System (ECCS) actuation, and uncontrolled rod withdrawal at low power, warranted additional staff reviews.

DNC's analysis indicates that the maximum pressurizer liquid volume following a loss of feedwater event increased from 1061 to 1731 cubic feet, as compared to a total pressurizer volume of 1800 cubic feet. DNC indicated that the margin can be increased by manually tripping the reactor coolant pumps. Additionally, the power operated relief valves are qualified for water relief.

Departure from Nucleate Boiling Ratio

To maintain acceptable Departure from Nucleate Boiling Ratio (DNBR) margins at the SPU condition, the radial peaking factor has been reduced from the current level of 1.70 to 1.65, automatic rod withdrawals have been eliminated, an electronic filter has been installed to smooth measured hot leg temperatures, and the power range high neutron flux setpoint has been decreased from 118 percent to 116.5 percent. With the proposed modifications, the DNBR margins at the SPU condition are acceptable.

LOCA Analysis

Westinghouse evaluated large-break LOCAs using the ASTRUM best-estimate methodology (this is a change from BART/BASH Appendix K Method used in previous generations of reload analyses). Small-break LOCAs have been evaluated using the NOTRUMP code. The small-break and large-break LOCA evaluations show substantial margins to the acceptance criteria.

Containment Analysis

MPS3 uses a large, dry, sub-atmospheric containment, with a design pressure of 45 psig and a containment liner design temperature of 280°F. In 1991, MPS3 changed its technical specifications to raise the normal containment pressure range from 8.9 - 12 psia to 10.6 - 14 psia. Calculations for postulated accidents at SPU conditions, which were performed using the GOTHIC code, show that containment pressure and containment liner temperature remain within design limits. The peak containment pressure following a large-break LOCA is 41.4 psig at the SPU power level as compared to the design pressure of 45 psig. Containment liner temperature remains acceptable at 241°F versus a design limit of 280°F.

During the recirculation phase, containment recirculation spray pumps and the ECCS pumps take suction from the containment sump. MPS3 has already modified its sump strainers to address the concerns of Generic Safety Issue - 191. Net positive suction head calculations show that containment overpressure credit is not needed for any design-basis accident.

Reactor Vessel Integrity

Reactor vessel integrity is most influenced by its chemical composition and the neutron fluence exposure to the vessel wall extrapolated to the end of plant life. Important vessel characteristics are evaluated by extrapolating material sample properties from surveillance capsules which are installed in the core region. At MPS3, there were six surveillance capsules, three of which remain in the reactor. The licensee has stated that the steady state core design will retain low-leakage characteristics. The proposed SPU will increase the reactor vessel neutron fluence by an amount proportional to the power increase of 7 percent. The staff has accepted this conclusion. Based on results obtained from the most recent surveillance capsule tests, DNC has estimated that the shift in pressurized thermal shock reference temperature (RT_{PTS}) at the end of extended life of the core at the stretch power rating will not exceed 270°F for plate and axial weld materials and will not exceed 300°F for circumferential weld materials. The upper shelf energy of the MPS3 reactor vessel will remain above the screening limit of 50 ft-lbs through the end of the current plant license term.

The staff's evaluation of long-term vessel integrity is that the RT_{PTS} and upper shelf energy are such that reactor vessel integrity will be maintained at the SPU condition for 54 effective full power years, without reanalysis.

Flow Accelerated Corrosion

The licensee uses the CHECWORKS code to predict the rate of wall thinning that could result from the higher flow rates when power is increased to the SPU level. DNC has used, and plans to continue to use, the CHECWORKS code to prioritize the elements of the monitoring program to detect loss of piping wall thickness. This program has been successfully implemented at MPS3 and is satisfactory for use at the SPU power level.

CONCLUSION

The licensee has submitted an acceptable application for SPU condition, including a revised supplemental environmental report, proposed technical specification changes, and changes to the associated bases, proposed instrument and control changes (including updating the simulator), and changes to the operator training program. The staff's review was thorough and complete. Based on our review of the matters associated with this application, we conclude that the DNC application for MPS3 SPU should be approved.

Sincerely,



William J. Shack
Chairman

REFERENCES

1. Letter from Gerald T. Bischof, Vice President Nuclear Engineering, Dominion Nuclear Connecticut, Inc. to the U.S. Nuclear Regulatory Commission, Subject: "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3 License Amendment Request, Stretch Power Uprate," July 13, 2007 (ML072000386, ML072000390, ML072000391, ML072000394, ML072000396, ML072000400, ML072000402).
2. Memorandum from Timothy J. McGinty, Acting Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation, to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards, Subject: "Millstone Power Station, Unit 3 – Advisory Committee on Reactor Safeguards Review of Proposed Stretch Power Uprate Amendment Draft Safety Evaluation," June 11, 2008 (ML081420393, ML081420426, ML081420440).
3. U.S. Nuclear Regulatory Commission, "Review Standard for Extended Power Uprates," RS-001, December 2003 (ML033640024).
4. U.S. Nuclear Regulatory Commission, "NRC Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors," September 13, 2004 (ML042360586).

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Letter to the Honorable Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS, dated July 23, 2008

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UNIT 3

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