NORTHEAST UTILITIES

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November 1, 1990

Docket No. 50-423
B13627
Re: 10CFR50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Proposed Change to Technical Specifications
Cycle 4 Reload Submittal

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 3. These proposed changes revise numerous Technical Specifications in support of refueling and operation for Millstone Unit No. 3 with the VANTAGE 5 Hybrid (5H) improved fuel design.

The proposed Technical Specification changes for Millstone Unit No. 3 Cycle 4 primarily result from changes in three areas: (1) change in fuel design, (2) use of improved analytical methodologies, and (3) associated fuel-/core-related changes:

1. Change in Fuel Design

Millstone Unit No. 3 is currently operating in Cycle 3 with a Westinghouse 17 x 17 standard (STD) fueled core. Millstone Unit No. 3 Cycle 4 and subsequent core loadings will have fuel assemblies that incorporate the low-pressure drop zircaloy grid and the intermediate flow mixer (IFM) grid. This upgraded feature is known as VANTAGE 5H with IFM and has been submitted as an Addendum(1) to the "VANTAGE 5 Reference Core Report," WCAP-10444-P-A.(2) The VANTAGE 5H Report has received generic

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(1) "VANTAGE 5H Fuel Assembly," WCAP-10444-P-A, Addendum 2, April 1988 and letter from W. Johnson (Westinghouse) to M. W. Hodges (NRC), Supplemental Information for WCAP-10444-P-A, Addendum 2, "VANTAGE 5H Fuel Assembly," dated July 29, 1988.

(2) "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A, September 1985

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approval.[3] The VANTAGE 5H fuel assembly design evolved from the current VANTAGE 5, optimized Fuel Assembly (OFA) and Standard Fuel Assembly designs. In addition to the above-mentioned VANTAGE 5H design features, Millstone Unit No. 3 Cycle 4 reload will also contain several VANTAGE 5 design features and other upgraded fuel design features used in the Cycle 3 core. Millstone Unit No. 3 Cycle 3, Region 5 fuel has already incorporated the VANTAGE 5 integral fuel burnable absorbers (IFBAs), the VANTAGE 5 axial blanket design, the VANTAGE 5 extended burnup design, the VANTAGE 5 reconstitutable top nozzle (RTN) design feature, as well as debris filter bottom nozzles (DFBNs), snag-resistant grids, and standardized fuel pellets.

By letter dated December 30, 1983, Westinghouse requested NRC to review the topical report WCAP-10444, "Westinghouse Reference Core Report, VANTAGE 5 Fuel Assembly." The NRC Staff reviewed WCAP-10444 and concluded that the Westinghouse topical report was an acceptable reference to support plant-specific application of VANTAGE 5 provided certain conditions were addressed. The conditions are addressed for the Millstone Unit No. 3 application requesting the use of VANTAGE 5H fuel assemblies. Attachment 3 provides responses to those conditions.

2. Use of Improved Analytical Methodologies

The existing thermal-hydraulic analysis of the 17 x 17 STD fuel used in the Millstone Unit No. 3 core is based on the standard thermal and hydraulic methods and the W-3 (R-Grid) DNB correlation as described in the Millstone Unit No. 3 Final Safety Analysis Report (FSAR). The DNB analysis of the core containing both 17 x 17 STD and VANTAGE 5H fuel assemblies have been modified to incorporate the WRB-1 and WRB-2 DNB correlations,[4][5] the revised Thermal Design Procedure (RTDP),[6] and

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(3) A. C. Thadani (NRC) letter to R. A. Wiesemann (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-10444-P-A, Addendum 2, VANTAGE 5H Fuel Assembly," November 1, 1988, and Clarifications on the Safety Evaluation of the Topical Report WCAP-10444-P-A Addendum 2, January 5, 1989.

(4) VANTAGE 5 Fuel Assembly Reference Core Report, WCAP-10444-P-A, September 1985.

(5) New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles With Mixing Vane Grids, WCAP-8762-P-A, July 1984.

(6) Revised Thermal Design Procedure, WCAP-11397-P-A, April 1989.

improved THINC IV Modeling.[7] The new DNB correlations take credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations. The W-3 correlation and the standard methods continue to be used when conditions are outside the limit of the WRB-1 or WRB-2 DNB correlation and of the RTDP. As a result of the new DNB correlations' improved accuracy, confidence at a 95/95 level that the limiting power rod will not experience DNB is provided with a limiting DNBR value of 1.17 versus the existing 1.30.

3. <u>Associated Fuel-/Core-Related Changes</u>

Numerous changes are being made which are closely related to the change in fuel type. Three of these changes represent physical changes to be made at the plant. These three changes are:

a. Thimble plug deletion--these devices will be removed over several cycles.

b. Rod cluster control assembly (RCCA) parked position change--the full-out position of the RCCA banks will vary in the range of 222-231 steps withdrawn.

c. Refueling water storage tank (RWST) boron concentration increase--the boron concentration maintained in the RWST will increase from the range 2300-2600 ppm to 2700-2900 ppm.

One of these changes involves a change in the plant procedures:

Relaxed axial offset control (RAOC)--the plant will utilize this method of control which allows a wider axial offset band compared to constant axial offset control which is currently being used.

The remainder of the changes in this category are modifications to the safety analysis procedures or inputs:

a. Increase in $F_{\Delta H}$ and F_Q.
b. Reactivity coefficient changes.
c. RTDP.
d. Revised non-RTDP uncertainties.
e. Reduced shutdown margin.
f. Modified OTΔT and OPΔT trip setpoints.

A description of the proposed Technical Specification changes related to the VANTAGE 5H fuel upgrade is provided in Attachment 1. In addition, two additional changes to the Millstone Unit No. 3 Technical Specifications described

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(7) Improved THINC IV Modeling for PWR Design, WCAP-8762, April 10, 1978.

in the attachment are not directly related to the Cycle 4 reload. The first set of changes are administrative in nature. Since they do not reduce the effectiveness of the existing or proposed Technical Specifications, they do not involve a significant hazard consideration. The other set of changes are due to cable insulation resistance effects on the engineered safety features (ESF) instrumentation. (Table 3.3-4 of Technical Specifications). The revised pages of the Technical Specifications are provided in Attachment 2.

The Plant Safety Evaluation[8] (PSE) provides the safety evaluation for the region-by-region reload transition from the Millstone Unit No. 3 Cycle 3 core with 17 x 17 STD fuel assemblies to a core containing the VANTAGE 5H upgraded features described above. This safety evaluation includes the mechanical, nuclear, thermal and hydraulic, and accident evaluations. This evaluation was accomplished utilizing the methodology described in WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology."

As requested by the NRC Project Manager for Millstone Unit No. 3, two copies of the PSE are being forwarded directly to him. Using the RTDP requires a review of temperature, pressure, power, and flow uncertainties used in the safety evaluations. For Millstone Unit No. 3, the uncertainties are calculated based on the installed plant instrumentation or special test equipment and Millstone Unit No. 3 calibration and calorimetric procedures. A report[9] has been prepared to describe the uncertainty evaluation. Since this report contains information proprietary to Westinghouse Electric Corporation, it will be submitted under a separate cover.

SAFETY ASSESSMENT

The PSE prepared by Westinghouse for Millstone Unit No. 3 includes accident evaluations. Specifically, Westinghouse reviewed all the FSAR Chapter 15 transients to determine which events need to be reanalyzed for Cycle 4. For each of the potentially impacted non-Loss-of-Coolant Accident (non-LOCA) transients, consideration was given to effects of the VANTAGE 5H fuel design features and modified safety analysis assumptions discussed in Section 5.1 of the PSE. As dictated by event-specific sensitivities, a decision was made for each transient with regard to the need for formal analysis, as opposed to simply evaluating the impact of the subject features and assumptions. An additional issue that requires consideration for the Millstone Unit No. 3 analyses is the possible difference in relative behavior of the three-loop cases as compared to those with four loops operating. For certain events, previous licensing basis analyses clearly demonstrate that four-loop results

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(8) Plant Safety Evaluation for Millstone Generating Station, Unit 3, VANTAGE 5H Fuel Upgrade, August 1990.

(9) WCAP-12621, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Millstone 3," August 1990.

bound those for three loops operating. In those cases, an evaluation of three-loop operation was performed for the VANTAGE 5H transition, even if analysis was determined to be necessary for the associated four-loop case. Table 5.1.2-1 of the PSE documents for each of the potentially impacted non-LOCA transients whether an analysis was performed or an evaluation was sufficient to assess the impact of the VANTAGE 5H transition. In addition, Section 5.2 of the PSE describes the large-break and small-break Loss-of-Coolant Accident (LOCA) analysis. The boron dilution analysis is being finalized at this time. The Technical Specification changes related to this analysis and the results of the analysis will be provided in a future submittal. The proposed changes to Technical Specifications result in some changes in the consequences of the design basis accidents. It is difficult to directly compare the results of the previous analyses to the results of the revised analyses, because of changes in methodology. For three non-LOCA accidents (locked rotor transient for four-loop and three-loop, and inadvertent ECCS Operation), the consequences have somewhat increased; however they are within the applicable acceptance criteria.

Large-break LOCA analyses were performed for a complete spectrum of breaks for VANTAGE 5H fuel and the limiting case with standard fuel. These analyses were performed for both four-loop and three-loop operation. Justification for use of approved LOCA methodology for the four-loop case in three-loop operation is given in paragraph 5.2.2 of the PSE. Peak clad temperature (PCT) in all cases analyzed remains below the 2200°F acceptance limit with the worst case being 2133°F versus 2132°F in the previous analyses.

Small-break LOCA analyses were performed for four-loop operation only, as the three-loop operation case is bounded by the four-loop case. PCT for the small-break LOCA analysis with four loops in operation was 1890°F versus 1483°F in the previous analysis. The consequences, however, are still calculated to be within the acceptance criteria for PCTs.

There are two aspects of these changes that have a potential effect on the radiological consequences of analyzed accidents.

The first is a change in the calculated amount of fuel that experiences cladding failure as a result of a locked rotor accident during three-loop operation. The value for Cycle 3 was 1 percent, whereas for the proposed change, the value is estimated to be 3.3 percent. The current design basis accident assumed 4.4 percent cladding failure. Hence, the proposed change is bounded by the current calculation and there are no increased consequences.

The second aspect is a change in the radial peaking factor. For the VANTAGE 5H fuel upgrade, the new peaking factor of 1.7 is assumed in the PSE. This is different from the current peaking factor of 1.65. The only accident for which a peaking factor is used to calculate radiological consequences is the fuel-handling accident. Hence, this accident was reanalyzed.

The TACTIII computer code was used for this evaluation. Due to the differences in the codes used by Northeast Utilities and Stone and Webster, the fuel-handling accident was also evaluated using the peaking factor of 1.65 in order to determine the effect of the change. The following assumptions were used in this calculation:

1. Activity released is one assembly plus 50 rods of another (314 total rods).

2. All of the gap activity in the damaged rods is released and consists of 10 percent noble gases (30 percent Kr-85) and 10 percent iodines.

3. Iodine species: 75 percent elemental, 25 percent organic.

   Pool decontamination factor: 100.

5. Exhaust is via the ventilation vent on the Turbine Building.

6. EAB X/Q: 4.3(-4) sec/m^3.

7. Minimum time after shutdown for transfer of fuel is 100 hours.

8. Fuel building filter efficiencies: 95 percent.

A fuel-handling accident in containment is not analyzed because it would be less severe than the fuel accident in the fuel building due to the isolation capability of the containment. Table 1 shows the results of the doses due to a fuel-handling accident with the old peaking factor of 1.65 and the new peaking factor of 1.7. The increase in the dose is only 3 percent. The doses are well within the Standard Review Plan acceptance criteria of 75 REM to the thyroid and 6 REM to the whole body. The new doses are also at or below the doses that the NRC reviewed from the previous Stone and Webster calculation included in the existing FSAR.

In summary, the proposed changes will somewhat increase the consequences of some of the design basis transients. However, in all cases, the changes result in a calculation of acceptable consequences. Therefore, even though there is an increase in consequences, there is still no impact on the protective boundaries.

In a letter dated February 26, 1990, NNECO submitted a proposed amendment request that would allow NNECO to operate Millstone Unit No. 3 with a maximum containment pressure of 14.0 psia during Modes 1 through 4. The safety analysis for the proposed change was performed at a maximum allowable containment pressure of 14.2 psia. As a part of the VANTAGE 5H fuel upgrade for Millstone Unit No. 3, Cycle 4, an evaluation was performed for the effects of increasing the containment pressure to 14.2 psia. The evaluation concluded that the increase in the containment pressure will have no adverse impact on the mass and energy release data reported in the Millstone Unit No. 3 FSAR.

SIGNIFICANT HAZARDS CONSIDERATION

NNECO has reviewed the proposed changes in accordance with 10CFR50.92(c) and has concluded that the changes do not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

   To determine any potential impact, the proposed changes can be grouped into three general categories. These are:

   a. Changes to Technical Specifications (Cycle 4 reload) due to change in fuel design and use of improved analytical methodologies.

   b. Changes that are not related to the Cycle 4 reload and are administrative in nature such as redrawing the existing figures in Technical Specifications and renumbering of pages from the existing Technical Specifications to allow removal of "intentionally blank" pages from the Technical Specifications.

   c. Changes due to cable insulation resistance effects on ESF instrumentation.

   Each of these groups of changes is discussed in more detail below.

   a. As discussed above, Westinghouse reviewed all FSAR Chapter 15 accidents and transients to determine which events needed to be reanalyzed for Cycle 4, assuming a mixed core or a core containing only VANTAGE 5H fuel. The PSE documents for each of the potentially impacted transients or accidents whether an analysis was performed or an evaluation was sufficient to assess the impact of the VANTAGE 5H transition. The evaluation of the impact of the proposed changes is discussed in Section 5.9.3 of the PSE. The evaluation addressed a full core of VANTAGE 5H as well as transition cores consisting of VANTAGE 5H and standard fuel. Four-loop and three-loop operation also was addressed. No increase in the probability of occurrence of any accident was identified, but extensive reanalysis, as described in the PSE was required to demonstrate compliance with the proposed changes to the Technical Specifications. These reanalyses applied methods which have been previously found acceptable by the NRC. The results, which includes transition core effects, show changes in consequences of accidents previously analyzed. However, as stated in SAFETY ASSESSMENT above, the increase in consequences are not significant and the results are clearly within pertinent acceptance criteria.

The increase in RWST boron concentration has been evaluated with respect to its effect on components in a postaccident environment. The corrosion of aluminum, zinc, and stainless steel components are affected by the pH of their environment. The volume and NaOH concentration of the Chemical Addition Tank have been modified to neutralize the effect of the higher boric acid concentration coming from the RWST. There is thus no effect on the performance of components containing these materials.

b. The renumbering of pages or redrawing of existing figures for clarity purposes do not reduce the effectiveness of the Technical Specifications. Also, these changes do not affect the existing or proposed limiting conditions for operation or surveillance requirements. Therefore, there is no impact on the design basis accidents.

c. The design basis accident which credits safety injection (SI) on pressurizer pressure low and steam line pressure low is a steam line break. Since the safety analysis SI setpoint for a steam line break is not changed, there is no increase in the consequences. Since the engineered safety features system will function as before, there is no increase in the probability or consequences of a steam line break.

On the basis of this review, NNECO concludes that there is no significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

a. Evaluations have been performed on the fuel assemblies, RCCAs, and other safety-related equipment to confirm that their function and reliability are not negatively impacted due to the upgrade to VANTAGE 5H and other changes described in Attachment 1. The conclusion was made that there was no negative impact on safety as a result of the proposed changes. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the fuel transition. The presence of VANTAGE 5H fuel assemblies in the core or the revised analytical assumption have no adverse effect and do not challenge the performance of any other safety-related system. Therefore, NNECO concludes that the proposed changes do not create any new or different kind of accident from those previously analyzed.

b. Since these changes do not affect plant operation, the potential for an unanalyzed accident is not created. No new failure modes are introduced.

c. The proposed changes have not modified the plant response and no new failure modes are introduced. Therefore, the potential for an unanalyzed accident is not created.

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

a. The upgrade to VANTAGE 5H fuel and the other changes discussed in Attachment 1 were evaluated against the applicable acceptance criteria. The criteria are discussed in Section 5.9.2 of the PSE and are summarized below.

(1) <u>Fuel-related criteria</u>:

(a) DNBR greater than safety analysis limit

(b) PCT less than 2200°F for LOCA

(c) Fuel centerline temperature less than 4900°F (BOL), 4800°F (EOL)

(d) Average fuel pellet enthalpy less than 200 cal/gm for rod ejection

(e) Fuel melting limited to 10 percent for rod ejection

(f) Remainder of 10CFR50.46 criteria (clad oxidation, hydrogen generation, coolable geometry, long-term cooling)

(2) <u>RCS pressure boundary-related criteria</u>:

(a) Pressure less than 110 percent for Condition II and III events

(b) Pressure less than 120 percent for Condition IV events

(3) <u>Containment pressure</u>:

(a) Pressure less than design pressure

A significant increase in DNB margin is provided by the IFM grids which are a feature of VANTAGE 5H. Additional DNB margin is created by the use of the WRB-2 correlation and the use of the revised thermal design procedure. This additional DNB margin can then be used to increase the FΔH limit as well as remove thimble tube plugging devices. The non-LOCA analyses confirm that the DNB design basis is met for Standard 17 x 17 and VANTAGE 5H fuel.

The non-LOCA analyses confirmed that the acceptance criteria for rod ejection were met. The results also showed that peak RCS pressure remains below 110 percent of design for all events.

Margin to PCT is obtained primarily through the use of improved LOCA evaluation models (BASH for large-break LOCA analysis and NOTRUMP for small-break LOCA analysis). Also, the zircaloy grids and IFM grids used in VANTAGE 5H fuel provide additional PCT margin. This margin can then be used to implement the changes described in Section 5.0.1 of the PSE. The LOCA analysis considered four- and three-loop operation as well as transition core effects. The results for Standard fuel provided the most limiting PCT. The LOCA analysis demonstrates that the PCT acceptance criteria contained in 10CFR50.46 of 2200°F continues to be met as well as the criteria related to clad oxidation and maximum hydrogen generation.

LOCA-related analyses demonstrate that the margin of safety with respect to blowdown reactor vessel and loop forces is preserved, thus satisfying the 10CFR50.46 criteria that the core remain amenable to cooling after a LOCA. Long-term cooling and post-LOCA subcriticality concerns are satisfied by increasing the RWST boron concentration. This in turn affects the concern with boron precipitation. The analysis shows that hot-leg switchover must be accomplished 9 hours after accident initiation, which is acceptable.

Mass and energy release from LOCA and steam line break are not adversely affected by any of the changes discussed in Section 5.3 of the PSE. Therefore, the input to the containment and subcompartment analysis continues to be valid.

In summary, performance of analyses and evaluations for the upgrade to VANTAGE 5H and associated changes have confirmed that the operating envelope defined by the technical specifications continues to be bounded by the revised analytical basis, which in no case exceeds the acceptance limits. Therefore, the margin of safety provided by the analyses in accordance with these acceptance limits is not reduced.

b. Since the proposed changes do not affect the consequences of any accident previously analyzed, there is no reduction in the margin of safety.

c. The proposed changes do not increase the consequences of an accident previously analyzed, and the protective boundaries are not affected by these proposed changes. Therefore, there is no reduction in the margin of safety.

Moreover, the Commission has provided guidance concerning the application of the standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51FR7751) of amendments that are considered not likely to involve a

significant hazards consideration. Example (iv) provides that a significant hazards consideration finding is unlikely for:

> A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or com‸nent specified in the Standard Review Plan; for example, a change resulting from a small refinement of a previously used calculation model or design method.

This example appears applicable to the portion of the proposed license amendment request involving a change in the fuel design and improved analytical methodology. As discussed above, for non-LOCA accidents, the consequences have somewhat increased. However, they are within the applicable acceptance criteria. Similarly, for LOCA accidents, the proposed changes result in an increase in the consequences (PCT); however in all cases, they remain within the acceptance criteria for PCT. For a fuel-handling accident, the increase in the consequences (Table 1) is not significant. Therefore, the proposed changes do not involve a significant hazards consideration. For other Technical Specification changes, there is no increase in the consequences of an accident previously analyzed. Therefore, based on the above, NNECO concludes that the proposed Technical Specification changes do not involve a significant hazards consideration.

ENVIRONMENTAL CONSIDERATION

NNECO has reviewed accidents analyzed in the Millstone Unit No. 3 FSAR with respect to the radiological source term and radiological consequences in association with the transition to VANTAGE 5H fuel. Included in this review is consideration of extended fuel burnup of 45,000 MWD/MTU for the batch average discharge. This extended burnup has a peak fuel rod average burnup of < 60,000 MWD/MTU. The existing Millstone Unit No. 3 Technical Specification permits use of reload fuel with a maximum nominal enrichment of 5.0 weight percent U-235. This allows a burnup level to 50,000 MWD/MTU (60,000 MWD/MTU peak rod burnup). The safety consideration associated with reactor operation with higher enrichment and extended irradiation has been evaluated by the NRC and the Staff concluded that such changes would not adversely affect plant safety.[10] The proposed change to VANTAGE 5H fuel will operate within the existing Technical Specification requirements; therefore, the existing source terms are applicable. For those events in Chapter 15 which had to be reevaluated or reanalyzed as a result of the proposed changes, no increase in the amount of fuel failures was calculated. Since there was no increase in failed fuel and the fission product source terms are unaffected, there is no

(10) D. H. Jaffe letter to E. J. Mroczka, Millstone 3--Issuance of Amendment, dated July 28, 1989.

radiological consequences resulting from the proposed changes. As demonstrated above, the proposed amendment request does not involve a significant hazards consideration. The proposed amendment request also does not result in a significant increase in individual or cumulative occupational radiation exposure. This is supported by the NRC as documented in a public notice, "Extended Burnup Fuel Use in Commercial LWRs; Environmental Assessment and Finding of No Significant Impact," dated February 23, 1988, and supplemented by the NRC in a public notice for Millstone Unit No. 3 (54FR27082). Therefore, NNECO concludes that there are no significant radiological or nonradiological environmental impacts associated with the proposed amendment request.

The Millstone Unit No. 3 Nuclear Review Board has reviewed and approved the attached proposed revisions and has concurred with the above determinations.

In a letter dated January 20, 1988, the NRC requested that for future Millstone Unit No. 3 reload core cycles, NNECO provide an expected anticipated transients without scram (ATWS) moderator temperature coefficient (MTC) at equilibrium Xenon conditions, pending the Staff's evaluation of a forthcoming Westinghouse Owners Group (WOG) response to a Staff request for information on ATWS MTC dated June 12, 1987. On March 1, 1989, the WOG submitted to the NRC a report, WCAP-11993, "Joint Westinghouse Owners Group/Westinghouse Program: Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs," (nonproprietary). This report presents results of a joint WOG and Westinghouse effort to quantify the frequency of core damage resulting from ATWS for Westinghouse pressurized water reactors (PWRs) and demonstrates compliance with the ATWS Rule as specified in SECY-83-293 for Westinghouse PWRs. Although the WOG submittal demonstrates compliance with the ATWS Rule, a plant-specific response to Millstone Unit No. 3 is provided below.

For the Cycle 4 core, full-power BOL equilibrium Xe MTC is -7.2 pcm/°F which is more negative than the -5.5 pcm/°F assumed in Westinghouse Letter NS-EPR-83-2833 (E. P. Rahe, Westinghouse, to S. J. Chilk, NRC, October 3, 1983).

The proposed amendment request needs to be approved to support Cycle 4 operation and prior to entry into Mode 4 after the February 1991 refueling outage. NNECO requests that these proposed changes be approved and effective by March 6, 1991. This would allow specific applicable Technical Specifications to be in place prior to any affected Mode change.

In accordance with 10CFR50.91(b), we are providing the State of Connecticut with a copy of this proposed amendment.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: E. J. Mroczka
Senior Vice President

BY: C. F. Sears
C. F. Sears
Vice President

cc: T. T. Martin, Region I Administrator
D. H. Jaffe, NRC Project Manager, Millstone Unit No. 3
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin McCarthy, Director
Radiation Control Unit
Department of Environmental Protection
Hartford, CT 06116

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me, C. F. Sears, who being duly sworn, did state that he is Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensee herein, and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lynn M. Sheridan
Notary Public

Table 1

EAB Dose in REM Due to a Millstone Unit No. 3
Fuel-Handling Accident

	1.65 Peaking Factor	1.7 peaking Factor
Thyroid	7.38(+0)	7.59(+0)
Whole Body	4.94(-1)	5.09(-1)

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Description of Proposed Technical Specification Changes

Cycle 4

November 1990

Millstone Nuclear Power Station, Unit No. 3
Description of Proposed Technical Specification Changes, Cycle 4

Millstone Unit No. 3 plans to refuel and operate with upgraded Westinghouse fuel features (VANTAGE 5 Hybrid [5H]) and increased peaking factors and the capability of operating with up to all of the assemblies' thimble tube plugs being removed.

The transition to and all VANTAGE 5H core evaluations/analyses in the plant Safety Evaluation Report(1) were performed at a reactor thermal full power level of 3411 MWt (100 percent rated power) for four-loop operation, and 2560 MWt (75 percent rated power) for three-loop operation. The following conservative assumptions were made in the safety evaluations/analyses: 10 percent uniform steam generator tube plugging with reactor coolant system (RCS) thermal design flow rates of 378,400 gpm and 294,900 gpm for four- and three-loop operation, respectively, core bypass flow increases from 6 to 8.6 percent to account for complete thimble plug removal and the use of the Intermediate Flow Mixers (IFMs), grids, the use of Relaxed Axial Offset Control (RAOC) power distribution for both three- and four-loop operation, and increases in F_Q and $F_{\Delta H}$. For four-loop operation, a vessel/core inlet coolant temperature of 557.0°F and a vessel coolant average temperature of 587.1°F were used. For three-loop operation, a vessel/core inlet coolant temperature of 550.2°F and a vessel coolant average temperature of 579.6°F were used.

Beginning with Cycle 4, the future cycles of operations for Millstone Unit No. 3 will use increased $F^N_{\Delta H}$ and F^T_Q peaking factors. The full power $F^N_{\Delta H}$ peaking factor design limit will increase from the current value of 1.55 to 1.70. The maximum four-loop F_Q peaking factor limit will increase from the current value of 2.32 to 2.60. The maximum three-loop F_Q peaking factor limit will increase from the current value of 2.25 to 3.0 at the maximum 75 percent rated power. The K(Z) envelopes will be modified. These changes will permit more flexibility in developing fuel management schemes (i.e., longer fuel cycles, improvement of fuel economy and neutron utilization, vessel fluence reduction).

The proposed Technical Specification changes have been prepared to support the Cycle 4 reload. The proposed changes are summarized below:

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(1) Plant Safety Evaluation for Millstone Unit 3, VANTAGE 5H Fuel Upgrade, August 1990.

A. Technical Specification Changes Due to the Cycle 4 Reload

1. Definitions--Allowed Power Level

Definitions 1.43 and 1.44 have been added for APL^{ND} and APL^{BL} which were included as part of the RAOC specifications.

2. Figures 2.1-1 and 2.1-2, Reactor Core Safety Limits

The reactor core safety limits figures have been revised to reflect the Departure from Nucleate Boiling (DNB) correlation used for the standard and VANTAGE 5H fuel.

3. Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, Functional Units 6 (Overtemperature ΔT) and 7 (Overpower ΔT)

The values of Total Allowance (TA), Z, and sensor error (S) for overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) have been revised due to the new core safety limits and instrumentation uncertainties. In addition, a separate set of values (TA, Z, S) for Channels 1 and 2 (Veritrak) and Channels 3 and 4 (Rosemount) are provided. Further, the following constants included in Notes 1, 2, 3, and 4 are revised.

a. For OTΔT

(1) Note 1

	Current	Revised
K1	1.08	1.20
K2	0.01313/F°	0.02456/F°
Tau1	12 sec	8 sec
Tau4	33 sec	20 sec
K3	0.00066/psi	0.001311/psi
f(ΔI) Penalty		
Break Points	-30%, +10%	-26%, +3%
Negative Slope	3.6%/%ΔI	3.55%/%ΔI
Positive Slope	2.0%/%ΔI	1.98%/%ΔT

(2) Note 2

		Current	Revised
Allowable Value	N	2.1%ΔT	2.7%ΔT
	N-1	3.6%ΔT	2.7%ΔT

b. For OPΔT

(1) Note 3

K6	0.00129/°F	0.00180/°F

(2) Note 4

Allowable Value	2.8%ΔT	2.7%ΔT

An asterisk (*) footnote on page 2-5 has been revised and moved to page 2-6 as it applies to Functional Unit 12, Reactor Coolant Flow-Low. The minimum measured RCS flow has been provided rather than the loop design flow due to the use of the revised thermal design procedure (RTDP). The calculational method utilized to meet the DNB design basis is the RTDP.

4. Bases Sections 2.1.1, Reactor Core (Page B2-1) and 2.2.1 (Pages B2-5, B2-7)

The existing thermal-hydraulic analysis of the standard fuel used in the Millstone Unit No. 3 core is based on the standard thermal and hydraulic methods and the W-3 DNB correlation as described in the Millstone Unit No. 3 FSAR. The DNB analysis of the core containing both the standard and VANTAGE 5H fuel assemblies has been modified to incorporate the WRB-1 and WRB-2 DNB correlations. The proposed changes to Bases Sections 2.1.1 and 2.1.2 reflect that.

5. Section 3.1.1.1, Boration Control

This section has been split into two; one for Modes 1 and 2 and the other for Modes 3 and 4. The shutdown margin considered in the revised safety analysis was reduced from 1.6 percent ΔK/K to 1.3 percent ΔK/K. Therefore, the shutdown margin included in Section 3.1.1.1 has been revised accordingly. Further, Surveillance 4.1.1.1.1.e has been deleted as it applies to Mode 3, and Surveillance 4.1.1.1.2 has been revised to reflect the deletion of Section 4.1.1.1.1.e. Shutdown margin in Modes 3 and 4 is not changed here, but will be changed in a future submittal.

6. <u>Section 3.1.2.5, Borated Water Sources--Shutdown</u>

The minimum boron concentration in the refueling water storage tank (RWST) is increased from 2300 ppm to 2700 ppm. This is to provide adequate shutdown margin.

7. <u>Section 3.1.2.6, Borated Water Sources--Operating</u>

A range in the RWST boron concentration has been changed from 2300-2600 ppm to 2700-2900 ppm. This is to provide adequate shutdown margin. The shutdown margin included in Action 'a' has been revised to 1.3 percent $\Delta K/K$ to match with Section 3.1.1.1, Boration Control. Shutdown margin in Modes 3 and 4 will be changed in a future submittal.

8. <u>Section 3.1.3.4, Rod Drop Time</u>

The rod drop time has been increased from 2.2 seconds to 2.7 seconds due to the fuel design changes. The effect of this increase on safety analysis has been considered.

9. <u>Section 3.2.1.1, Axial Flux Difference--Four Loops Operating</u>

<u>Section 3.2.1.2, Axial Flux Difference--Three Loops Operating</u>

Currently, Millstone Unit No. 3 uses the constant axial offset control (CAOC) procedure for the control of axial power distribution (see Specifications 3.2.1.1 and 3.2.1.2). Beginning with Cycle 4, Millstone Unit No. 3 will have an option of the relaxed axial offset control (RAOC) or base load. This procedure defines the allowed operating space of axial flux difference (AFD) versus thermal power. The limits are selected by considering a range of axial xenon distributions which could occur as a result of large variations of AFD. Also, a direct F_q surveillance will now be used. The proposed Technical Specification sections for four and three loops operating are consistent with the Westinghouse Standard Technical Specifications (WSTS).

10. Section 3.2.2.1, Heat Flux Hot Channel Factor-$F_Q(Z)$--Four Loops Operating

Section 3.2.2.2, Heat Flux Hot Channel Factor-$F_Q(Z)$--Three Loops Operating

As stated above, Millstone Unit No. 3 will be using the RAOC. The proposed Technical Specification surveillance sections for four- and three-loop operation are based on the WSTS. F_Q measurement for three-loop operation is now based on a safety analysis full power of 75 percent. Previously, all analyses were done at 75 percent except LOCA and a portion of the locked reactor coolant pump rotor accident which used 65 percent.

11. Section 3.2.3.1, RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor--Four Loops Operating

Section 3.2.3.2, RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor--Four Loops Operating

The RCS flow now included in Technical Specification Sections 3.2.3.1 and 3.2.3.2 is the minimum measured flow due to the use of RTDP. In addition, the conservative value of the uncertainty for RCS flow for four-loop operation is ±2.4 percent flow versus the previous ±1.8 percent and for three-loop operation is ±2.8 flow versus ±2.0 percent flow. This is reflected in the proposed Technical Specification changes.

12. Table 3.2-1, DNB Parameters

The DNB parameters (indicated RCS T_{avg} and indicated pressurizer pressure) for four- and three-loop operation has been modified due to the revised instrument uncertainties.

13. Table 3.3-2, Reactor Trip System Instrumentation Response Times

The response time for the pressurizer water level-high reactor trip has been specified. This has been credited in the revised safety analysis for the Rod Cluster Control Assembly (RCCA) bank withdrawal at power transient.

14. Section 3.4.1.2, Reactor Coolant System--Hot Standby

Three reactor coolant loops are now required to be in operation in Mode 3. This is due to the revised RCCA bank withdrawal from a subcritical or low power condition accident. This is reflected in the proposed Technical Specification change.

15. Section 3.4.1.6, Reactor Coolant System--Isolated Loop Start-Up

The boron concentration of the isolated loop is increased from 2300 ppm to 2600 ppm. Shutdown margin in Modes 5 and 6 are not changed here but will be changed in a future submittal.

16. Section 3.5.1, Accumulators

The boron concentration in the accumulators has been changed from 2200 to 2600 ppm to 2600 to 2900 ppm. This is consistent with the revised safety analyses.

17. Section 3.5-2, Emergency Core Cooling Systems

A 10 percent reduction in safety injection (SI) flow has been incorporated in the safety analyses as well as a 10 gpm SI pump flow imbalance. This is reflected in surveillance 4.5.2.f.2 and 4.5.2.h.2, respectively.

18. Section 3.5.4, Refueling Water Storage Tank

The boron concentration in the RWST is increased from 2300 to 2600 pm to 2700 to 2900 ppm. The effect of this increase on safety analyses has been considered.

19. Section 3.6.2.3, Spray Additive System

The proposed change will increase the range of acceptable sodium hydroxide concentrations in the chemical addition tank (CAT) to 3.4 to 4.1 percent from the previous range of 2.4 to 3.1 percent. The volume (level) in the CAT is reduced to a range of 17,760 to 18,760 gallons from the previous range of 18,000 to 19,000 gallons.

20. Section 3.9.1.1, Refueling Operations--Boron Concentration

The proposed change will increase the boron concentration in the filled portion of the RCS and refueling canal during Mode 6 to correspond with the new minimum RWST boron concentration.

21. Bases Sections 3/4.1.1.1 and 3/4.1.1.2, Shutdown Margin

Bases Section 3/4.1.2, Boration Systems

Bases Section 3/4.1.3, Movable Control Assemblies

Bases Section 3/4.2, Power Distribution Limits

Bases Section 3/4.2.1, Axial Flux Difference

Bases Sections 3/4.2.2 and 3/4.2.3, Heat Flux Hot Channel Factor and RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor

Bases Section 3/4.2.5, DNB Parameters

Bases Section 3/4.4.1, Reactor Coolant Loops and Coolant Circulation

The above bases sections are being revised to reflect the proposed Technical Specification changes.

22. Section 6.9.1.6, Core Operating Limits Reports

Sections 6.9.1.6.a.2 and 6.9.1.6.a.3 have been revised to correct the references for specification numbers. Section 6.9.1.6.a.4, 6.9.1.6.a.5, and 6.9.1.6.a.6 have been revised to make them consistent with the revised Sections 3/4.2.1.1, 3/4.2.1.2, 3/4.2.2.1, and 3.4.2.2.2.

Section 6.9.1.6.b has been revised to reflect the methods used in the plant safety analyses for the upgraded VANTAGE 5H fuel. It is noted that these methods have been reviewed and approved previously by the NRC. However, these methods will be used for Millstone Unit No. 3 for the first time beginning with the Cycle 4 reload analyses.

B. Other Technical Specification Changes

1. Section 1, Definitions, 1-26, Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)

Reference to Specification 6.16 has been changed to 6.13. This is to correct a typographical error.

2. Figures

The following figures are redrawn for clarity. There are no technical changes to these figures.

Figure Numbers

3.4-1
3.4-2
3.4-3
3.4-4a
3.4-4b

3. Renumbering of Technical Specification Pages

Table 2.2-1, pages 2-5 through 2-11, were retyped and renumbered to accommodate the proposed changes described above (see Change A.3). Pages 3/4 2-1 through 3/4 2-24 were also retyped to accommodate the proposed changes to Section 3.2.1.1, 3.2.1.2, 3.2.2.1, 3.2.2.2, 3.2.3.1, 3.2.3.2, and Table 3.2-1. In addition, these pages were renumbered to delete the blank pages from Technical Specifications which were left by Amendment No. 50. Bases Section pages B 3/4 2-1 through B 3/4 2-6 were retyped and renumbered to accommodate the proposed changes described above (see Change A.21). As a result of the above, appropriate pages of the Technical Specification Index were revised.

C. Technical Specification Changes Due to Cable Insulation Resistance Effects on ESF Instrumentation

1. Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints

The proposed changes revise the total allowance (TA), Z, sensor error (S), and allowable value in Table 3.3-4 for pressurizer pressure low and steam line pressure low. In addition, a separate pressurizer pressure low trip value for Channels 1 and 2 (Veritrak) and Channels 3 and 4 (Rosemount) are provided. The proposed changes are accomplished by adjusting instrument settings and do not involve any physical addition, deletion, or modification to plant components. The revised pressurizer pressure low and steam line pressure low trip setpoints and other associated values such as total allowance, etc., are calculated by the methodology described in WCAP-10991, "Westinghouse Setpoint Methodology for Protection System,

Millstone Unit No. 3," which is also the method used in calculating other setpoints in Table 3.3-4 of Technical Specifications. The proposed changes are based on recent data which indicates there is an additional error due to decreased cable insulation resistance at high temperature. The new setpoints better represent the error associated with the pressure measurement; the changes will maintain the assumed performance of engineered safety features actuation systems. In addition, an attempt to regain some lost margin on the pressurizer pressure safety injection (SI) signal was made by taking credit to a lower safety analysis setpoint and by splitting the specification into a separate set of values for the different channels of instrumentation, Veritrak and Rosemount. The low steam line pressure SI setpoint used in the safety analysis is not affected by the inclusion of instrument cable insulation resistance error.