UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. DPR-28

## VERMONT YANKEE NUCLEAR POWER CORPORATION

## VERMONT YANKEE NUCLEAR POWER STATION

## DOCKET NO. 50-271

### Introduction

By letter dated June 21, 1978, Vermont Yankee Nuclear Power Corporation (the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed changes relate to the replacement of 96 fuel assemblies constituting refueling of the core for sixth cycle operation at power levels up to 1593 Mwt out to end of cycle conditions minus 2 GWD/T.

In support of the reload application, the licensee has provided the Reload 5 licensing submittal and the proposed Technical Specification changes (Reference 1), information on the VYNP Loss of Coolant Accident (LOCA) analysis (References 1 and 3), and responses to NRC requests for additional information (Reference 4).

This reload involves loading of General Electric (GE) 8x8 fuel and GE Retrofit 8x8R fuel. The description of the nuclear and mechanical design of the 8x8R fuel and the 8x8 fuel is contained in GE's licensing topical report for BWR reloads (Reference 5). Reference 5 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing 7x7, 8x8 and 8x8R fuel.

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other various design parameters are provided in Reference 5. Additional plant and cycle dependent information are provided in the reload application, (Reference 1), which closely follows the outline of Appendix A of Reference 5.

Reference 7, describes the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 5. The above mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

Our safety evaluation (Reference 7) of the GE generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel, and GE's analytical methods for nuclear, thermal-hydraulic, and transient and accident calculations as applied to mixed cores containing 7x7, 8x8 and 8x8R fuel are acceptable. Approval of the nuclear and mechanical design of 8x8 fuel was determined based on information in Reference 6 and expressed in the staff's status report (Reference 8) on that document.

Because of our review of a large number of generic considerations related to use of 8x8R fuel in mixed loadings with 8x8 and 7x7 fuel, and on the basis of the evaluations which have been presented in Reference 7, only a limited number of additional areas of review have been included in this safety evaluation. For evaluations of areas not specifically addressed in this safety evaluation, the reader is referred to Reference 7.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 6 operation of VYNPS, 36 fresh 8x8 fuel bundles of type 8D274H and 60 fresh 8x8R bundles of type 8DPB289 will be loaded into the core (Reference 1). The remainder of the 368 fuel bundles in the core will be 7x7 and 8x8 fuel exposed during the previous cycles. The fresh fuel will be loaded in a core pattern as shown in Figure 3.2 of Reference 1, which is acceptable.

Based on the data presented in section 5 of Reference 1, both the control rod systems and the standby liquid control system will have acceptable shutdown capability during Cycle 6.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 5, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling.

The safety limit MCPR for VYNPS is being raised from 1.06 to 1.07 because the distribution of fuel rod power within the 8x8R fuel bundles is flatter than that of the 8x8 fuel. The reason for the flatter power distribution is the presence of two rather than one water rods in 8x8R fuel. The issue has been addressed in Reference 7 and the 1.07 limit has been found acceptable to BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 5, for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 5. It has been proposed in Table 6.1 of Reference 1 that these conditions are applicable for VYNPS up to EOC-2 GWD/T. The applicability will be verified in the physics startup tests discussed in Section 3.0.

2.2.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the CPR below the intended operating limit MCPR during Cycle 6 operation of VYNPS. The most limiting of these operational transients up to EOC-2 GWD/T have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the critical power ratio (ΔCPR) during the earlier part of Cycle 6.

The transients evaluated were the feedwater controller failure at maximum demand, loss of a 100°F feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6.3, 7.2 and 7.4 of Reference 1 were assumed. The most limiting transients for the later part of Cycle 6 (EOC-2 GWD/T to EOC) will be completed in the next several months. The results of the EOC analysis will be then used to establish the operating limit MCPRs for the later part of Cycle 6.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Table 1 lists the ΔCPRs for the various fuel types at the specified Cycle exposure. Also included in Table 1 are the results of the maximum vessel pressure discussed in Section 2.4.

TABLE 1

| Transient | Limiting Exposure Time | ΔCPR 7x7/8x8/8x8R | Operating Limit MCPR 7x7/8x8/8x8R |
|---|---|---|---|
| Load Rejection without Bypass | EOC-2 GWD/T to EOC | +++ | +++ |
| Turbine Trip without Bypass | EOC-2 GWD/T to EOC | +++ | +++ |
| Loss of 100°F Feedwater Heater | BOC-EOC | .16/.15/.15 | 1.23/NA/NA |
| Feedwater Controller Failure | BOC-EOC | .05/.07/.07 | NA |

+++ Operating limit MCPRs for these transients which are limiting during the later part of Cycle 6 shall be determined prior to EOC-2 GWD/T.

| Condition | | | |
|---|---|---|---|
| Rod Withdrawal | BOC-EOC | NA/.15/.17 | NA/1.22/1.24 |
| Overpressurization (MSIV Closure) | Peak vessel pressure assuming one failed SRV is 1307 psi | | |
| Fuel Loading Error | BOC-EOC | NA/.24/.24 | ** |

2.3 Fuel Loading Error

The potential fuel loading errors (FLE) involving misoriented bundles and mislocated bundles have been evaluated. The analysis of the fuel loading error is discussed in Reference 5 and approved in Reference 7.

The limiting fuel loading error (a mislocated bundle) ΔCPR was calculated by the more conservative older GE analysis. For the VY plant, FLE analyses for an earlier cycle showed conservatisms of approximately 50% in ΔCPR when the old approved analysis is compared to the newly approved GE analysis (Reference 9). Even though we recognize the large conservatisms in the calculated values of the older analysis methods cannot be approved since it was not for this specific cycle. In the interim, until a new analysis is provided for Cycle 6, the ΔCPR value for the fuel loading error is accepted as 0.24 which gives credit for 25% of the previously identified conservatisms.

This ΔCPR value, when added to the safety limit MCPR of 1.07, would result in an operating limit MCPR of 1.31 for the 8x8 and 8x8R fuel. If a FLE occurs during this cycle some of the fuel rods in the bundle could experience boiling transition and fail.

In this event, one means for detection of abnormal fuel degradation at VYNPS will be accomplished by measurements of off-gas radioactivity levels at the steam jet air ejector. To assure that further fuel degradation as a result of a fuel loading error will not occur, VY has proposed the installation of a new alarm on SJAE activity. This alarm will serve to warn the operator that there could be a FLE event in the core. If an activity level of 0.236 Ci/sec persists for 15 minutes or 1.18 Ci/sec persists for one minute Technical Specifications have been added which require that the operator take action to increase the operating limit MCPRs to ≥1.31. This action assures that the worst mislocated bundle would remain above the safety limit MCPR. The activity levels are chosen to correspond to the maximum activity release expected from a single mislocated bundle. Continued operation of the plant would then be determined by the most limiting condition relative to the MCPR value of 1.31 or the Technical Specification limit of offgas listed in Technical Specification Table 3.2.4.

**See Section 2.3

In addition to the detection capabilities and Technical Specification requirements, the core verification procedures have been augmented by an independent verification by Yankee Atomic Electric Company personnel and the NRC Office of Inspection and Enforcement.

In summary, we find the above procedures which require operating limit MCPRs as shown in Table 1 be raised to 1.31, in the event of indicated fuel degradation, in addition to the Technical Specification limits on offgas acceptable for this cycle of operation.

2.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure eyent, has been performed in accordance with the requirements of Reference 7. As specified in Reference 7, the sensitivity of peak vessel pressure to failure of one SRV has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure (1375 psi) to allow for the failure of at least one valve. Therefore the limiting overpressure events as analyzed by the licensee is acceptable.

2.5 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the Natural Circulation - 105% Rod Line intersection (which is the least stable physically attainable point of operation) are below the stability limit.

Because operation in the natural circulation mode is prohibited by Technical Specifications, there will be added margin to the stability limit. We find this is acceptable.

2.6 Accident Analysis

2.6.1 ECCS Appendix K Analysis

Input data and results for the ECCS analysis have been given in References 1 and 3. The information presented fulfills the requirements for such analyses outlined in Reference 7.

We have reviewed the analyses and information submitted for the reload and conclude that the VYNPS will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: (1) it is operated with the "MAPLHGR VERSES AVERAGE PLANAR EXPOSURE" values given in Tables 3.11-1A through 3.11-1G of the Technical Specifications, (2) is it operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant-Accident, as described in Section 2.2.) and 2.3)

2.6.2 Control Rod Drop Accident

The worst case control rod drop accident (CRDA) can occur under startup conditions when the characteristic parameters for the accident meet the requirements for bounding analyses described in Reference 5, this is adequate to show that the design basis of 280 cal/gm peak fuel enthalpy for a startup C 'A is met (Reference 7).

For VYNPS, the characteristic accident parameters for the worst startup CRDA satisfy the requirements for bounding analyses as described in Reference 5. Therefore the postulated CRDA would be $\leq$280 cal/gm which is acceptable.

3.0 Physics Startup Testing

The licensee in accordance with Technical Specification requirements and Reference 2 will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed in the transient and accident analysis calculations will be met during Cycle 6. The tests will also check that the core is loaded as intended and that the incore monitoring system and control rod worths and operations are functioning as expected. A written report of the startup tests will be provided to NRC within approximately 45 days as discussed in Reference 2.

4.0 Technical Specifications

The changes to the Technical Specification as proposed by the licensee are acceptable with the following exceptions:

1. The operating limit MCPR for the 7x7 fuel shall be changed to 1.23. Other changes in MCPR have been required upon evidence of high gas activity so that the core wide safety limit will not be violated for the worst case fuel loading error. This is discussed in Section 2.3.

2. The proposed wording in the Technical Specifications relating to action if limiting values of ALHGR, LHGR and MCPR are exceeded was not included in this amendment. As discussed with the licensee, these matters will be considered separately for a possible later license amendment, after the licensee has provided further supporting arguments.

5.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

6.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 10, 1978

References


1. Letter, D. E. Vanderburgh, to NRC WVY 78-59, dated June 21, 1978.

2. Letter, R. H. Groce, to NRC, WVY 78-64, dated July 12, 1978.

3. Letter, W. P. Johnston, to NRC, WVY 77-71, dated August 12, 1977.

4. Letter, D. E. Vanderburgh, to NRC, WVY 78-89, dated September 20, 1978.

5. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P, May 1977.

6. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.

7. NRC Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-20411-P), April 1978.

8. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April 1975.

9. Letter, Ronald Engel, GE to Darrell Eisenhut, NRC, Fuel Assembly Loading Error, June 1, 1977.