



# VERMONT YANKEE NUCLEAR POWER CORPORATION

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FVY 82-39

April 8, 1982

REPLY TO:  
ENGINEERING OFFICE

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United States Nuclear Regulatory Commission  
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation

- References:
- (1) License No. DPR-28 (Docket No. 50-271)
  - (2) Letter, VYNPC to USNRC, WVY 77-115, dated December 29, 1977: Proposed Change No. 70
  - (3) Letter, VYNPC to USNRC, WVY 79-9, dated January 30, 1979: Proposed Change No. 77
  - (4) Letter, USNRC to VYNPC, dated February 26, 1981
  - (5) Letter, VYNPC to USNRC, FVY 81-102, dated July 1, 1981
  - (6) Letter, VYNPC to USNRC, FVY 81-153, dated November 4, 1981

Subject: Augmented Inservice Inspection and Leakage Detection Requirements

Dear Sir:

Pursuant to Section 50.59 of the Commission's Rules and Regulations, Vermont Yankee Nuclear Power Corporation hereby proposes the following modification to Appendix A of the Operating License as a supplement to Reference (3).

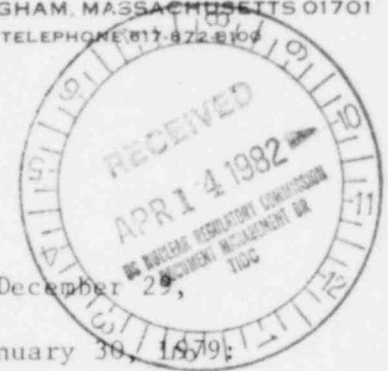
Proposed Change: We propose to supplement Reference (3) and amend the Operating License DPR-28 as follows:

Modify the Technical Specifications and Reference (3) as delineated in Enclosure A to comply with the intent of NUREG-0313, Revision 1.

Basis and Reason for Change:

This change was prompted by Reference (4), which requested that we propose changes to our Technical Specifications incorporating as appropriate the provisions of NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." Reference (5) originally responded to Reference (4), presenting Vermont Yankee's position with regard to NUREG-0313, Revision 1, and providing a schedule for the submittal of the required Technical Specifications. Reference (6) revised this schedule, and this supplement fulfills that commitment.

Vermont Yankee had previously addressed the concerns of NUREG-0313 in Proposed Change No. 70 to Technical Specifications, which was submitted to the Commission with Reference (2). We request that Proposed Change No. 70 be withdrawn from further consideration in light of this submittal.



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We take exception to the requirement of NUREG-0313, Revision 1 that we provide for additional reactor coolant leakage monitoring in our Technical Specifications. The leakage limits that are analyzed in our FSAR and presently included in our Technical Specifications are based on:

- 1) Detection of a percentage of the flow that would be expected from the critical-size crack assuming that the entire unidentified leakage comes from a single crack in the most limiting line;
- 2) The capability of RCIC to make up for this leakage independently of the feedwater system, normal AC power, and the core standby cooling system; and
- 3) The capability of the drywell sumps to remove this leakage under normal conditions.

The basis for establishing the frequency of drywell leakage monitoring is to attain an assurance that corrective action can be taken before the integrity of the nuclear process barrier is significantly compromised.

NUREG-0313, Revision 1, requests changes to be made to our existing limits, but it provides no safety evaluation or analysis to show that these changes are required to reduce the probability of a catastrophic pipe rupture. BWR operating experience and analytical research continue to confirm that IGSCC in austenitic stainless steel results in ductile failure characterized by tight cracks which exhibit readily detectable leakage significantly before the critical crack length is reached. Since NUREG-0313 supports these conclusions, and due to the lack of technical justification therein for the requested changes, Vermont Yankee intends to retain its present leakage limits.

#### Safety Considerations:

Vermont Yankee has performed the reviews requested in Reference (4) and has determined that certain reactor coolant pressure boundary systems contain nonconforming material as defined in NUREG-0313, Revision 1.

In keeping with our commitment to ALARA, and for technical reasons previously discussed in Reference (2) and reiterated below, replacement of such material in systems further categorized as "service-sensitive" is considered impracticable. This determination is technically justified because:

- 1) Type 304 stainless steel piping, because of its highly ductile nature, is extremely unlikely to suffer sudden, brittle-type fracture, even in cases where relatively large flaws or cracks are present. Should through-wall cracking occur, the existing drywell leakage detection system will assure that corrective action is taken in a timely manner.
- 2) Plant conditions can be maintained within acceptable safety limits with the Emergency Core Cooling Systems in the event of a design basis loss-of-coolant accident (LOCA).