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Docket No. 50-271

Vermont Yankee Nuclear Power Corp.
ATTN: Mr. Albert A. Cree, President
77 Grove Street
Rutland, Vermont 05701

Gentlemen:

In your letter of April 3, 1973 (L. H. Heider to Directorate of Licensing), you state that you have completed the modification of the actuation logic for the Primary Containment Isolation System, which we approved in our letter dated March 6, 1973 (V. A. Moore to Albert Cree). In our letter of March 6, 1973, we stated that appropriate changes to the Technical Specifications will be made when you complete the modifications. These changes add a 40% steam flow trip to the containment isolation system in place of a high water level trip when the reactor is not in the run mode, as protection against reactor vessel blowdown from inadvertent opening of all bypass valves. This change in the trip actuation logic is designed to reduce the potential for unnecessary isolations of the primary system. As we concluded in our letter of March 6, 1973, the modifications you have made and the above cited changes to the Technical Specifications do "not present significant hazards considerations not described or implicit in the safety analysis report and that there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change."

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station are changed by removing the pages numbered 41, 44, and 64, all dated 12/1/71, and replacing them with the enclosed revised pages 41, 44, and 64, all dated 4/11/73.

Sincerely,
Original signed by
Walter Butler

for Voss A. Moore, Assistant Director
for Boiling Water Reactors
Directorate of Licensing

Enclosure:
Revised pages to
Technical Specifications

LB

cc: See next page

OFFICE ▶	L:BWR-1	L:BWR-1	L:AD:BWR		
SURNAME ▶	GPOwsley:ld	WRButler WB	VAMoore WB		
DATE ▶	4/10/73	4/10/73	4/10/73		

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TABLE 3.2.2.1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Condition for Operation are not Satisfied (Note 2)
2	Low-Low Reactor Vessel Water Level	$\geq$6' 10.5" above the top of active fuel	A
2 of 4 in each of 2 channels	Steam Line Area High Temperature	$\leq 200^{o}F$	B
2/steamline	Steam Line High Flow	$\leq$120 % of rated flow	B
2 (Note 1)	Main Steam Line Low Pressure	$\geq$850 psig	B
2 (Note 6)	Steam Line High Flow	$\leq$40% of rated power	B
2	Low Reactor Vessel Water Level	Same as Reactor Protection System	A
2	Main Steam Line High Radiation	$\leq$7 X background at rated power	B
2	High Drywell Pressure	Same as Reactor Protection System	A
1	Trip System Logic	———	A

4/11/73

TABLE 3.2.2 NOTES

1. The main steam line low pressure need be available only in the "Run" mode.

2. If the minimum number of operable instrument channels is not available for one trip system, that trip system shall be tripped. If the minimum number of operable instrument channels is not available for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate an orderly shutdown and have reactor in the cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have reactor in "Hot Standby" within 8 hours.

3. Close isolation valves in system and comply with Specification 3.5.

4. One trip system arranged in a two-out-of-two logic.

5. One trip system arranged in a one-out-of-two twice logic.

6. The main steam line high flow is available only in the "Refuel", "Shutdown", and "Startup" modes.

Pressure instrumentation is provided which trips when reactor pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1295°F; thus, there is no release of fission products other than those in the reactor water.

The HPCI and/or RCIC high flow, steam supply pressure, and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings are such that core uncovering is prevented and fission product release is within limits.

The instrumentation which initiates CSCS action is arranged in a dual channel system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. Permanently installed circuits and equipment may be used to trip instrument channels. In the non-fail safe systems which require energizing the circuitry, tripping an instrument channel may take the form of providing the required relay function by use of permanently installed circuits. This is accomplished in some cases by closing logic circuits with the aid of the permanently installed test jacks or other circuitry which would be installed for this purpose.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCHFR does not decrease to 1.0. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, six IRM's or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. The RBM is an operational guide and aid only and is not needed for rod withdrawal.

The APRM rod block trip is flow referenced and prevents a significant reduction in MCHFR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCHFR is maintained greater than 1.0.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCHFR approaches 1.0.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented.