VERMONT YANKEE NUCLEAR POWER CORPORATION

SEVENTY SEVEN GROVE STREET

RUTLAND, VERMONT 05701

VYV-3071

REPLY TO:
P. O. BOX 157
VERNON, VERMONT 05354

November 14, 1973

Director
Directorate of Licensing
United States Atomic Energy Commission
Washington, D.C. 20545

REFERENCE: Operating License DPR-28
Docket No. 50-271
Abnormal Occurrence No. AO-73-31

Gentlemen:

As defined in Section 6.7.B.1 of the Technical Specifications for the Vermont Yankee Nuclear Power Station, we are reporting the following Abnormal Occurrence as AO-73-31.

On November 7, 1973, at 2101, while the plant was in a shutdown condition and while the required Control Rod Friction testing was being performed on control rod 26-23, a reactor scram occurred initiated by a high-high flux signal from the Intermediate Range Neutron Monitoring System.

An immediate investigation revealed that rod 30-23 was in the fully withdrawn position while rod 26-23 was being withdrawn for its friction test. This situation was a result of inadequate implementation of administrative or procedural controls and constituted a violation of Section 1.A.8 of the Technical Specifications.

Section 14.5.3.2 of the Vermont Yankee FSAR deals with control rod withdrawal errors when the reactor is at power levels below the power range. The most severe case occurs when the reactor is just critical at room temperature and an out-of-sequence rod is continuously withdrawn. The results of these analyses indicate that no fuel damage will occur due to the rod withdrawal.

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The station had been in a planned shutdown condition since September 28, 1973, in order to perform core reconstitution and interconnection of the Advanced Off-Gas System. On November 7, 1973, work had progressed to the point where final core loading had been completed. At that point, it became desirable to perform final core verification concurrent with control rod timing and friction tests. In order to accomodate both requirements, it was necessary to install jumpers to the refuel interlock portion of the Reactor Manual Control System in order to allow traversing of the television camera mounted on the fuel grapple while performing control rod friction and timing tests. Although the intent of installing the jumpers was reasonable and proper, the ensuing implementation of this program went beyond the scope of original intent. The reasons for this were the inadequacy of interdepartmental communications; in addition, certain procedures demonstrated inadequacies, specifically AP 504, Lifted Leads Log, OP 408, Control Rod Drive System. Further, the control rod friction testing was being performed in accordance with a Startup Test Procedure; an approved operating procedure did not exist. The result of the jumper installation was a condition of interlocks which did not prevent withdrawal of more than one control rod at a time. The operating personnel were not adequately informed of the jumpered interlock status; control rod testing was resumed concurrent with core verification. As control rod testing progressed, rod 30-23 was inadvertantly left in the fully withdrawn position. After core verification was completed, and since the reactor operator was not cognizant that control rod 30-23 was still withdrawn, an adjacent lateral control rod 26-23 was selected and its continuous withdrawal begun in preparation for the friction test. Between notch position 20 and 26, the operator noticed rapid source range monitor response. He immediately initiated control rod insertion. At this time a full rod scram was initiated by the intermediate range monitor high-high flux signals. It was later demonstrated that control rod 30-23 digital position display was functioning properly. The reactor operator could not explain his failure to observe the indication of control rod 30-23 being fully withdrawn.

The immediate action of the Shift Supervisor on duty was to notify higher plant management and to determine if personnel were on the refueling floor during the incident and to request dosimeter readings of all personnel at that location on the conservative assumption that a criticality may have occurred. Five personnel were on the refueling floor at the time in areas not adjacent to the open vessel. The maximum dosimeter reading of the personnel involved was 25 mr; however this total was accumulated over a five hour work period and not attributable to this incident alone. It was also verified that the local area monitors, the continuous air monitor on the refueling floor, as well as the Reactor Building Ventilation Exhaust monitors showed no increased level of radiation.

Following the arrival on site of the Assistant Plant Superintendent and the Reactor Engineer, further evaluation determined that the scope of installed jumpers was beyond the original intent. The jumpers were removed and it was decided to perform a subcriticality test on each of the two involved control rods which verified their proper effectiveness. Based upon the above evaluations, it was determined that no fuel failure had occurred and no radiation problem existed. The installed interlock jumpers were removed and a verification test conducted to determine that the rod block interlock was restored.

On November 8, 1973, consultation with off-site higher management and engineering personnel resulted in the removal of the involved fuel assemblies from the core for sipping and visual inspection. No evidence of leakage or visual degradation was observed. The following is a listing of the assemblies examined and their location:

Assembly Number	Core Location
VT 164*	27-22
VT 171*	29-22
VT 167	27-24
VT 175	29-24
VT 049	31-32

In addition, a two rod critical test was conducted utilizing control rods 30-23 and 26-23. As a result of this test, it was determined that with control rod 30-23 in the fully withdrawn position, criticality was achieved when control rod 26-23 was withdrawn to notch 16.

The film badges assigned to personnel on the refueling floor at the time of the incident were sent out for processing. The results of the badge bearing neutron sensing indicated a total of 50 mr beta-gamma and zero neutron exposure. This total badge exposure was accumulated over a two day work period. The results of the remaining four badges indicated that two badges measured 20 mr beta-gamm and two badges measured 0 mr beta-gamma.

Subsequent calculations by General Electric Co. verified criticality at notch 16 on rod 26-23 with rod 30-23 fully withdrawn. Further calculation by General Electric Co. determined that with rod 30-23 fully withdrawn and rod 26-23 at notch 26, the excess reactivity was 0.67% ΔK, and had rod 26-23 been fully withdrawn, the excess reactivity would have been 0.97% ΔK.

* These assemblies were visually inspected.

General Electric personnel with recognized competency in the area of core kinetics, and in particular control rod drop accidents, uncontrolled withdrawal incidents, etc., did a qualitative evaluation of what transpired based on the above statistical information. An estimate based upon many previous calculations of a similar nature, was that the bounding results were as follows. The peak fuel center line temperature would have increased no more than 500°F and the peak clad temperature would have increased no more than 50°F from the starting conditions. Therefore, the fuel center line temperature was no higher than 585°F and the peak clad temperature was no higher than 135°F.

Plant management has discussed at length with all involved personnel the significance of this incident and stressed the areas of inadequate personnel performance. Further, a review has been made of the past and present performance of the employees directly involved in this incident. This assessment has determined that these employees are capable, sincere, and conscientous and that every reasonable assurance exists that they are adequately qualified in all respects to continue in their present assigned job responsibilities.

Upon completion of an indepth evaluation of the total incident and the various now apparent inadequacies, it is concluded that no singular outstanding area was predominant.

The Plant Operations Review Committee (PORC), met to review the incident and made the following recommendations and/or conclusions:

1. The original intent of the jumpers was reasonable; however, the final condition obtained was improper and the applied jumpers should have been removed immediately following the completion of core verification.

2. The results obtained from the fuel assemblies sipped and inspected on November 8, 1973, showed no observed indications which would preclude plant startup.

   The Plant Operations Review Committee questioned whether adequate sensitivity to sipping still existed considering the elapsed shutdown time and recommended taking two known leakers previously removed during this shutdown and sipping to determine if adequate sensitivity still existed. On November 14, 1973, two fuel assemblies were sipped in an attempt to prove I^{131} and I^{132} sensitivity. The positive results obtained verify the adequacy of sipping sensitivities observed on November 8, 1973.

3. Subcritical testing results of the two involved control rods and the management evaluation of the plant condition on November 7, 1973, were deemed sufficient to permit further control rod friction testing following the incident.

4. Administrative Procedure AP 504 "Lifted Lead Log" was not adhered to. Jumper installation was not recorded in the general plant log.

5. All plant procedures relating to control rod movement shall be modified to reflect interlock requirements imposed by the reactor mode switch position.

6. Specific operating procedures addressing control rod friction and settling tests shall be developed.

7. The present AP 504, Lifted Leads Log procedure, is inadequate and a PORC sub-committee has been appointed to review and/or revise the current procedure.

8. Until the above appointed PORC sub-committee performs its task, no installation of jumpers or lifted leads shall be performed on the circuitry associated with the Reactor Protection System, the Primary Containment Isolation System, any ECC System, the Reactor Manual Control System and any refuel interlock until approved by PORC.

9. No further two (2) rod critical testing shall be performed on side by side rods.

10. The following items contributed to the incident:

   a. A lack of definition on the interfacing of responsibilities on an interdepartmental level.

   b. Failure by plant supervision to exercise rigorous skepticism relative to abnormal or inadequate plant conditions that are encountered.

   c. Operator error.

At the request of the Manager of Operations, the Nuclear Safety Audit and Review Committee met in a special meeting on November 14, 1973, to review the incident. The NSAR returned the following conclusions:

1. No unreviewed safety question was involved.

2. The health and safety of the public and plant personnel was not impaired.

3. There is no undue risk to the health and safety of the public if the plant is started up and operated in accord with the proposed schedule.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

B.W. Riley

B.W. Riley
Plant Superintendent

BWR/EPC/kbd

Preliminary
Abnormal Occurrence
Report No. 73-29

SUBJECT: Violation of the Technical Specifications, paragraph 3.8.A, in that during power operation, by virtue of the fact that an inoperable snubber existed on steam lines to each of the two Isolation Condensers, both condensers were considered to be inoperable.

This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15B and D. Notification of this event, as required by the Technical Specifications, paragraph 6.6.2.a, was made to AEC Region I, Directorate of Regulatory Operations, by telephone on Saturday, November 3, 1973, at 0850, and by telecopier on Monday, November 5, 1973, at 1315.

SITUATION: While conducting an inspection of the hydraulic shock and sway arrestors (snubbers) located on various systems in the Reactor Building, but outside of the Drywell, the accumulators on one unit on the steam line to the A Isolation Condenser and one unit on the steam line to the B Isolation Condenser were found to be devoid of fluid. Both units were considered to be inoperable.

CAUSE: To be determined upon inspection.

REMEDIAL ACTION:

As per the requirements of the Technical Specifications, paragraph 3.8.D, an orderly plant shutdown was commenced upon noti-

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fication of the situation at 1830. Meanwhile, immediate efforts were made to refill the snubber accumulator on the A Isolation Condenser steam line. This action was completed by 1845, returning the snubber to service. The load drop which had been started was halted and output was again increased to the initial level. Follow-up action included replacement of the accumulator on the snubber installed on the B Isolation Condenser steam line, then replacement of the entire snubber unit on the A Isolation Condenser steam line. This action was completed by 1910 Friday evening. A follow-up check was then made on Saturday evening to insure that no fluid loss problems existed.

SAFETY SIGNIFICANCE:

Amendment 67 to the FDSAR details the requirements for at least one Isolation Condenser to be available as a heat sink in the event of a Loss of Coolant Accident. In this situation, it can be postulated that this requirement might not have been met, had an earthquake occurred which would require the snubber to be fully operable.

Prepared by: [signature] Date: 11/5/73