RO Bulletin No. 73-6

November 26, 1973

INADVERTENT CRITICALITY IN A BOILING WATER REACTOR

We recently received an abnormal occurrence report from the Vermont Yankee Nuclear Power Corporation relating to an inadvertent criticality incident that was experienced at their Vermont Yankee facility. A copy of the abnormal occurrence report is attached to this Bulletin to provide you with pertinent details of this event.

At the time of the inadvertent criticality incident, the reactor vessel and primary containment heads were removed, the refueling cavity above the reactor vessel was flooded, control rod friction tests were in progress, the rod worth minimizer was bypassed, and core verification had been in progress. As a result of the incident, no measurable radioactivity was released, no fuel damage resulted and no personnel exposures were experienced. The incident is currently under review and evaluation by the Regulatory Staff.

Action requested by this bulletin is contained in Section A.

A. Action Requested by Licensees

In light of this occurrence you are requested to take the following actions. Upon completion of these actions you are requested to inform this office in writing, within 45 days of the date of this letter, of the status of each item identified in each paragraph and subparagraph listed below:

1. Procedural Review

a. Control Rod Drive Operating and Testing Procedures

(1) Conduct a review of your control rod drive operating and testing procedures to determine that approved procedures exist for all operations and tests.

(2) Verify that appropriate prerequisites are included in the procedures to require testing of associated interlock and protective features before control rod testing is permitted.

(3) Assure that prerequisites and detailed instructions are provided that demonstrate compliance with technical specification requirements and design bases.

Dupe

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b. Bypass Installation Procedures (Jumpers or Lifting of Leads)

Assure that existing bypass installation procedures have been conservatively reviewed for technical adequacy and for administrative controls.

c. Radiation Protection Procedures

Assure that procedures for access control and personnel accountability in areas subject to accidents are current.

d. Shift Transition Procedure (Turnover)

Assure that complete and detailed procedures are in effect that provide instructions for a proper and conservative turnover of shift responsibilities. Such procedures must include requirements for communicating the status of all safety related equipment and conditions.

2. Management Controls

Assure that your management controls that are in effect provide for qualified technical and administrative reviews and approvals of temporary circuitry changes and temporary off-normal plant conditions. This review should assure that the responsibilities and requirements associated with the review and approval, installation, verification, removal, and subsequent testing of temporary circuitry changes and temporary off-normal plant conditions are clearly delineated in station procedures, are understood by the station staff, and are being properly implemented.

3. Licensed Operator Performance

Assure that management provides the necessary opportunities and time so that operators are adequately trained to carry out their assigned responsibilities. In particular, confirm that shift crew members are provided special training for safety related activities that are infrequent, complex, or have unusual safety significance.

If you have any questions concerning this request, please contact this office.

Attachment:

Vermont Yankee AO No. 73-31 - Letter dated November 14, 1973 to the Directorate of Licensing, USAEC, Washington, D.C.

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.

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-gamma 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/WFC/kbd

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MEMO ROUTE SLIP Form AEC-93 (Rev. May 14, 1947) AECM 0240		See me about this. Note and return.		For concurrence. For signature.	 X	For action. For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
RO Chief, FS&EB RO:HQ (5) RO Files Central Mail & Files Regulatory Standards (3)	INITIALS DATE	NIAGARA MOHAWK NINE MILE POINT NUCLEAR POWER STATION DN 50-220 AO 73-11-17
Directorate of Licensing (13) Regional Directors, (RO II, III, IV) OGC	INITIALS DATE	The above abnormal occurrence is forwarded for information. Distribution will be made by this office to the PDR, Local PDR, NSIC, DTIE, and State Representatives.
	INITIALS DATE	

FROM (Name and unit)	REMARKS
E. J. Brunner E. J. Brunner Region I	
PHONE NO. 8-337-1246 — DATE 11/28/73	

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-459

TO: MR. JAMES P. O'REILLY - USAEC - KING OF PRUSSIA, PA. RO:1

FROM: T. J. PERKINS - ST[redacted]N SUPERINTENDENT - NINE MILE [redacted]IT #1

ABNORMAL OCCURRENCE # 73-11-17

LOSS OF OFF SITE POWER

PURSUANT TO TECHNICAL SPECIFICATIONS 1.13 d AND 3.6.3, WE ARE REPORTING AN ABNORMAL OCCURRENCE WHICH WAS TELEPHONED TO YOUR OFFICE ON 11/18/73 AT APPROXIMATELY 1030 HRS.

ON NOVEMBER 17, 1973 AT 1626 HRS., NINE MILE POINT NUCLEAR STATION, UNIT #1 EXPERIENCED A LOSS OF ALL OFF SITE POWER.

FOLLOWING OPERATOR CRITICAL DEMONSTRATIONS, THE REACTOR WAS IN THE JUST CRITICAL CONDITION, MODE SWITCH IN START-UP, MODERATOR TEMPERATURE 207°F, WITH ONE 115 KV RESERVE LINE DE-ENERGIZED TO CONNECT THE 115 KV LINE TO JAFNPP. IN THE COURSE OF THE RELAY WORK ASSOCIATED WITH THIS CONNECTION, A PLANT ELECTRICIAN BUMPED 50 FDS/SI, A SEAL IN RELAY, WHICH TRIPPED AND DE-ENERGIZED RELAY 94C CAUSING THE LOSS OF THE REMAINING LINE.

A REACTOR SCRAM OCCURRED 5.4 SECONDS LATER AS MG SETS #131 AND #141 UPON LOSS OF POWER WERE COASTING DOWN AND TRIPPED AT 55 HZ. WITHIN 9 SECONDS FROM THE LOSS OF ALL A.C., THE TWO DIESEL GENERATOR SYSTEMS WERE UP AND HAD ENERGIZED THE TWO CONTROL ROD DRIVE PUMPS TO MAINTAIN LEVEL. REACTOR WATER LEVEL NEVER DECREASED AND WAS MAINTAINED WITHIN THE NORMAL EXPECTED RANGE FOR THIS TRANSIENT. ALL SYSTEMS FUNCTIONED AS DESIGNED WITH A LOSS OF 115 KV WITHOUT THE TURBINE-GENERATOR UNIT AVAILABLE.

THE 115 KV LINE WAS AUTOMATICALLY RESTORED TO SERVICE IN APPROXIMATELY 10 SECONDS. THE REACTOR WAS RESTARTED AT APPROXIMATELY 9 PM AFTER RESTORATION OF BOTH 115 KV LINES AND NORMAL CONDITIONS ESTABLISHED WITHIN THE PLANT.

THERE WAS AT NO TIME A HAZARD PRESENTED TO THE GENERAL PUBLIC AS A RESULT OF THIS TRANSIENT.

THOMAS J. PERKINS
STATION SUPERINTENDENT
NINE MILE POINT NUCLEAR STATION

MEMO ROUTE SLIP Form AEC-93 (Rev. May 14, 1947) AECM 0240		[ ] See me about this. [ ] Note and return.	[ ] For concurrence. [ ] For signature.	[ ] For action. [X] For information.
TO (Name and unit) RO Chief, FS&EB RO:HQ (5) RO Files Central Mail & Files Regulatory Standards	INITIALS DATE (3)	REMARKS NINE MILE POINT NUCLEAR POWER STATION DN 50-220 AO 73-11-20		
TO (Name and unit) Directorate of Licensing (13) Regional Directors, (RO II, III, IV) OGC	INITIALS DATE	REMARKS The above abnormal occurrence is forwarded for information. Distribution will be made by this office to the PDR, Local PDR, NSIC,		
TO (Name and unit)	INITIALS DATE	REMARKS DTIE, and State representatives.		
FROM (Name and unit) E. J. Brunner Region I E. J. Brunner	REMARKS			
PHONE NO. 8-337-1246	DATE 11/28/73			

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-469

Branch File

TO: J. O'REILLY

RO: 1

PHOTOCOPY

ABNORMAL OCCURRENCE 73-11-20

FAILURE TO FULLY INSERT CONTROL RODS ON A SCRAM

ON NOVEMBER 20, 1973, AT APPROXIMATELY 0230 HRS. A REACTOR RECIRCULATION FLOW TRANSIENT OCCURRED RESULTING IN A HIGH FLUX SCRAM. THE REACTOR WAS OPERATING AT 77% POWER AT THIS TIME. UPON OPERATOR VERIFICATION OF THIS SCRAM 11 RODS WERE FOUND TO BE POSITIONED AT THE 02 NOTCH POSITION. THE OPERATOR IMMEDIATELY INSERTED THESE RODS TO 00. SUBSEQUENT TESTING (STALL LEAKAGE FLOW MEASUREMENTS AND SCRAM TIMING) OF THESE RODS SHOWED EXCESSIVE FLOW PAST THE STOP PISTON SEALS, WHICH WOULD RESULT IN THE CONTROL ROD DRIVE SLOWING DOWN DURING THE LAST 5% OF ITS TRAVEL. THESE DRIVES HAVE BEEN SCHEDULED FOR OVERHAUL DURING THE NEXT REFUELING OUTAGE, MARCH, 1974. THE OPERATOR VERIFIED THAT NEUTRON INSTRUMENTATION INDICATED A HIGHLY SUBCRITICAL REACTOR EVEN WITH THESE RODS AT 02 NOTCH POSITION, THEREFORE NO HAZARD WAS PRESENTED TO THE GENERAL PUBLIC. THIS EVENT IS DEEMED AN ABNORMAL OCCURRENCE PERSUANT TO TECHNICAL SPECIFICATION 1.13d.

THOMAS J. PERKINS
SUPERINTENDENT
NINE MILE POINT NUCLEAR POWER STATION

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MEMO ROUTE SLIP Form AEC-93 (Rev. May 14, 1947) AECM 0240		See me about this. Note and return.		For concurrence. For signature.	X	For action. For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
RO Chief FS&EB RO:HQ (5) RO Files OGC Central Mail & Files	INITIALS DATE ✓	NINE MILE POINT NUCLEAR STATION AO 73-11-13
Regulatory Standards Directorate of Licensing (13) Regional Directors (RO II, III, IV)	INITIALS (3) DATE	The above abnormal occurrence is forwarded for information. Distribution will be made by this office to the PDR, Local PDR, NSIC,
	INITIALS DATE	DTIE, and state representatives.

FROM (Name and unit)	REMARKS
E. J. Brunner E. J. Brunner Region I	
PHONE NO. 8/337/1246 — DATE 11/16/73	

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-459

A/D

FROM: T. J. PERKINS - NINE MILE POINT NUCLEAR STATION

ABNORMAL OCCURRENCE 73-11-13

LOSS OF ONE RESERVE POWER TRANSFORMER

IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS 1.13 d AND SPECIFICATION 3.6.3 d -

SPECIFICATION:

d. FOR ALL REACTOR OPERATING CONDITIONS EXCEPT COLD SHUTDOWN, THERE SHALL NORMALLY BE AVAILABLE TWO 115 KV EXTERNAL LINES, TWO DIESEL-GENERATOR POWER SYSTEMS AND TWO BATTERY SYSTEMS ...

A GROUND OCCURRED ON THE 4 KV SECONDARY OF 101S RESERVE POWER TRANSFORMER REQUIRING TRANSFER OF LOAD TO 101N AND THE REMOVAL OF 101S TRANSFORMER FROM SERVICE AT 1200 HOURS ON 11/13/73. BOTH 115 KV LINES WERE OPERABLE AS WERE THE TWO DIESEL GENERATOR SYSTEMS. HOWEVER, THE RESERVE TRANSFORMER FORMS AN INTEGRAL PART OF THE 115 KV RESERVE POWER SYSTEM FOR NINE MILE POINT UNIT 1 AND THUS ONE OF THESE TWO SYSTEMS WAS INOPERABLE WITH THE REMOVAL OF 101S TRANSFORMER. SUBSEQUENT INVESTIGATION SHOWED A 600 VOLT WIRE SUPPLYING THE DUCT HEATER, ACROSS ONE PHASE OF THE SECONDARY OF THE TRANSFORMER. THE WIRE WAS REMOVED AND THE TRANSFORMER PLACED BACK IN SERVICE AT 1713 HOURS 11/13/73.

BASED UPON THE FACT THAT TWO DIESEL GENERATORS WERE AVAILABLE AS WELL AS ONE 115 KV RESERVE POWER SYSTEM NO DIFFICULTIES WOULD HAVE BEEN ENCOUNTERED HAD THESE SYSTEMS BEEN REQUIRED FOR OPERATION. THEREFORE, NO HAZARD WAS PRESENTED TO THE GENERAL PUBLIC DURING THE TIME 101S TRANSFORMER WAS REMOVED FROM SERVICE.

T. J. PERKINS
STATION SUPERINTENDENT
NINE MILE POINT NUCLEAR STATION

Rec'd by Facsimile 11/15/73 — 0945 a.m.

MEMO ROUTE SLIP Form AEC-93 (Rev. May 14, 1947) AECM 0240		
See me about this. Note and return.	For concurrence. For signature.	For action. For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
RO:Chief FS&EB RO:HQ (5) RO Files Central Mail & Files	INITIALS: DATE: ✓	AO-73-11-10 NIAGARA MOHAWK - NINE MILE POINT DN-50-220
Regulatory Standards Directorate of Licensing (13) Regional Directors (RO II, III, IV)	INITIALS: (3) DATE:	The above AO is forwarded for information. Distribution will be made by this office to the PDR, Local PDR, NSIC, DTIE, and State
	INITIALS: DATE:	Representatives.

FROM (Name and unit)		REMARKS
E. J. Brunner E. J. Brunner RO:I		
PHONE NO. 8-337-1246	DATE 11/13/73	

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-469

Abnormal Occurrence

[illegible]

TELECOPY TO: [redacted] Mr. James O'Reilly - AEC [redacted] G OF PRUSSIA, PA.

ABNORMAL OCCURRENCE 73-11-10 FAILURE OF ONE LOW LOW LEVEL SENSOR

DURING ROUTINE SURVEILLANCE TESTING ON 11-10-73 AT 1509 HOURS, ONE OF TWO INSTRUMENT CHANNELS WAS FOUND TO ACTUATE ABOVE NORMAL SET POINT, BUT WITHIN THE ALLOWABLE RANGE OF TABLE 3.6.2 OF TECHNICAL SPECIFICATIONS. HOWEVER, FOLLOWING A REVIEW ON 10/12/73 - MONDAY - OF THE SURVEILLANCE TESTING, IT WAS NOTED THAT IN THE BASES OF THE TECHNICAL SPECIFICATION, THE INSTRUMENT WAS BEYOND ITS NORMAL DEVIATION. THEREFORE, AS DEFINED IN TECHNICAL SPECIFICATIONS 1.13d, WE ARE REPORTING AS AN ABNORMAL OCCURRENCE THE FAILURE OF ONE LOW LOW LEVEL SENSOR. TECHNICAL SPECIFICATION 3.6.2 (PROTECTIVE INSTRUMENTATION) STATES: "THE SET POINTS, MINIMUM NUMBER OF TRIP SYSTEMS, AND MINIMUM NUMBER OF INSTRUMENT CHANNELS THAT MUST BE OPERABLE FOR EACH POSITION OF REACTOR MODE SWITCH SHALL BE AS GIVEN IN TABLES 3.6.2a TO 3.6.2j.

PARAMETER	MINIMUM NO. OF TRIPPED OR OPERABLE TRIP SYSTEMS	MINIMUM NO. OF OPERABLE INSTRUMENT CHANNELS PER OPERABLE TRIP SYSTEM	SET POINT	*DEVIATION
LOW LOW REACTOR WATER LEVEL	2	2	≤5 ft. below minimum normal water level	±2.6"

THE REACTOR LOW LOW WATER LEVEL SENSORS (2 EACH PER TRIP CHANNEL) WERE ROUTINELY TESTED. RE02D WAS FOUND TO ACTUATE AT 4' 9" BELOW MINIMUM NORMAL WATER LEVEL. NORMAL ACTUATION IS 5' 0" BELOW MINIMUM NORMAL ±2.6".

SYSTEM	REQUIRED LOGIC	REMARKS
PRIMARY COOLENT ISOLATION CONTAINMENT ISOLATION EMERGENCY COOLING INITIATION START CORE SPRAY PUMPS	2 SENSORS PER TRIP CHANNEL 1 SENSOR PER TRIP CHANNEL TO INITIATE	WOULD CAUSE REQUIRED FUNCTION AT 5 FT BELOW MIN. NORMAL LEVEL
INITIATES CONTAINMENT SPRAY	2 SENSORS PER CHANNEL (ONE SENSOR PER TRIP CHANNEL TO PARTIALLY INITIATE) ALSO NEED HIGH DRYWELL PRESSURE	WOULD CAUSE REQUIRED FUNCTION AT 5 FT. BELOW MIN. NORMAL LEVEL AND 3.5 PSI

ALL THE SYSTEMS LISTED ABOVE WOULD FUNCTION AT THE REQUIRED SET POINT. THE CHANNEL 12 TRIP WOULD OCCUR AT 4' 9" BELOW MINIMUM NORMAL AND THE CHANNEL 11 TRIP WOULD OCCUR AT 5' BELOW MINIMUM NORMAL (ACTUATING THE PROTECTIVE FUCTION). BOTH TRIP CHANNELS MUST FUNCTION TO OBTAIN THE PROTECTIVE FUNCTION AND EVEN IF THE 12 TRIP CHANNEL FUNCTION OCCURRED FIRST, THE SYSTEM PROTECTION FUNCTION WOULD NOT OCCUR

UNTIL THE 11 TRIP SYSTEM ACTUATED. TESTING OF ALL SENSORS IN THIS LOGIC SHOWED NORMAL PROTECTIVE FUNCTION WOULD BE INITIATED AT THE DESIRED LEVEL (5' BELOW MINIMUM NORMAL). THE INSTRUMENT, RE02D, HAS RECALIBRATED AND VERIFIED TO OPERATE AT THE 5' BELOW MINIMUM NORMAL SET POINT.

BASED ON THE ABOVE, NO UNDUE SAFETY HAZARD HAS PRESENTED TO THE GENERAL PUBLIC NOR WOULD ANY SYSTEM REQUIRED TO FUNCTION FAIL TO FUNCTION PROPERLY AT ITS REQUIRED SET POINT.

* THE DEVIATIONS ARE FOUND IN THE BASES OF THE TECHNICAL SPECIFICATION 3.6.2 AND ARE NOT PART OF THE TABLE.

P. ALLISTER BURT
GENERAL SUPERINTENDENT
NUCLEAR GENERATION
NINE MILE POINT NUCLEAR STATION

TO (Name and unit)	INITIALS / DATE	REMARKS
H. D. Thornburg, Chief, FS&EB	INITIALS: DATE:	Licensee: Niagara Mohawk Power Corp. Docket No.: 50-220 Abnormal Occurrence: Facsimile Message received 8/30/73 - Core power distribution fuel bundle segments found operating in excess of T.S.
RO:HQ (5) DR Central Files (1) Regulatory Standards (3) Dir. of Licensing (13)	INITIALS: DATE:	The attached report from the subject licensee is forwarded in accordance with RO Manual Chapter 1000.
RO Files Central Mail & Files	INITIALS: DATE:	The action taken by the licensee is considered appropriate. Followup will be performed during the next inspection as appropriate. Copies of
FROM (Name and unit) R. T. Carlson R. T. Carlson, Chief Facility Operations Branch		REMARKS the report have been forwarded to the PDR, Local PDR, NSIC, DTIE and State representatives. The licensee will submit a 10 day written report to Licensing.
PHONE NO.	DATE 8/30/73	

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-4[illegible]

Abn Occ

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TO: MR. JAMES P. O'REILLY

FROM: MR. THOMAS J. PERKINS

NINE MILE POINT NUCLEAR STATION
DOCKET NO. 50-220

On or about 1600 hours, August 29, 1973 following a routine calibration of the Local Power Range Monitors (LPRM) and subsequent core power distribution calculation fuel bundle segments were found to be operating slightly in excess of the allowable average planar LHGR as shown in Figure 3.1.7 The Fuel Bundles in question were reload 1 type. The maximum average planar LHGR was found to be approximately 10.75 KW/FT. For this fuel type and nodal exposure of approximately 1930 MWD/ST the limit shown on Figure 3.1.7 is approximately 10.6 KW/FT. Immediately steps were taken to reduce the maximum average planar LHGHR by reducing Core Thermal Power using Reactor Recirculation Flow. Subsequent core power distributions calculations showed all fuel types to be operating within the limits shown on Figure 3.1.7.

In order to maintain the Core Power Distribution input data (LPRM) reliability at a high level the frequency of LPRM calibrations will be increased from the nominal full power month frequency to every full power two week period.

In addition, steps will be taken to alter the existing rod pattern to acheive a flatter power distribution which will improve the margin between operating levels and limits.

50-220

MEMO ROUTE SLIP
Form AEC-93 (Rev. May 14, 1947) AECM 0240

- See me about this.
- Note and return.
- For concurrence.
- For signature.
- For action.
- For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
H. D. Thornburg, Chief, FS&EB, RO	INITIALS: DATE:	BLUE SHEET - NIAGARA MOHAWK POWER CORP. (NMP-1) The attached blue sheet is forwarded for record purposes only, with the knowledge that
cc: RO Files ~~DR~~ Central Files	INITIALS: DATE:	a decision was made by RO:HQ not to issue it.
	INITIALS: DATE:	

FROM (Name and unit)
R. T. Carlson
R. T. Carlson, RO:I

REMARKS:

PHONE NO.:

DATE: 6/14/73

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-469

To: H. D. Thornburg, Chief, FS&EB, RO
From: R. T. Carlson, RO:I
DRAFT 6/12/73

DIRECTORATE OF REGULATORY OPERATIONS
NOTIFICATION OF AN INCIDENT OR OCCURRENCE

Facility: Niagara Mohawk Power Corporation (Nine Mile Point 1)

Problem:

RO Region I (Newark) was informed by the licensee by telephone on June 12, 1973, that during planned surveillance testing of the pressure vessel electromatic relief valves at 12% of rated power, one of the valves failed to reseat. The main steam bypass valves were open at the time, thus enabling reactor pressure to be maintained relatively constant at 950 psig throughout the 43 minute period of steam release to the torus, which was terminated by closure of a manual block valve. The following preliminary information was provided by the licensee:

1. Time of occurrence - 8:08 a.m., 6/12/73.

2. 12,200 gallons of water were dumped to the torus during the incident.

3. Water level in the reactor pressure vessel was normal throughout the occurrence.

4. No safety equipment was called upon to function.

5. No radioactivity was released to the environment.

Action:

1. The relief valve is being disassembled to investigate the cause of failure.

2. The licensee will submit 24 hour and 10 day written reports as required by the Technical Specifications.

3. The Region I (Newark) office is following closely the licensee's investigation. Further action by RO, including generic considerations, will be based on developments.

4. The Northeast office of the Office of Information Services and the State of N. Y. have been informed of this occurrence.

5. The Technical Assistants to the Commissioners and the Staff of the JCAE are being informed by copy of this notification.

(Further information on individuals to contact and distribution are to be prepared by RO:HQ.)

Prepared by: T. Young, Jr.

MEMO ROUTE SLIP Form AEC-93 (Rev. May 14, 1947) AECM 0240		See me about this. Note and return.	For concurrence. For signature.	For action. For information.
TO (Name and unit) H. D. Thornburg, Chief, FS&EB	INITIALS DATE	REMARKS Licensee: Niagara Mohawk Power Corporation Docket No.: 50-220 Abnormal Occurrence: TWX dated 6/12/73		
TO (Name and unit) RO:HQ (5) ~~DR~~ Central Files (1) Regulatory Standards (3) Dir. of Licensing (13)	INITIALS DATE	REMARKS The attached report from the subject licensee is forwarded in accordance with RO Manual Chapter 1000		
TO (Name and unit) RO Files	INITIALS DATE	REMARKS The action taken by the licensee is considered appropriate. Followup will be performed during the next inspection as appropriate. Copies of		
FROM (Name and unit) R T Carlson R. T. Carlson, RO:I	REMARKS the report have been forwarded to the PDR, Local PDR, NSIC, DTIE and State representatives. The licensee will submit a 10 day written report to Licensing.			
PHONE NO. / DATE 6/14/73				

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445

Western Union Telegram

NKA484(1842)(2-096543E163)PD 06/12/73 1841
ICS IPMMTZZ CSP
3153422049 TDMT OSWEGO NY 194 06-12 0641P EST
PMS KEE YOUNG, TLX
CARE ATOMIC ENERGY COMMISSION DIVISION OF REACTIVE COMPLIANCE
REGION 1 970 BROAD ST
NEWARK NJ

REACTOR OPERATING AT 12 PERCENT POWER WITH 30 PERCENT BYPASSED STEAM (100 PERCENT BYPASS EQUALS 40 PERCENT TOTAL STEAM). REACTOR PSIG 950, TORUS TEMPERATURE 75 DEGREES F TORUS LEVEL 3.2 FEET SUBMERGENCE. TESTING OF ELECTROMATIC RELIEF VALVES IN PROGRESS. ALL VALVES HAD RESPONDED SATISFACTORILY UP TO THIS TIME. OPENED FIFTH VALVE OF SIX (113) AT 0808 HOURS. INDICATED BYPASS STEAM DROPPED TO 16 PERCENT. ELECTROMATIC VALVE SWITCH THEN CLOSED (DURATION OF CYCLE 1.32 SECONDS). BYPASS INDICATION REMAINED AT 16 PERCENT. TORUS TEMPERATURE STARTED SLIGHT INCREASE TERMINATION

SF-1201 (R5-69)

Western Union Telegram

OF ELECTROMATIC OPERATION BY CLOSING MANUAL BLOCKING VALVE UPSTREAM OF 113 ELECTROMATIC. BYPASS VALVE POSITION RETURNED TO 27 PERCENT (REACTOR SHUT DOWN HAD BEGUN DURING ELECTROMATIC VALVE 113 OPERATION, THEREFORE DIFFERENCE IN STARTING AND ENDING BYPASS POSITION) TORUS TEMPERATURE HEAT AT 125 DEGREES F, LEVEL AT 3.6 FEET SUBMERGENC E. REACTOR PERAMETERS UNCHANGED DURING INCIDENT. CALCULATED MAXIMUM BLOWDOWN OF COOLANT 25,000 GALLONS, APPROXIMATELY 207 K POUNDS OF STEAM. DURATION OF ELECTROMATIC OPERATION 43 MINUTES. REACTOR ORDERLY SHUT DOWN COMPLETED TO LESS THAN 110 PSIG AT 1300 HOURS. TORUS COOLANT ACTIVITY INCREASED FROM 470 C/M/ML TO 586 C/M/ML DURING INCIDENT. WRITTEN REPORT TO FOLLOW BY JUNE 22 1973

T J PERKINS 9 MILE POINT NUCLEAR POWER STATION