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.

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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 adminisstrative 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

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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.

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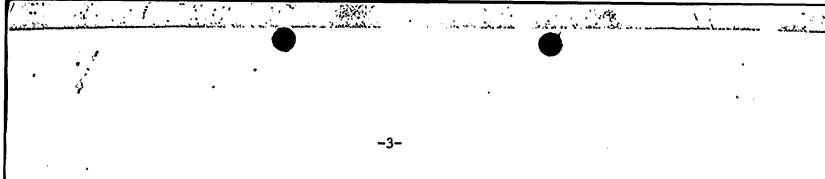
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If you have any questions concerning this request, please contact this office.

Attachment:

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

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VERMONT YANKEE NUCLEAR POWER CORPORATION

SEVENTY SEVEN OROVE STREET

RUTLAND, VERMONT 05701

VYV-3071

REPLY TO: P. O. DOX 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. A0-73-31

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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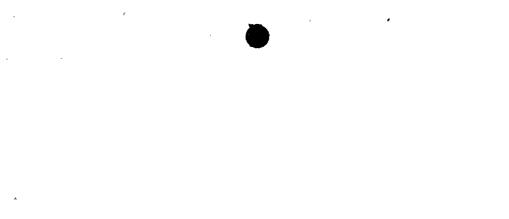
VERMONT YANKEE NUCLEAR POWER CORPORATION

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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.



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VERMONT YANKEE NUCLEAR POWER CORPORATION

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Directorate of Licensing November 14, 1973 Page 3

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	r	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% AK, and had rod 26-23 been fully withdrawn, the excess reactivity would have been 0.97% AK.

* These assemblies were visually inspected.

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VERMONT YANKEE NUCLEAR POWER CORPORATION

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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.

1. 6. 5.

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 1131 and 1132 sensitivity. The positive results obtained verify the adequacy of sipping sensitivities observed on November 8, 1973.

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VERMONT YANKEE NUCLEAR POWER CORPORATION

Directorate of Licensing November 14, 1973 Page 5

- 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.

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VERMONT YANKEE NUCLEAR POWER CORPORATICY

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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.

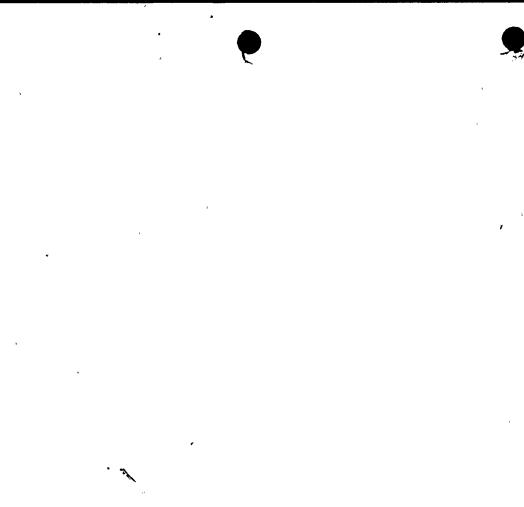
Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

B.W. Riley

Plant Superintendent

BWR/WFC/kbd



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TO: MR. JAMES P. O'REILLY - USAEC - KING OF PRUSSIA, PA. RO:I FROM: T. J. PERKINS - STATION SUPERINTENDENT - NINE MILE

ABRÉRHAL OCCURRENCE # 73-11-17

LOSS OF OFF SITE POWER

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

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

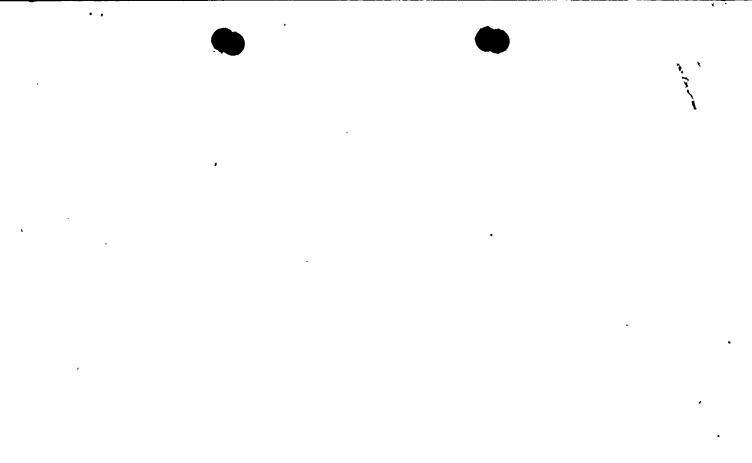
FOLLOWING OPERATOR CRITICAL DEMONSTRATIONS, THE REACTOR WAS IN THE JUST CRITICAL CONDITION, MODE SWITCH IN START-UP, MODERATOR TEMPERATURE 207°F, WITH ONE 115 KV RESERVE LINE DE-EMERGIZED TO CONNECT THE 115 KV LINE TO JAFNPP. IN THE COURSE OF THE RELAY WORK ASSOCIATED WITH THIS CONNECTION, A PLANT ELECTRICIAN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

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

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

THERE WAS AT IN THE A WALARD PRESERVED TO THE CONCLUCT FUELIC AS A SPECIFIC FUELS.

THOMAS J. PERKINS STATION SUPERINTENDENT NINE MILE POINT NUCLEAR 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.
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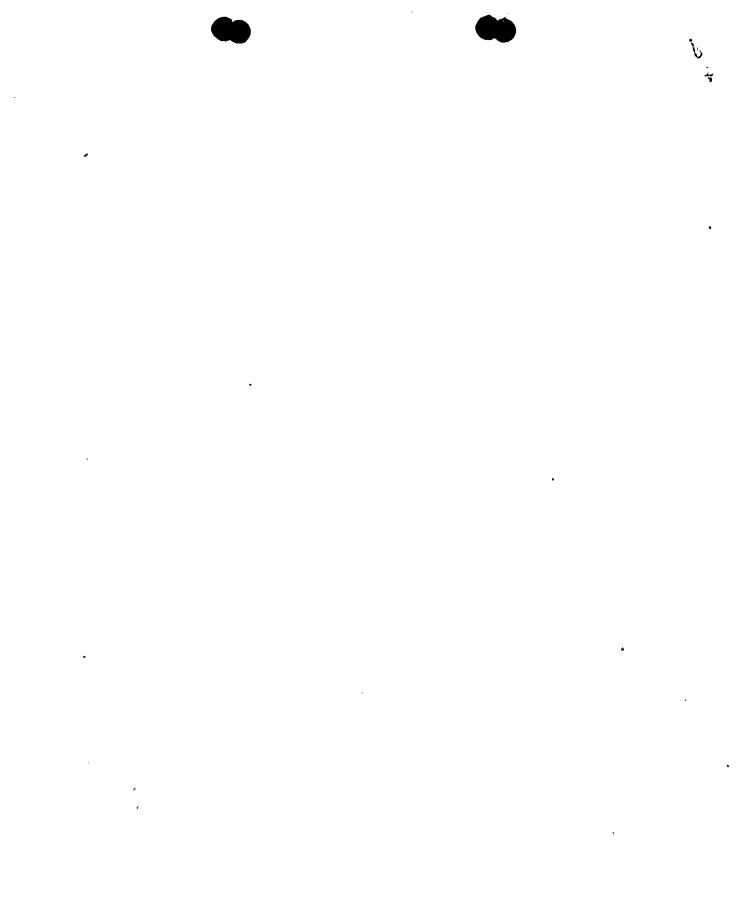
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TO: J. O'REILLY

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RO: 1

ABNORMAL OCCURPENCE 73-11-20

FALLURE TO PULLY INSERT CONTROL RODS ON A SCRAM

ON NOVENEER 20, 1973, AT APPROXIMATELY 0230 HRS. A REACTOR REGIRGULATION PLOW TRANSTENT OCCURRED RESULTING IN A HIGH FLUX SORAM. 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 INDEDIATELY INSERTED THESE RODS TO 00. SUBSEQUENT THETING (STALL LEAKAGE FLOW MEASUREMENTS AND SCRAM TIMING) OF THESE RODS SHOWED EXCERSIVE FLOW PAST THE STOP FISTON SEALS, WHICH WOULD RESULT IN THE CON-TROL ROD DRIVE SLOWING DOWN DURING THE LAST 5% OF FTS TRAVEL. THESE DRIVES HAVE BEEN SCHEDULED FOR OVERHAUL DURING THE NEXT REFUELING OUTAGE, MARCH, 1974: THE OPERATOR VERIFIED THAT NEUTRON INSTRUGENTATION INDICATED A HIGHLY SUBCRITICAL REACTOR EVEN WITH THESE RODS AT 02 NOTCH POSITION, THEREFORE NO HAZARD WAS PRESENTED TO THE GENERAL FUELC. THIS EVENT IS DEEMED AN ABNORMAL OCCURRENCE PERSUANT TO TECHNICAL SPECIFICATION 1.134.

> THOMAS J. PERKINS SUPERINTENDENT NINE MILE POINT NUCLEAR FOWER STATICS

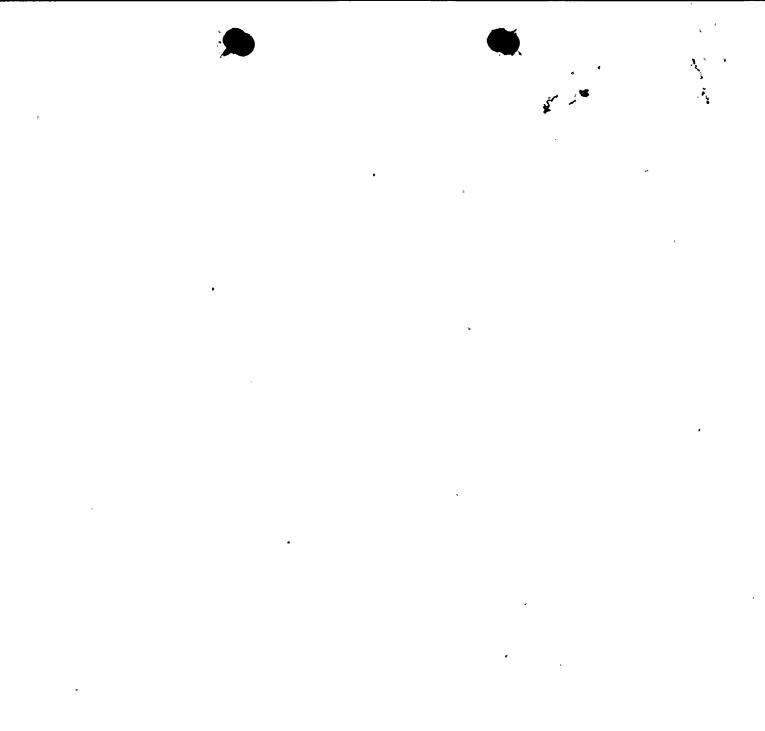
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FROM: T. J. PERIGHS - REPERILE POINT NUCLEAR STATION

ABRORNAL OCCURRENCE 73-11-13

LOSS OF OHE RESERVE POHER TRANSFORMER

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

SPECIFICATION:

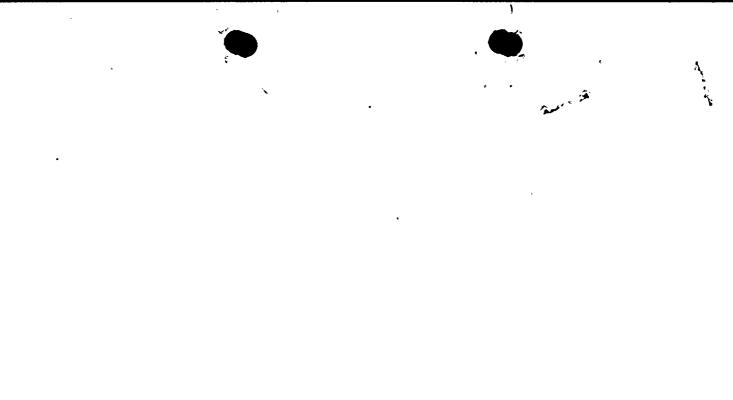
4. FOR ALL REACTOR OPERATING CONDITIONS EXCEPT COLD SHUTDOWN, THERE SHALL NORMALLY BE AVAILABLE THE 115 KY EXTERNAL LINES, THE DIESEL-GENERATOR POWER SYSTEMS AND TWO BATTERY SYSTEMS ...

A GROUND OCCURNED 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 HERE OPERABLE AS HERE THO DIESEL GEN-ENATOR SYSTEMS. HOWEVER, THE RESERVE TRANSFORMER FORMS AN INTEGRAL PART OF THE 116 KV RESERVE POWER SYSTEM FOR NINE MILE POINT UNIT 1 AND THUS ONE OF THESE THO 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 HOULD 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 HINE MILE POINT NUCLEAR STATION

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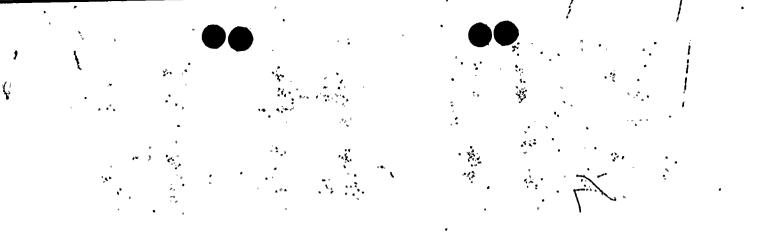


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TULECOPY TO: TO: Hr. James O'Reilly + AEC G G OF PRUSSIA, PA.

ABNORMAL OCCURRENCE 73-11-10 FAILURE OF ONE 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. THERE-FORE, AST 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 HODE SWITCH SHALL BE AS GIVEN IN TABLES 3.6.2a TO 3.6.21.

PARAMETER	MINIHUM NO. OF TRIPPE OR OPERABLE TRIP SYSTEMS	D MINIMUM NO. OF OPERABLE INSTRUMENT CHANNELS PER OPERABLE TRIP SYSTEM	
LOW LOW REACTOR WATER LEVEL	~ · Ż ·	·2	<5 ft. below <u>+2</u> .6" Minimum normal water level
THE REACTOR LON LO	W WATER LEVEL SENSORS (2 EACH PER TRIP CHANNEL)	WERE ROUTINELY

TESTED. REDZD WAS FOUND TO ACTUATE AT 4' 9" BELOW MINIMUM NORMAL WATER LEVEL. NORMAL ACTUATION IS 5' O" BELOW MINIMUM NORMAL +2.6".

System	REQUIRED LOGIC	REMARKS
PRIMARY COOLENT ISOLATION CONTAINMENT ISOLATION EMERGENCY COOLING 121TIATION START CORE SPRAY PUMPS	2 SENSORS PER TRIP CHANNE 1 SENSOR PER TRIP CHANNE TO INITIATE	EL WOULD CAUSE L REQUIRED FUNCTION AT 5 FT BELON MIN. NORMAL LEVEL
INITIATES CONTAINMENT SPRAY	2 SENSORS PER CHANNEL (ONE SENSOR PER TRIP CHANNEL TO PARTIALLY INITIATE) ALSO NEED HIG URYWELL PRESSURE	HOULD CAUSE REQUIRED FUNCTION AT H 5 FT. BELOW MIN. NORMAL

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TELECOPY TO J. D'REILLY - AEL - 11/12/13

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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 HUSY FUNCTION TO OBTAIN THE PROTECTIVE FUNCTION AND EVEN IF THE 12 TRIP CHANNEL FUNCTION OCCURRED FIRST, THE SYSTEM PROTECTION FUNCTION WOULD NOT OCCUR

JUNTIL THE II TRIP SYSTEM ACTUATED. TESTING OF ALL SENSORS IN THIS LOGIC SHOWED NORMAL PROTECTIVE FUNCTION HOULD BE INITIATED AT THE DESIRED LEVEL (5' BELOW MINIMUM NORMAL). THE INSTRUMENT, REO2D, WAS RECALIBRATED AND VERIFIED TO OPERATE AT THE 5' BELOW MINIMUM NORMAL SET POINT.

BASED ON THE ABOVE, NO UNDUE SAFETY HAZARD WAS 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 HILE POINT NUCLEAR STATION

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	TO (Name and unit)	INITIALS .	REMARKS					
·	AF	· · · ·		Licensee: Niagara Mohawk Power Corp.					
. .	H. D. Thornburg, Chief, FS&EB TO (Name and unit) RO:HQ (5)		DATE	Docket No.: 50-220					
;				Abnormal Occurrence: Facsimile Message received					
			INITIALS REMARKS 8/30/73 - Core power distri- bution fuel bundle segments for						
-									
	DR Central	DR Central Files (1)		DATE The attached report from the subject licensee is					
	Regulatory Standards Dir. of Licensing (13) TO (Name and unit) RO Files Central Mail & Files		3) forwarded in accordance with RO Manual Chapter 1000.						
			INITIALS REMARKS The action taken by the licensee is considered						
	•		appropriate. Followup will be performed during						
	FROM (Name and unit) R. T. Carlson, Chief Facility Operations Branch PHONE NO. DATE			the next inspection as appropriate. Copies of					
			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						
:	PHONE NO.	date 8/30/7 <u>3</u>	•	Licensing.					
•	•••••	· · ·	;	USE OTHER SIDE FOR ADDITIONAL REMARKS GPO : 1971 0 - 413-					
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TO: NR. JAMES P. O'REILLY FROM: NR. 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 (LPRN) and subsequent core power distribution calculation fuel bundle sugments were found to be operating slightly in excess of the allowable avorage planar LHGR as shown in Figure 3.1.7 The Fuel Bundles im question were reload 1 type. The maximum average planar LHGR was found to be approximately 10.75 KM/FT. For this fuel type and nodal exposure of approximately 1930 MMD/ST the limit shown on Figure 3.1.7 is approximately 10.6 KM/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) relizbility at a high level the frequency of LPRM calibrations will be increased from the nominal full power month frequency to every full power two weak 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.

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MEMO ROUTE SL		See me about this.	Foi concurrence		action.		
Form AEC-93 (Rev. May 14, 1947) AECM 0240 INITIALS	Note and return. REMARKS	For signature.	For	Information,		
v (namy and unit)		BLUE SHEET - R	IAGARA MOHAWK	POWER CORP.	(NMP-1)		
H. D. Thornburg, Chie	ŧ						
FS&EB, RO	DATE	The attached l	The attached blue sheet is forwarded for				
•	}	record purpose	s only with	the knowledg	e that		
fO (Name and unit)	INITIALS .	REMARKS	Cheven J. J. M. H. M.				
		a decision was	s made by RO:H	<u>Q not to iss</u>	ue it.		
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6/14/73							
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To: H. D. Thornburg, Chief, FS&EB, RO From: R.T. Carlson, RO:I DRAFT 6/12/73

DIRECTORATE OF REGULATORY OPERATIONS NOTIFICATION OF AM INCIDENT OR OCCURRENCE

Facility: <u>Niagara Mohawk Power Corporation (Nine Mile Point 1)</u>

Problem:

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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.
- 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.)

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ا دمهاهادو تغاويس د MEMO ROUTE SLIP See me about this, For concurrence. For action. Form AFC-93 (Rev. May 14, 1917) AFCM 0240 Note and return, For signature. For Information, INITIALS . REMARKS TO (Name and unit) Licensee: Niagara Mohawk Power Corporation H. D. Thornburg, DATE Docket No.: 50-220 Chief, FS&EB Abnormal Occurrence: TWX dated 6/12/73 INITIALS REMARKS TO (Hame and unit) RO:HQ (5) The attached report from the subject licensee is DR Central Files (1) DATE Regulatory Standards. (3) forwarded in accordance with RO Manual Chapter 1000 Dir. of Licensing (13 TO (Name and unit) INITIALS REMARKS The action taken by the licensee is considered **RO Files** appropriate. Followup will be performed during DATE the next inspection as appropriate. Copies of FROM (Name and unit) REMARKS the report have been forwarded to the PDR, Local PDR, NSIC, DTIE and State representatives. The R. T. Carlson, RO:I licensee will submit a 10 day written report to PHONE HO. DATE Licensing. 6/14/73 USE OTHER SIDE FOR ADDITIONAL REMARKS GPO : 1971 O

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T J PERKINS 9 MILE POINT NUCLEAR POWER STATION

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