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Docket No. 50-271

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Mr. Robert L. Smith
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Dear Mr. Smith:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment revises the Technical Specifications in response to your applications of March 17, 1980 and August 28, 1980, as supplemented by your letters of May 9, August 13, September 23 and October 14, 1980.

The revised Technical Specifications authorize replacement of existing pressure switches that sense reactor pressure and water level with analog loops, installation of an ATWS recirculation pump trip, and modification of testing requirements for certain containment isolation valve tests to reflect changes brought about to permit post-accident hydrogen sampling capability.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 58
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

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10/21/80

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10/29/80

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD	Legal review of Amendment and Fed. Reg. notice only
SURNAME	SNorris	VRooney	TAIppolito	TMNovak	R. BACHMANN	
DATE	10/24/80	10/27/80	10/28/80	10/30/80	10/30/80	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 58
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated March 17, 1980 and August 28, 1980, as supplemented May 9, August 13, September 23, and October 14, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-28 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 58 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 58

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Review Appendix A as follows:

1. Remove the pages listed and replace with identically numbered revised pages.

22
24
25
31
40
50
51
52
53
54
55
56
57
63
135
136
174
176
178
179

2. Add pages 34a, 39a, 53a, and 174a

TABLE 4.1.1

SCRAM INSTRUMENTATION AND LOGIC SYSTEMS FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION, LOGIC SYSTEMS AND CONTROL CIRCUITS

Instrument Channel	Group (3)	Functional Test (7)	Minimum Frequency (4)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (5)	Before Each Startup & Weekly during refueling (6)
Inoperative	C	Trip Channel and Alarm	Before Each Startup & Weekly during refueling (6)
APRM			
High Flux	B	Trip Output Relays (5)	Once Each Week
High Flux (Reduced)	B	Trip Output Relays (5)	Before Each Startup & Weekly during refueling (6)
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays (5)	Once Each Week
Flow Bias	B	Trip Output Relays (5)	(1)
High Reactor Pressure	B	Trip Channel and Alarm (5)	(1)
High Drywell Pressure	A	Trip Channel and Alarm	(1)
Low Reactor Water Level (2) (8)	B	Trip Channel and Alarm (5)	(1)
High Water Level in Scram Discharge Volume	A	Trip Channel and Alarm	Every 3 Months
High Main Steamline Radiation (2)	B	Trip Channel and Alarm (5)	Once Each Week
Main Steamline Iso. Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Con. Valve Fast Closure	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)

TABLE 4.1.1 NOTES

1. Initially once per month; thereafter, with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Vermont Yankee.

2. An instrument check shall be performed on reactor water level and reactor pressure instrumentation once per day and on steamline radiation monitors once per shift.

3. A description of the three groups is included in the basis of this Specification.

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the Instrument Functional Test Definition (1.G.). This Instrument Functional Test will consist of injecting a simulated electrical signal into the measurement channels.

6. Frequency need not exceed weekly.

7. A functional test of the logic of each channel is performed as indicated. This coupled with placing the mode switch in shutdown each refueling outage constitutes a logic system functional test of the scram system.

8. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This test will be performed every month after the completion of the monthly tests program.

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group(1)	Calibration Standard(4)	Minimum Frequency(2)
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Output Signal (Reduced)	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
APRM	B(5)	Using TIP System	Every 1000 equiv full pwr hr
High Reactor Pressure	B	Standard Pressure Source	Once/Operating Cycle
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 months
High Drywell Pressure	A	Standard Pressure Source	Every 3 months
High Water Level in Scram Discharge Volume	A	Water Level	Refueling Outage
Low Reactor Water Level	B	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	(6)	Refueling Outage
High Main Steamline Radiation	B	Appropriate Radiation Source (3)	Refueling Outage
First Stage Turbine Pressure Permissive	A	Pressure Source	Every 6 months and after refueling
Maine Steamline Isolation Valve Closure	A	(6)	Refueling Outage

Amendment No. 58

Bases:

4.1 REACTOR PROTECTION SYSTEM

A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B and C. Sensors that make up Group A are of the on-off type and will be tested and calibrated at the indicated intervals. Initially the tests are more frequent than Yankee experience indicates necessary. However, by testing more frequently, the confidence level with this instrumentation will increase and testing will provide data to justify extending the test intervals as experience is accrued.

Group B devices utilize an analog sensor followed by an amplifier and bi-stable trip circuit. This type of equipment incorporates local and/or control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bi-stable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, The IRM is active during start-up and inactive during full-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The peak heat flux and total peaking factor shall be checked once per day to determine if the APRM gains require adjustment. This will normally be done by checking LPRM readings. Because few control rod movements or power changes occur, checking these parameters daily is adequate.

I. RECIRC PUMP TRIP INSTRUMENTATION

During reactor power operation, the Recirc Pump Trip Instrumentation shall be operative in accordance with Table 3.2.1.

I. RECIRC PUMP TRIP INSTRUMENTATION

The Recirc Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.

TABLE 3.2.1

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

Recirculation Pump Trip - A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation are not Satisfied
2	Low-Low Reactor Vessel Water Level	$\geq$ 6' 10.5" above top of active fuel	Note 2
2	High Reactor Pressure	$\geq$ 1150 psig	Note 2
2	Time Delays	$\leq$10 sec.	Note 2
1	Trip System Logic	---	Note 2

TABLE 3.2.1 NOTES

1. Each of the two Core Spray, LPCI and RPT, subsystems are initiated and controlled by a trip system. The subsystem "B" is identical to the subsystem "A".

2. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits. If the channel cannot be tripped by the means stated above, that channel shall be made operable within 24 hours or an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

3. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.

4. One trip system with initiating instrumentation arranged in a one-out-of two logic.

5. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.

6. Any one of the two trip systems will initiate ADS. If the minimum number of operable channels in one trip system is not available, the requirements of Specification 3.5.F.2 and 3.5.F.3 shall apply. If the minimum number of operable channels is not available in both trip systems, Specifications 3.5.F.3 shall apply.

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

TABLE 4.2.1

MINIMUM TEST & CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Core Spray System			
Trip Function	**Functional Test (8)**	**Calibration (8)**	**Instrument Check**
High Drywell Pressure	(Note 1)	every 3 months	---
Low-Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	once each day
Low Reactor Pressure	(Note 1)	every 3 months	---
Pump 14-1A, Discharge Press	(Note 1)	every 3 months	---
Auxiliary Power Monitor	(Note 1)	every refueling	once each day
Pump Bus Power Monitor	(Note 1)	none	once each day
High Sparger Pressure	(Note 1)	every 3 months	---
Trip System Logic except relays 14A-K11A 14A-K11B 14A-K19A 14A-K19B	every 6 months (Note 2)	every 6 months (Note 3)	---

TABLE 4.2.1 (Continued)

Low Pressure Coolant Injection System

Trip Function	Function Test (8)	Calibration (3)	Instrument Check
Low Reactor Pressure #1	(Note 1)	every 3 months	----
High Drywell Pressure #1	(Note 1)	every 3 months	----
Low-Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	once each day
Reactor Vessel Shroud Level	(Note 1)	every 3 months	----
Low Reactor Pressure #2	(Note 1)	every 3 months	----
RHR Pump Discharge Pressure	(Note 1)	every 3 months	----
High Drywell Pressure #2	(Note 1)	every 3 months	----
Low Reactor Pressure #3	(Note 1)	every 3 months	----
Auxiliary Power Monitor	(Note 1)	every refueling outage	once each day
Pump Bus Power Monitor	(Note 1)	None	once each day
LPCI Crosstie Monitor	None	None	once each day
Trip System Logic	Every 6 Months (Note 2)	every 6 months (Note 3)	----

TABLE 4.2.1 (CONT)

High Pressure Coolant Injection System			
Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	once each day
Low Condensate Storage Tank Water Level	(Note 1)	every 3 months	—
High Drywell Pressure	(Note 1)	every 3 months	—
High Suppression Chamber Water Level	(Note 1)	every 3 months	—
Bus Power Monitor	(Note 1)	None	once each day
Trip System logic	every 6 months (Note 2)	every 6 months (Note 3)	—

TABLE 4.2.1 (CONT)

Automatic Depressurization System			
Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	once each day
High Drywell Pressure	(Note 1)	every 3 months	---
Bus Power Monitors	(Note 1)	none	once each day
Trip System Logic (except solenoids of valves)	every 6 months (Note 2)	every 6 months (Note 3)	---

TABLE 4.2.1 (CONT)

Recirculation Pump Trip Actuation System			
Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Low-Low Reactor Vessel Water Level (4)	(Note 1)	Once/Operating Cycle	Once Each Day
High Reactor Pressure (4)	(Note 1)	Once/Operating Cycle	Once Each Day
Trip System Logic	(Note 1)	Once/Operating Cycle	---

TABLE 4.2.2

MINIMUM TEST & CALIBRATION FREQUENCIES

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	once each day
High Steam Line Area Temperature	(Note 1)	each refueling outage	---
High Steam Line Flow	(Note 1)	every 3 months	once each day
Low Main Steam Line Pressure	(Note 1)	every 3 months	---
Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	---
High Main Steam Line Radiation	(Notes 1 & 7)	each refueling outage	once each day
High Drywell Pressure	(Note 1)	every 3 months	---
Condenser Low Vacuum	(Note 1)	every 3 months	---
Trip System Logic except relays 16A-K13 16A-K14 16A-K15 16A-K16 16A-K26 16A-K27	every 6 months (Note 2)	every 6 months (Note 3)	---

TABLE 4.2.2 (CONT'D)

MINIMUM TEST & CALIBRATION FREQUENCIES

HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
High Reactor Water Level	(Note 1)	once/operating cycle	---
High Steam Line Space Temperature	(Note 1)	each refueling outage	---
High Steam Line d/p (Steam Line Break)	(Note 1)	every 3 months	---
Low HPCI Steam Supply Pressure	(Note 1)	every 3 months	---
Main Steam Line Tunnel Temperature	(Note 1)	each refueling outage	---
Bus Power Monitor	(Note 1)	None	once each day
Trip System Logic	every 6 months (Note 2)	every 6 months (Note 3)	---

TABLE 4.2.2 (CONT'D)

MINIMUM TEST & CALIBRATION FREQUENCIES

REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Main Steam Line Tunnel Temperature	(Note 1)	each refueling outage	---
High Steam Line Space Temperature	(Note 1)	each refueling outage	---
High Steam Line d/p (Steam Line Break)	(Note 1)	every 3 months	---
High Reactor Water Level	(Note 1)	once/operating cycle	---
Low RCIC Steam Supply Pressure	(Note 1)	every 3 months	---
Bus Power Monitor	(Note 1)	none	once each day
Trip System Logic	every 6 months (Note 2)	every 6 months (Note 3)	---

TABLE 4.2.3

MINIMUM TEST & CALIBRATION FREQUENCIES

REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT SYSTEM ISOLATION

Trip Function	Functional Test(8)	Calibration (8)	Instrument Check
Low Reactor Vessel Water Level	(Note 1)	once/operating cycle	---
High Drywell Pressure	(Note 1)	every 3 months	---
Reactor Building Vent Exhaust Radiation	Monthly	every 3 months	once each day
Refueling Floor Zone Radiation	Monthly	every 3 months	once each day during refueling
Reactor Building Vent Trip System Logic	every 6 months (Note 2)	every 6 months Note 3)	---
Standby Gas Treatment Trip System Logic	every 6 months (Note 2)	every 6 months (Note 3)	---
Logic Bus Power Monitor	(Note 1)	none	once each day

The low-low reactor water level instrumentation is set to trip when reactor water level is 6'10.5" or -44.5" H_2O indicated on the reactor water level instrumentation above the top of the active fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS, RPT and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation, RPT and primary system isolation so that no melting of the fuel cladding will occur and so that post-accident cooling can be accomplished and the limits of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation, RPT and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum of breaks and meets the above criteria.

The low low low reactor water level instrumentation is set to trip when the reactor water level has reached a level greater than two thirds of the core height. This value was selected as the minimum water level in the reactor vessel, following a design basis accident, that emergency core cooling system water can be diverted from its normal injection path, into the reactor vessel. At two thirds core height level and with high drywell pressure, the appropriate valves may be manually operated to allow primary containment spray operation.

The high drywell pressure instrumentation is a backup to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation; thus the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 120 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10 CFR 100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the limits of 10 CFR 100 are exceeded.

TABLE 4.7.2.a

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time(Sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A, D & 2-86A, D)	4	4	5(note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SC
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9)		1	10	Closed	SC
1	Drywell Air Purge Inlet (16-19-8)		1	10	Open	GC
1	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Open	GC
3	Exhaust to Standby Gas Treatment System (16-19-6)		1	10	Open	GC
3	Containment Purge Supply (16-19-23)		1	10	Open	GC
3	Containment Purge Makeup (16-20-20, 16-20-22A, 16-20-22b)		3	NA	Closed	SC
5	Reactor Cleanup System (12-15, 12-18)	1	1	25	Open	GC
5	Reactor Cleanup System (12-68)		1	45	Open	GC
6	HPCI (23-15, 23-16	1	1	55	Open	GC
6	RCIC (13-15, 13-16)	1	1	20	Open	GC
	Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B)		2	NA	Closed	SC
	Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B)		2	NA	Closed	Process
	Control Rod Hydraulic Return Check Valve (3-181)			NA	Open	Process
3	Containment Air Sampling (VG 23, VG 26, 109-76A&B)		4	5	Open	GC

Table 4.7.2.b

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES NOT SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time(sec)	Normal Position	Action on Initiating Signal
2	RHR Return to Suppression Pool (10-39A,B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A,B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A,B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A,B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A,B)		2	20	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
4	RHR Reactor Head Cooling (10-32, 10-33)	1	1	25	Closed	SC
	Feedwater Check Valves (2-28 A,B)	2	2	NA	Open	Proc.
	Reactor Head Cooling Check Valve (10-29)	1		NA	Closed	Proc.
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Proc.
*	Hydrogen Monitoring (109-75 A, 1-4; 109-75 E-D, 1-2) Sampling Valves - Inlet		10	NA	NA	NA
*	Hydrogen Monitoring (VG-24, 25, 33, 34)		4	NA	NA	NA

*These valves are remote manual sampling valves which do not receive an isolation signal. Only one valve in each line is required to be operable.

3.10 LIMITING CONDITIONS FOR OPERATION

4.10 SURVEILLANCE REQUIREMENTS

b. The undervoltage automatic starting circuit of each disel generator shall be tested once a month.

c. Once per operating cycle, the actual conditions under which the diesel generators are required to start automatically will be simulated and a test conducted to demonstrate that they will start within 13 seconds and accept the emergency load and start each load within the specified starting time. The results shall be logged.

2. Batteries

The following battery chargers shall be operable:

a. Four battery chargers for the ± 24 volt neutron monitor and process radiation batteries.

b. Two of the three battery chargers for the 125 volt station batteries.

c. One of the two battery chargers for the 125 volt switchyard batteries.

2. Batteries

a. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured and logged.

b. Every three months the voltage of each cell to nearest 0.01 volt and specific gravity of each cell to the nearest 0.005 ap.gr. shall be measured and logged.

c. Once each operating cycle each station 125 volt battery shall be subjected to a rated loud discharge test. The specific gravity and voltage of each cell shall be measured after the discharge test and logged.

d. Two of the three battery chargers for the 24 volt ECCS Instrumentation.

3.10 LIMITING CONDITIONS FOR OPERATION

B. Operation with Inoperable Components

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.9.A shall be met except:

1. Diesel Generators

From and after the date that one of the diesel generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel generator is operable, the requirements of Specification 3.5.H.1 shall be satisfied.

2. Batteries

a. From and after the date that ventilation is lost in the battery room, portable ventilation equipment shall be provided.

b. From and after the date that one of the two 125 volt station battery systems is made or found to be inoperable for any reasons, continued reactor operation is permissible only during the succeeding three days provided Specification 3.5.H is met unless such battery system is sooner made operable.

c. From and after the date that one of the two 24 volt ECCS Instrumentation battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding three days unless such battery system is sooner made operable.

4.10 SURVEILLANCE REQUIREMENTS

B. Operation with Inoperable Components

1. Diesel Generators

When it is determined that one of the diesel generators in inoperable the requirements of Specification 4.5.H.1 shall be satisfied.

2. Batteries

Samples of the battery room atmosphere shall be taken daily for hydrogen concentration determination.

AUXILIARY ELECTRIC POWER SYSTEMS

A. The objective of this specification is to assure that adequate power will be available to operate the emergency safeguards equipment. Adequate power can be provided by any one of the following sources: either of the startup transformers, backfeed through the main transformer, the 4160 volt line from the Vernon Hydroelectric Station or either of the two diesel generators. The backfeed through the main transformer and 4160 volt Vernon line are both delayed-access offsite power sources. Backfeeding through the main transformer can be accomplished by disconnecting the main generator from the main transformer and energizing the auxiliary transformer from the 345 kv switchyard through the main transformer. The time required to perform this disconnection is approximately six hours. The 4160 volt line from the Vernon Hydroelectric Station can be connected to either of the two emergency buses within seconds by simple manual switching operation in the main control room.

Two 480 V Uninterruptible Power Systems; each consisting of a battery bank, battery charger, and a sold state inverter, supply power to the LPCIS valves via designated motor control centers. The 480 V Uninterruptible Power Systems are redundant and independent of any onsite power sources.

This Specification assures that at least two offsite, two onsite power sources, and both 480 V Uninterruptible Power Systems will be available before the reactor is taken beyond "just critical" testing. In addition to assuring power source availability, all of the associated switchgear must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.

Station service power is supplied to the station through either the unit auxiliary transformer or the startup transformers. In order to startup the station, at least one startup transformer is required to supply the station auxiliary load. After the unit is synchronized to the system, the unit auxiliary transformer carries the station auxiliary load, except for the station cooling tower loads which are always supplied by one of the startup transformers. The station cooling tower loads are not required to perform an engineered safety feature function in the event of an accident, therefore, an alternate source of power is not essential. Normally one startup transformer supplies 4160 volt buses 1 and 3 and the other supplies buses 2 and 4, however, the two startup transformers are designed with adequate capacity such that, should one become or be made inoperable temporary connections can be made to supply the total station load (less the cooling towers) from the other startup transformer.

A battery charger is supplied for each battery. In addition the two 125 volt station batteries and the two 24 volt ECCS instrumentation batteries each have a spare charger available. Since one spare 24 volt and one 125 volt charger are available, one battery charger can be allowed out of service for maintenance and repairs.

B. Adequate power is available to operate the emergency safeguards equipment from either startup transformer or for minimum engineered safety features from either of the emergency diesel generators. Therfore, reactor operation is permitted for up to seven days with both delayed-access offsite power sources lost.

Each of the diesel generator units is capable of supplying 100 percent of the minimum emergency loads required under postulated design basis accident conditions. Each unit is physically and electrically independent of the other and of any offsite power source. Therefore, one diesel generator can be allowed out of service for a period of seven days without jeopardizing the safety of the station.

In the event that both startup transformers are lost, adequate power is available to operate the emergency safeguards equipment from either of the emergency diesel generators or from either of the delayed-access offsite power sources. Also, in the event that both emergency diesel generators are lost, adequate power is available immediately to operate the emergency safeguards equipment from at least one of the startup transformers or from either of the delayed-access offiste power sources within six hours. The plant is designed to accept one hundred percent load rejection without adverse effects to the plant or the transmission system. Network stability analysis studies indicate that the loss of Vermont Yankee unit will not cause inability and consequent ripping of the connecting 345 kv and 115 kv lines. The Vernon feed is an independent source. Thus, the availability of the delayed-access offsite power sources is assured in the event of a turbine trip. Therefore, reactor operation is permitted with the startup transformers out of service and with one diesel generator out of service provided the NRC is notifed immediatley of the event and restoration plans.

Either of the two station batteries has enough capacity to energize the vital buses and supply d-c power to the other emergency equipment for 8 hours without being recharged. In addition, two 24 volt ECCS Instrumentation batteries supply power to instruments that provide automatic initiation of the ECCS and some reactor pressure and indication in the control room.

Due to the high reliability of probability of unwarranted shutdown by providing adequate time for reasonable repairs. This minimuzes the probability of unwarranted shutdown by providing adequate time for reasonable repairs. A station battery, ECCS Instrumentation battery, or an Uninterruptible Power System battery is considered inoperable if more than one cell is out of service. A cell will be considered out of service if its float voltage is below 2.13 volts and the specific gravity is below 1.190 at 77°F.

The battery room is ventilated to prevent accumulation of hydrogen gas. With a complete loss of the ventilation system, the accumulation of hydrogen would not exceed 4 percent concentration in 16 days. Therefore, on loss of battery room ventilation, the use of portable ventilation equipment and daily sampling provide assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

C. The minimum diesel fuel supply of 25,000 gallons will supply one diesel generator for a minimum of seven days of operation satisfying the load requirements for the operation of the safeguards equipment. Additional fuel can be obtained and delivered to the site from nearby sources within the seven day period.

4.10 AUXILIARY ELECTRICAL POWER SYSTEMS

Bases:

A. The monthly tests of the diesel generators are conducted to check for equipment failures and deterioration. The test of the undervoltage automatic starting circuits will prove that each diesel will receive a start signal if a loss of voltage should occur on its emergency bus. The loading of each diesel generator is conducted to demonstrate proper operation at less than the continuous rating and at equalibrium operating conditions. Generator experience at other generator stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 Introduction

By letters dated March 17, 1980 and August 28, 1980, as supplemented May 9, August 13, September 23 and October 14, 1980, Vermont Yankee Nuclear Power Corporation (licensee) requested amendments to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The proposed amendments would revise the Technical Specifications to authorize the licensee to:

A. Replace existing pressure switches that sense reactor pressure and reactor water level with analog loops,

B. Install an ATWS recirculation pump trip (RPT) system of the Monticello design, to trip on low-low reactor water level or high reactor pressure, and

C. Require Type C testing of certain containment isolation valves to reflect changes brought about for post-accident hydrogen sampling resulting from NUREG-0578, "TMI Lessons Learned Task Force Status Report on Short Term Recommendations."

2.0 Evaluation

2.1 Analog Trip System

The licensee has proposed certain modifications to Appendix A of the operating license for the Vermont Yankee Plant. These modifications involve installing a new design improvement for safety system instrumentation for General Electric Company (GE) boiling water reactors for the reactor protection system (RPS) and emergency core cooling system (ECCS). The proposed modification is referred to as the analog trip system.

This analog trip system is similar to that developed by GE and described in GE's Topical Report NEDO-21617 of April 1977 and NEDO-21617-1 of January 1978 entitled, "Analog Transmitter/Trip Unit System" (ATTUS). GE submitted this topical report to the NRC staff for review and it was found acceptable by the staff as stated in the letter to GE dated June 27, 1978.

Since the licensee had not referenced this approved GE topical report, we requested the licensee to compare their analog trip system design to that described in GE's topical report. Our evaluation of the licensee's proposed design change (March 17, 1980) and responses to our inquiries (May 9, 1980; and August 13, 1980 and September 23, 1980) is as follows:

The licensee identified the analog trip system to be a replacement of pressure and differential pressure switches which sense reactor pressure and reactor water level with analog channels each consisting of a transmitter, indicator and trip unit. This analog trip system is designed to increase plant reliability, reduce setpoint drift and improve safety of the plant. The equipment to be used includes Rosemount Model 1152 analog transmitters with "E" output codes and Rosemount Model 510DU trip units.

During our review of the licensee's proposed change and the results of the comparison of their design to GE's Topical Report NEDO-21617, several items were identified as having possible safety implications. The areas identified pertained to: 1) ATWS recirculation pump trip circuitry separation, 2) power supply, 3) arrangement of panel equipment, and 4) environmental qualification.

1) The ATWS recirculation pump trip circuitry is to be included within the ECCS cabinets for the analog trip system. The NRC staff was concerned with the adequacy of separation between the Class 1E and non-class 1E wiring for ATWS pump trip. After discussions with the licensee, we were informed that the ATWS recirculation pump trip circuitry will be a Class 1E system and its wiring will be routed with the ECCS Class 1E wiring. Therefore, we consider this concern resolved.

2) The GE topical report states that DC power supplies can be connected in parallel to obtain a higher output current. However, the Vermont Yankee plant design is such that a single power supply is sufficient to handle its associated divisional/channel load. Hence, the licensee does not plan to use parallel power supplies. Each division/channel will be provided with a separate and independent power supply. Therefore, we find this design acceptable.

3) The location of equipment and wiring of the trip unit panel differ from that described in the GE topical. The trip units and associated trip relays are to be arranged differently and all field wiring will be routed through the top of the cabinet instead of the bottom. Separation will be maintained between Class 1E and non-Class 1E wiring. Therefore, this proposed change in physical arrangement does not pose a concern and is acceptable.

4) The licensee states that analog trip system cabinets are to be located in the reactor building. According to the GE topical report, the power supplies and trip relays located within these cabinets are qualified for use in control room environments. Discussions with the licensee resulted in a commitment from the licensee assuring that all equipment installed in the reactor building will be seismically and environmentally qualified for normal plant operation and the worst-case accident reactor building environment.

We have reviewed the service environment for this equipment and conclude that it is a mild environment. The qualification of safety-related electrical equipment to function in environmental extremes not associated with accident conditions is the responsibility of the licensee to evaluate and document in a form that will be available for the NRC to audit. Qualification to assure functioning in mild environments must be completed by June 30, 1982.

Based on our prior review and approval of GE's Topical Report NEDO-21617 and our recent review of the licensee's submittals which included a comparison of Vermont Yankee's design to the GE topical, we conclude that the proposed modifications (analog trip system) meet the applicable provisions of IEEE 279-1971, GDC 13, and GDC 20 as described above.

2.2 ATWS Recirculation Pump Trip

Installation of an RPT, is for the purpose of providing a partial backup system to the Reactor Protection System (RPS). In the unlikely event that the plant experienced an operational transient without a subsequent scram, the RPT would reduce core power generation by rapidly reducing core flow, thus mitigating the short term consequences of the transient. The basis for the system's design is the Monticello type RPT with a time delay on low-low reactor water level which is described in the letter from NRC to Vermont Yankee, dated January 8, 1979.

Because of the increment of safety added by installation of an RPT of such a design, on February 21, 1980 the NRC ordered that this installation be completed by December 31, 1980. We find this modification acceptable.

2.3 Containment Isolation Valve Testing

The licensee has requested certain changes in Type C containment valve testing in order to meet the requirements of NUREG-0578, "TMI Lessons Learned Task Force Status Report and Short-Term Recommendations," with respect to post-accident hydrogen sampling. It should be noted that we are currently in the process of reviewing all generic issues pertaining to Appendix J leak rate testing for operating nuclear power plants at the Franklin Institute.

The licensee has proposed that certain valves of the hydrogen monitoring system (part of the CAD system), be deleted from Table 4.7.2b of the Technical Specifications. The hydrogen monitoring system was designed and installed to meet Seismic Category I requirements and as an integral part of containment. The system, which does not require an isolation system, has valves 109-75A-D; 1 & 2 and 109-76 A & B. These valves do not receive PCIS signals and are remotely operated from the main control room and have not previously been subject to Type C leakage tests. The licensee regards this system as an extension of primary containment not requiring a system isolation and remaining open in a post-accident condition. The containment hydrogen monitoring system would therefore operate continuously to satisfy the associated short term lessons learned requirement.

General Design Criterion 56 explicitly states that each line that connects to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other basis. In this regard, the Standard Review Plan at Section 6.2.4, paragraph II.3.b permits the deviation from GDC 56 for lines that are part of engineered safety systems. These systems may have manual valves, but provisions should be made to detect possible leakage from these lines outside of containment.

On this basis we are permitting the use of remote manual valves in the hydrogen monitoring system for those lines which penetrate the primary containment, since they are an integral part of an engineered safety system. However, we require that for this system one valve in every line penetrating the primary containment be regarded as an isolation valve and be appropriately included in Table 4.7.2b. With regard to these valves, we currently are reviewing at the Franklin Institute all requirements for Appendix J testing, which will address this system as well as other systems in the Vermont Yankee Plant. Therefore, until this review is completed, we will defer judgment on requiring Type C testing for these isolation valves. However, the licensee has agreed that the hydrogen monitoring system will be subject to a Type A Integrated Leak Rate Test.

The licensee has also proposed that solenoid valves VG-23, VB-26 and 109-76 A&B, which were installed in the radiation monitor inlet and outlet lines to close on receipt of a primary containment isolation signal and provide containment isolation, be added to Table 4.7.2b and thus be subject to Type C leakage tests.

Appendix J, 10 CFR Part 50, requires that valves designed to operate subsequent to a design basis accident, which may become a part of the containment isolation system barrier during post-accident operation, be subject to Type C leakage rate tests.

By letter dated October 15, 1980, the licensee indicated that the isolation valves VG-23 and VG-26 are located as close as practical to the hydrogen monitoring system, which is considered an extension of containment and that valves 109-76 A&B are located as close as practical to the torus wetwell. Based on the above considerations we find this change in containment isolation valve testing acceptable.

3.0 Environmental Consideration

We have determined that the license amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 3, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility) located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to authorize replacement of existing pressure switches that sense reactor pressure and water level with analog loops; installation of an ATWS recirculation pump trip; and modification of testing requirements for containment isolation valve tests to reflect changes brought about to permit post-accident hydrogen sampling capability.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated March 17, 1980, and August 28, 1980, as supplemented May 9, August 13, September 23, and October 14, 1980, (2) Amendment No. 58 to License No. DPR-28, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 3rd day of November, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
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