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Docket No. 50-271

Yankee Atomic Electric Company
ATTN: Mr. G. Carl Andognini
Assistant to the Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility License No. DPR-28. This amendment includes Change No. 20 to the Technical Specifications, and is in response to Vermont Yankee's request dated August 23, 1974.

This amendment incorporates (1) a limiting condition for operation relative to the off-gas system isolation instrumentation and the associated surveillance requirements, (2) a limiting condition for operation relative to the condenser low vacuum level trip and the primary containment isolation instrumentation and the associated surveillance requirements, and (3) clarification of the intent of the limiting conditions for operation relative to the core and containment cooling systems availability during refueling and the associated surveillance requirements. In addition, non-applicable data concerning reactor water chemistry effects on primary system materials has been deleted from the bases of the Technical Specifications.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

CHebron
HJMcAlduff, ORO
JRBuchanan, ORNL
TBAbernathy, DTIE

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Enclosures:
1. Amendment No. 9
   w/Change No. 20
2. Safety Evaluation
3. Federal Register Notice

LB

cc w/encls:
See attached

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Docket No. 50-271

Yankee Atomic Electric Company
ATTN: Mr. G. Carl Andognini
Assistant to the Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility License No. DPR-28. This amendment includes Change No. 20 to the Technical Specifications, Appendix A, and is in response to Vermont Yankee's request dated August 23, 1974.

This amendment incorporates (1) the limiting condition for operation of the off-gas system isolation instrumentation and the associated surveillance requirements, (2) the limiting condition for operation of the condenser low vacuum level trip on the primary containment isolation instrumentation and the associated surveillance requirements, (3) clarification of the intent for the limiting conditions for operation of the core and containment cooling systems availability during refueling and the associated surveillance requirements, and (4) the deletion of now-applicable data from the bases concerning reactor water chemistry effects on primary system materials.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 9
   w/Change No. 20
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See attached

bcc: H. J. McAlduff, ORO
J. R. Buchanan, ORNL
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DATE ▶	10/4/74	10/4/74	10/4/74	10/22/74	10/22/74

Form AEC-318 (Rev. 9-53) AECM 0240   GPO 643-16-81465-1 445-678 cc w/encls:

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cc w/encls and copy of 8/23/74 application:
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Boston, Massachusetts 02203

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VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
License No. DPR-28

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated August 23, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

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

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DATE ▶						

B. Technical Specifications

The Technical Specifications contained in Appendices A and B as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 20.

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

FOR THE ATOMIC ENERGY COMMISSION

Origin
Karl R.

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 20 to the
Technical Specifications

Date of Issuance:
OCT 23 1974

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ATTACHMENT TO LICENSE AMENDMENT NO. 9

CHANGE NO. 20 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-28

Delete pages 33, 41, 44, 46, 54, 58, 64, 65, 94, 95, 101, 104, 119, 120, 137 and 164 from the Technical Specifications and insert the attached replacement pages bearing the same numbers. The changes are shown by marginal lines.

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3.2 LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Isolation

When primary containment integrity is required, in accordance with Specification 3.7, the instrumentation that initiates primary containment isolation shall be operable in accordance with Table 3.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

The instrumentation that initiates the isolation of the reactor building ventilation system and the actuation of the standby gas treatment system shall be operable in accordance with Table 3.2.3.

20 D. Off-Gas System Isolation

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

E. Control Rod Block Actuation

During reactor power operation the instrumentation that initiates control rod block shall be operable in accordance with Table 3.2.5.

4.2 SURVEILLANCE REQUIREMENT

B. Primary Containment Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.3.

20 D. Off-Gas System Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.

E. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5.

TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Condition for Operation are not Satisfied (Note 2)
2	Low-Low Reactor Vessel Water Level	≥6' 10.5" above the top of active fuel	A
2 of 4 in each of 2 channels	High Main Steam Line Area Temperature	<212°F	B
2/steamline	High Main Steam Line Flow	≤120% of rated flow	B
2/(Note 1)	Low Main Steam Line Pressure	≥ 850 psig	B
2/(Note 6)	High Main Steam Line Flow	≤40% of rated flow	B
2	Low Reactor Vessel Water Level	Same as Reactor Protection System	A
2	High Main Steam Line Radiation (7) (8)	≤3 X background at rated power (9)	B
2	High Drywell Pressure	Same as Reactor Protection System	A
20 2/(Note 10)	Condenser Low Vacuum	≥12" Hg absolute	A
1	Trip System Logic	______________	A

TABLE 3.2.2 NOTES

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

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

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

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

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

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

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

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

7. This signal also automatically closes the mechanical vacuum pump suction line isolation valves.

8. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

9. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

10. A key lock switch is provided to permit bypass of this trip function to enable plant startup and shutdown
20 provided that (1) both the turbine stop valve and bypass valve are closed and (2) the condenser vacuum is less than 12 inches Hg absolute.

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TABLE 3.2.4

OFF GAS SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Condition for Operation are not Met
2	Air Ejector Off-Gas Radiation	≤ 1.5 Ci/sec (3) ≤ 0.3 Ci/sec (4)	Notes 1 & 2
20 1	Time Delay (Air Ejector Suction Valve Isolation)(17-157&17-157A)	≤ 15 minutes ≤ 1 minute	Notes 1 & 2
1	Logic Bus Power Monitor	---	Notes 1 & 2
1	Trip System Logic	---	Notes 1 & 2
1	Augmented Off-Gas Radiation	≤ 0.07 Ci/sec (6)(7)	Note 5
20 1	Time Delay (Stack Off-Gas Valve Isolation)(15TD & 16TD)	≤ 2 minutes ≤ 30 minutes	Note 5
1	Trip System Logic	---	Note 5

Note 1 - If the minimum number of operable instrument channels in one trip system are not available, that system shall be tripped downscale and reactor power operation is permissible for only seven successive days, unless the system is sooner made operable.

Note 2 - If the minimum number of operable instrument channels in both systems are not available, initiate an orderly shutdown and have the reactor in the cold shutdown condition within 24 hours.

Note 3 - If the radiation level exceeds 1.5 Ci/sec (30 minute decay level), the air ejector suction valves shall close within one minute.

20 Note 4 - If the radiation level exceeds 0.3 Ci/sec (30 minute decay level), the air ejector suction valves shall close unless the radiation level decreases to less than 0.3 Ci/sec within 15 minutes.

Note 5 - At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

Note 6 - If the radiation level exceeds 0.07 Ci/sec the stack isolation valve shall close within 2 minutes if the dryer-absorber system bypass valves (OG-145 & -146) are open and within 30 minutes if these valves are closed unless the radiation level has decreased below the trip point sooner.

Note 7 - During plant startup or shutdown conditions (changing disintegration energy; $\bar{E}\gamma$) the trip point may be adjusted upward but must always remain below a rate corresponding to $0.08/\bar{E}\gamma$.

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)	every 3 months	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)	every 3 months	---
High Main Steam Line Radiation	(Notes 1 & 7)	each refueling outage	once each day
High Drywell Pressure	(Note 1)	every 3 months	---
20 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)	---

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TABLE 4.2.4

MINIMUM TEST & CALIBRATION FREQUENCIES

20 OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

	Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
20	Air ejector off-gas radiation	(Notes 1 & 7)	each refueling outage	once each day
	Logic Bus Power Monitor	(Note 1)	none	once each day
	Trip System Logic (SJAE)	every 6 months (Note 2)	every 6 months (Note 3)	---
	Augmented off-gas radiation	(Notes 1 & 8)	every 3 months	once each day
20	Trip System Logic (AOG)	every 6 months (Note 2)	every 6 months (Note 3)	---

3.2 (continued)

High radiation monitors in the mean steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 limits are not exceeded for the control rod drop accident and 10 CFR 20 limits are not exceeded for gross fuel failure during reactor operations. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

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

Low condenser vacuum has been added as a trip of the Group 1 isolation valves to prevent release of radioactive gases from the primary coolant through condenser. The set point of 12 inches of mercury absolute was selected to provide
20 sufficient margin to assure retention capability in the condenser when gas flow is stopped and sufficient margin below normal operating values.

The HPCI and/or RCIC high flow, steam supply pressure, and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 6 valves. The trip settings are such that core uncovering is prevented and fission product release is within limits.

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

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

3.2 (continued)

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

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

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

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors provide isolation capability on the air ejector suction line. Isolation is
initiated when either instrument reaches its upscale trip point. The immediate trip (within 1 minute) set point
20 of 1.5 Ci/sec (30 minute decay) is based upon limiting the whole body dose at the site boundary to less than 5 Rem
in the unlikely event of a boundary failure in the off-gas system concurrent with a spike release of radioactivity
from the fuel. The assumption has been made that the rate of radioactivity increase within the 1 minute valve
closure time period would be less than a factor of 5 based upon actual experience with such events. The delayed
trip (within 15 minutes) set point of 0.3 Ci/sec (30 minute decay) is based upon limiting the whole body dose at
the site boundary to less than 5 Rem in the event of off-gas system boundary failure concurrent with an off-gas
release from the fuel of a lower value than considered above.

Two radiation monitors provide an isolation capability on the off-gas line at the plant. Stack Isolation is initiated
when either instrument reaches its upscale trip point. The trip point of 0.07 Ci/sec has been derived from the
release limit of $0.08/\bar{E}_\gamma$ assuming minimum holdup and corresponding maximum average disintergration energy and an
isotopic mix corresponding to power operation. An energy shift is concurrent with plant shutdown, and consequently,
20 the trip point may be adjusted to accommodate the change in mix yet remain below $0.08/\bar{E}_\gamma$. The limit, $0.08/\bar{E}_\gamma$, is
established to prevent an off site annual whole body dose of 500 mRem (the 10CFR20 limit). The time delays are
established based upon the flow path (e.g. 30 minutes if the carbon beds are in service and 2 minutes if they are
bypassed).

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. Any one upscale trip or two downscale trips of either set of monitors will cause the desired action. Trip settings for the monitors on the refueling floor are based upon initiating normal ventilation isolation and

3.5 LIMITING CONDITIONS FOR OPERATION

3. If the requirements of Specification 3.5.G cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

H. <u>Minimum Core and Containment Cooling System Availability</u>

1. During any period when one of the standby diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the low pressure core cooling and containment cooling subsystems connection to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all core and containment cooling subsystems may be inoperable provided no
20 work is permitted which has the potential for draining the reactor vessel.

4.5 SURVEILLANCE REQUIREMENTS

H. <u>Minimum Core and Containment Cooling System Availability</u>

1. During reactor operation, when it is determined that one of the standby diesel generators is inoperable, all low pressure core cooling and containment cooling service water systems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

2. During a refueling outage, the applicable surveillance requirements of Specification 4.5.H.1
20 shall be performed immediately and monthly except that the operable diesel shall be tested immediately and weekly.

3.5 LIMITING CONDITIONS FOR OPERATIONS

4. When irradiated fuel is in the reactor vessel
and the reactor is in the refueling condition,
both LPCI subsystems, or both Core Spray systems,
20 or one diesel generator may be inoperable provided
that a source of water of greater than 300,000 gal.
is available to the operable core cooling sub-
system.

I. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5 SURVEILLANCE REQUIREMENTS

I. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Every month prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI subsystem or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

3.5 (Cont'd)

G. Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System (RCIC) is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the emergency core cooling systems. The RCIC meets this requirement. Reference Section 14.5.4.4 FSAR. The HPCIS provides an incidental backup to the RCIC system such that in the event the RCIC should be inoperable no loss of function would occur if the HPCIS is operable.

H. Minimum Core and Containment Cooling System Availability

The core cooling and the containment cooling subsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that adequate cooling capacity is available by precluding any combination of inoperable components from fulfilling the core and containment cooling function. It is permissible, based upon the low heat load and other methods available to remove the residual heat, to disable all core and containment cooling systems for maintenance if the reactor is cold and shutdown and there is no potential for draining the reactor vessel. However, if refueling operations
20 are in progress, one coolant injection system, one diesel and a residual of at least 300,000 gallons is required to assure core flooding capability.

I. Maintenance of Filled Discharge Pipe

Full discharge lines are required when the core spray subsystems, HPCI and RCIC are required to be operable to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start.

J. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300°F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 even considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the IAC limit.

The maximum average planar LHGR shown in Figures 3.5.1.A and 3.5.1.B is the same as that shown on the curve labeled "Ω" (omega) on Figures 1-G and 2-G in the General Electric letter of J. A. Hinds to V. A. Moore, "Plant Evaluations with GEGAP-III," dated December 12, 1973, based on calculations employing the models described in the General Electric reports NEDM-10735 as modified by the General Electric report NEDO-20181 and the aforementioned General Electric letter of December 12, 1973.

4.5 (cont'd)

The pump operability check will be performed by starting the turbine manually, valves will also be stroked by manual actuation of the operators.

G. Reactor Core Isolation Cooling System

Frequency of testing of the RCIC system is the same as the HPCIS and demonstrates that the system is operable if needed.

H. Minimum Core and Containment Cooling System Availability

Immediate testing followed by daily tests of all low pressure core cooling subsystems and containment cooling
service water systems including the operable standby diesel generator upon determination of one inoperable
diesel generator adequately demonstrates the availability of core and containment cooling subsystems. This
20 testing frequency is reduced to monthly during a refueling outage to permit various surveillance inspections
on equipment. However, at least one diesel is maintained fully operable and tested weekly.

I. Maintenance of Filled Discharge Pipe

Observation of water flowing from the discharge line high point vent monthly assures that the core cooling subsystems will not experience water hammer damage when any of the pumps are started. Core spray subsystems and LPCI subsystems will also be vented through the discharge line high point vent following a return from an inoperable status to assure that the system is "solid" and ready for operation.

J. and K. Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

3.6 & 4.6 (cont'd)

calculated on the basis of the radioiodine concentration limit of 1.1 uCi of I-131 does equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a X/Q = 3.9 x 10^{-3} sec/m^3 (Pasquill D and 0.33 m/sec equivalent) and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

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Figure Deleted

Table 4.7.2 NOTES

1. Isolation signals are as follows:

Group 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Low-low reactor water level
2. High main steam line radiation
3. High main steam line flow
4. High main steam line tunnel temperature
5. Low main steam line pressure (run mode only)

20 6. Condenser low vacuum

Group 2: The valves in Group 2 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure

Group 3: The valves in Group 3 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High/low radiation - reactor building ventilation exhaust plenum or refueling floor

Group 4: The valves in Group 4 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High reactor pressure

Group 5: The valves in Group 5 are closed upon low reactor water level.

Group 6: The valves in Group 6 are closed upon any signal representing a steam line break in the HPCI system's or RCIC system's respective steam line. The signals indicating a steam line break for the respective steam line are as follows:

1. High steam line space temperature
2. High steam line flow
3. Low steam line pressure
4. High temperature in the main steam line tunnel (30 minute delay for the HPCI and the RCIC)

2. The closure time shall not be less than 3 seconds.

3.9 LIMITING CONDITIONS FOR OPERATION

radiation monitor is operating and the off-gas system pressure and temperature are measured continuously.

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C. Liquid Process Monitors

When the systems are in operation, the activity in the service water and the cooling tower water systems shall be continually monitored and either of the following conditions shall be met:

1. The process liquid radiation monitor shall be operable, or

2. Samples shall be taken daily and analyzed for gross radioactivity.

4.9 SURVEILLANCE REQUIREMENTS

C. Liquid Process Monitors

The liquid process monitors shall be functionally tested by injecting a simulated electrical signal into the measurement channels monthly and calibrated quarterly with an appropriate radiation source. Each monitor shall have an instrument check at least daily.

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. DPR-28

(CHANGE NO. 20 TO THE TECHNICAL SPECIFICATIONS)

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

INTRODUCTION

By letter dated August 23, 1974, the Vermont Yankee Nuclear Power Corporation (VYNPC) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The proposed changes would:

1. add the condenser low vacuum trip function to the primary containment isolation instrumentation,

2. add the augmented off gas (AOG) system radiation monitor high level trip function to the off gas system isolation instrumentation, and

3. reword the section relating to core and containment cooling system availability during refueling for clarification of intent.

DISCUSSION

1. As previously discussed in the Safety Evaluation (Sections 3.1 and 4.1) for Change No. 13 to the Technical Specifications dated January 17, 1974, regarding a low vacuum in the turbine condenser, closure of the main steamline isolation valve (MSIV) upon condenser low vacuum is desirable to reduce the release of radioactive gases from the primary coolant through the condenser. VYNPC will use the instrumentation previously installed for a reactor scram from a low vacuum

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signal to accomplish MSIV closure. The modification to the primary containment isolation instrumentation necessary to accomplish MSIV closure upon condenser low vacuum will be completed during the Fall 1974 refueling outage scheduled to occur in late September. A trip setpoint of 12 inches Hg absolute has been established to provide sufficient margin to assure retention of the gases and sufficiently below normal operating values to prevent unnecessary MSIV closures. Bypass of this trip function is required during startup but the bypass is permissible only if the turbine bypass valve and stop valve are closed to prevent release of the gases through the turbine and the vacuum is below the trip setpoint. The reactor would be at low power level during the bypass period. At low reactor power only small quantities of radioactive gases would be expected to be released into the reactor coolant and a vacuum below 12 inches Hg absolute exists for only a short period of time. The change adds a desirable feature to the primary system into the condenser in the event of low condenser vacuum. Even in the reactor startup condition when trip must be bypassed, the change in specifications still provides additional control of possible releases by assuring that the turbine bypass valve and stop valve are closed. The addition of this system does not affect any other safety feature of the facility.

2. As previously discussed in the Safety Evaluation (Sections 3.2 and 4.2) for Change No. 13 to the Technical Specifications dated January 17, 1974, the equipment modifications necessary to provide automatic isolation of the AOG system would not be available until the Fall 1974 refueling outage. Prior to providing automatic isolation, operator action was required to isolate the AOG system at the established trip level settings. Automatic isolation provided by the change will provide more rapid closure of isolation valves upon high level readings than possible by operator action. The instrumentation which informs the reactor operator of the high radiation levels are retained so that the ability of the operator to take appropriate action is not adversely affected. Only the speed of isolation is effected and that is enhanced. The addition of this system does not affect any safety feature of the facility. Requirements for operation of the AOG monitoring system previously included in Specification 3.9.B.2 have been transferred to Table 3.2.4. The trip setting on the AOG radiation monitors is consistent with the limit established in Specification 3.8.C.1.a for maximum release rate of gross radioactivity.

3. The intent of Specification 3.5.H.3. and 4. was to allow drainage of the torus for periodic inspection and surveillance during refueling outages. However, the specifications were worded to limit periods in which certain core and containment cooling subsystems could be inoperable for up to 30 days. This wording could be interpreted

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as preventing inspection, surveillance and maintenance activities which extended beyond 30 days. The specifications have been reworded to clarify the intent of the requirements for the minimum core and containment cooling system availability during refueling. An additional requirement has been added to Specification 3.5.H.4 regarding availability of cooling water in case certain subsystems of the core and containment cooling system are inoperable. Therefore the limiting 30 day period in which a subsystem was allowed to be inoperable during refueling operations has been deleted and replaced by the additional requirement for a known source of cooling water in the specification. The change in specification deletes the 30-day limitation as such and replaces it with the additional requirements for an adequate cooling water source during refueling operations, which was the basic purpose for the 30-day limitation initially and eliminates the possible interpretation that the 30-day limit applied without regard to an adequate alternative cooling water source.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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.

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 9 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, located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment incorporates (1) the limiting conditions for operation of the off-gas system isolation instrumentation and the associated surveillance requirements, (2) the limiting conditions for operation of the condenser low vacuum level trip on the primary containment isolation instrumentation and the associated surveillance requirements, and (3) clarification of the intent for the limiting conditions for operation of the core and containment cooling systems availability during refueling and the associated surveillance requirements.

The application for the amendment complies 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.

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For further details with respect to this action, see (1) the application for amendment dated August 23, 1974, (2) Amendment No. 9 to License No. DPR-28, with Change No. 20, and (3) the Commission's concurrently issued 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 at 224 Main Street, Brattleboro, Vermont 05301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 23rd day of October, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

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