MAR 2 8 1974

Docket No. 50-271

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. James E. Griffin, President
77 Grove Street
Rutland, Vermont 05701

Change No. 16
License No. DPR-28

Gentlemen:

Your letters dated February 21, March 8, and March 19, 1974, proposed changes to the Technical Specifications of Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would reduce the relief valve trip settings for end-of-cycle conditions and allow reactor operation at a reduced power level in the event that one relief valve was inoperable. The proposed changes would become effective following the earliest convenient outage which you have scheduled for March 29, 1974. Other changes clarify the intent of the specifications for maximum worth of an insequence control rod and function of the relief valve in the automatic depressurization system (ADS) operation.

During our review, we informed your staff that certain modifications to the proposed changes were necessary to meet Regulatory requirements. These modifications have been made.

Our letter dated March 1, 1974, provided an interpretation of surveillance frequency and interval as requested by a letter dated February 11, 1974, from the Yankee Atomic Electric Company. The definition is appropriate for inclusion in the Vermont Yankee Technical Specifications. We informed your staff that appropriate definitions for surveillance frequency and surveillance interval would be added to these proposed changes. These definitions have been added.

On the basis of our review reflected in the enclosed Safety Evaluation, we have concluded that the proposed changes do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Vermont Yankee Nuclear Power
Corporation

MAR 28 1974

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility License No. DPR-28 are hereby changed by replacing pages 4, 16, 17, 70, 76, 92, 100, 108, 121, and 122 with the enclosed revised pages.

Sincerely,

Orig...
Den...

for/ Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Technical Specification
(pages 4, 16, 17, 70, 76,
92, 100, 108, 121, and 122)

cc: See next page

Distribution
Docket File
AEC PDR
Local PDR
Branch Reading
JRBuchanan, ORNL
TBAbernathy, DTIE
DJSkovholt, L:OR
TJCarter, L:OR
ACRS (16)
RO (3)
OGC
DLZiemann, L:ORB #2
FDAnderson, L:ORB #2
RMDiggs, L:ORB #2
NDube, L:OPS
M Jinks, DRA (4)
BScharf, DRA (15)
PCollins, L:OLB
SVarga, L:RO
S Kari, L:RP

AG discussed this with Tom Englehardt indicating urgent need to issue and that it would be done without OGC concurrence unless he chooses to elevate the issue. Mr Englehardt did not elect to elevate the issue
DJS 3/28/74

OFFICE ➤ X7403	L:ORB #2	L:ORB #2	L:ORB #2	L:OR DJS		
SURNAME ➤	FDAnderson:sjh	RMDiggs	DLZiemann	DJSkovholt		lg
DATE ➤	3/26/74	3/26/74	3/26/74	3/28/74		

Mr. Lawrence E. Minnick, Vice President
Vermont Yankee Nuclear Power Corporation
Turnpike Road, Route 9
Westboro, Massachusetts 01581

John A. Ritsher, Esquire
Ropes and Gray
225 Franklin Street
Boston, Massachusetts 02110

Gregor I. McGregor, Esquire
Assistant Attorney General
Department of the Attorney General
State House, Room 370
Boston, Massachusetts 02133

Richard E. Ayres, Esquire
David Schoenbrod, Esquire
National Resources Defense Council, Inc.
15 West 44th Street
New York, New York 10036

Honorable Kimberly B. Cheney
Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

John A. Calhoun
Assistant Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

Jonathon N. Brownell, Esquire
Paterson, Gibson, Noble & Brownell
26 State Street
Montpelier, Vermont 05602

Peter S. Paine, Jr., Esquire
Cleary, Gottleib, Steen & Hamilton
52 Wall Street
New York, New York 10005

J. Eric Anderson, Esquire
Fitts and Olson
16 High Street
Brattleboro, Vermont 05301

William H. Ward, Esquire
Assistant Attorney General
Office of the Attorney General
State Capitol Building
Topeka, Kansas 66612

Donald W. Stever, Jr., Esquire
Office of the Attorney General
State House Annex
Concord, New Hampshire 03301

Chairman, Vermont Public Service
Corporation
Seven School Street
Montpelier, Vermont 05602

John W. Stevens, Director
Conservation Society of Southern
Vermont
Post Office Box 256
Townshend, Vermont 05353

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

Mr. David M. Scott
Radiation Health Engineer
Agency of Human Services
Division of Occupational Health
Post Office Box 607
Barre, Vermont 05641

cc w/enclosures and VY ltrs dtd
2/21, 3/8, and 3/19/74:
Mr. Wallace Stickney
Environmental Protection Agency
JFK Federal Building
Boston, Massachusetts 02203

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

CHANGE NO. 16 TO TECHNICAL SPECIFICATIONS

INTRODUCTION

By letters dated February 21, March 8, and March 19, 1974, Vermont Yankee Nuclear Power Corporation (VYNPC) proposed changes to the Technical Specifications of Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would reduce the relief valve trip settings for end-of-cycle conditions and allow reactor operation at a reduced power level in the event that one relief valve was inoperable. These proposed changes fulfilled our request for such information in our letter dated November 16, 1973. The Safety Evaluation associated with Change No. 13 dated January 17, 1974, stated that these changes to the relief valve settings would be required for the current operating cycle as well as changes to set point error allowances. Based upon a request dated February 11, 1974, from the Yankee Atomic Electric Company, we provided an interpretation for several plants, including Vermont Yankee, regarding allowable deviation from any surveillance frequency. Concurrent with this request, we informed the Vermont Yankee staff that appropriate definitions for surveillance frequency and surveillance interval should be added to the Technical Specifications.

DISCUSSION

Section 1.0 - Definitions

Definitions for surveillance frequency and surveillance interval have been added to clarify the intent of surveillance requirements. These definitions are consistent with those given for other nuclear power reactors.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Section 2.2 - Reactor Coolant System

To maintain the margin between the maximum pressure for any overpressure transient and the safety valve trip settings, the relief valve trip settings need to be reduced by 10 psi for end-of-cycle conditions. The proposed change in the relief valve trip settings is consistent with the information submitted by VYNPC in a letter dated November 12, 1973, and fulfills the requirements stated in our letter dated November 16, 1973, for necessary technical specification changes. Our review of the proposed relief valve trip setting reduction was completed prior to our approval for the use of the reload fuel in the Vermont Yankee reactor given in our letter dated November 16, 1973. The incorporation of the allowable set point error into the nominal settings for each relief and safety valve requires the licensee to use settings on the valve which will take into account possible set point errors. VYNPC was informed that this allowance for set point errors should be incorporated in the proposed changes to the safety and relief valve trip settings.

Section 3.3.B - Control Rods

Specification 3.3.B.4(a) has been rewritten to clarify the intent of the limit on reactivity worth for any insequence control rod. For the postulated control rod drop out accident analyzed to establish the maximum reactivity worth for any insequence control rod, the assumption was made that an insequence control rod being withdrawn from the core was decoupled from the rod drive and remained in the core until it dropped out of the core causing a rapid reactivity input. Due to the control rod design, the postulated decoupled control rod can only fall the distance which the control rod drive has been withdrawn and not necessarily the entire length of the control rod. The rewrite of Specification 3.3.B.4(a) clarifies the control rod withdrawal conditions and reflects the relationship between the control rod design feature and the postulated accident conditions.

Section 3.5.F - Automatic Depressurization System

The current 24-hour limit on reactor operation with a relief valve inoperable was incorporated by Change No. 15 to the Technical Specifications since all four relief valves were assumed operable in the accident analysis. However, the relief valves of the Automatic Depressurization System (ADS) are a backup to the HPCIS and only two of the four valves are required during the postulated small pipe break with HPCI failure. To clarify the operation of the ADS during such pipe break accidents and to define the function of the relief valves, Specification 3.5.F.2 has been rewritten to recognize the function

OFFICE ➤						
SURNAME ➤						
DATE ➤						

performed by the relief valves of the ADS with pressure relief for the postulated small pipe break. Changes to Specification 3.6.D.1 also have been made concurrently to reflect this need for clarification.

Section 3.6.D - Safety and Relief Valves

To complete the clarification of the function of the relief valves for small pipe breaks with HPCI failure and for postulated reactor transient accidents, the relationship of relief valve operation to mitigate overpressure transients for postulated transient accidents has been transferred from Specification 3.5.F.2 to Specification 3.6.D.1. As previously stated, all four relief valves have been assumed operable in the accident analysis for reactor transients. The fourth relief valve is required to open only to mitigate over-pressure transients at high power levels. Therefore, reactor power reduction can be initiated immediately after a relief valve is found inoperable. The reactor power would be reduced to a power level below which the fourth relief valve would not be required in case of the controlling accident. Reactor operation at the reduced power level with the inoperable relief valve can continue indefinitely without changing the normal safety margin for reactor operation. The analysis used to determine the reduced power level has been reviewed and is acceptable to the Regulatory staff. Until the scram reactivity concerns at end-of-cycle are resolved on a generic basis, an analysis to determine the acceptable reduced power level must be performed for each operating cycle.

CONCLUSION

On the basis of our review, we have concluded that the changes proposed by VYNPC, as modified, do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The changes to the Technical Specifications as proposed by Vermont Yankee and modified by the Regulatory staff should be issued.

/s/

Fredric D. Anderson
Operating Reactors Branch #2
Directorate of Licensing

/s/

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

Date: MAR 28 1974

OFFICE➤						
SURNAME➤						
DATE➤						

W. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

3. Shutdown means conditions as above such that the effective multiplication factor (keff) of the core shall be less than 0.99.

X. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate circuit in question.

Y. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Z. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

AA. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

1.2 SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings for controlling reactor system pressure.

Objective:

To provide for protective action in the event that the principle process variable approaches a safety limit.

Specification:

A. Reactor coolant high pressure scram shall be less than or equal to 1055 psig.

B. Primary system relief and safety valve nominal settings shall be as follows:

1 valve at ≤1080 psig
2 valves at ≤1090 psig
1 valve at ≤1100 psig
2 valves at ≤1240 psig (safety valves)

Bases:

1.2 REACTOR COOLANT SYSTEM

The reactor coolant system is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and the coolant system piping. The respective design pressures are 1250 psig at 575°F and 1148 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1148 = 1378 psig).

The safety valves are sized to keep the reactor coolant system pressure below 1375 psig with credit taken for the relief valves but no credit taken for the turbine bypass system. Credit is taken for the neutron flux scram, however. (See Supplement 2 to Proposed Change No. 14, November 12, 1973)

2.2 REACTOR COOLANT SYSTEM

The settings on the reactor high pressure scram, reactor coolant system relief and safety valves, have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition to preventing power operation above 1055 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients. (See FSAR Section 14.5 and Supplement 2 to Proposed Change No. 14, November 12, 1973.)

3.3 LIMITING CONDITIONS FOR OPERATION

2. The control rod drive housing support system shall be in place when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 10% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

   (a) if after withdrawal of at least twelve control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

   (b) if all rods except those that can not be moved with control rod drive pressure are fully inserted, no more than two rods may be moved.

4. Control rod patterns and the sequence of withdrawal or insertion shall be established such that:

   (a) when the reactor is critical and below 10% power the maximum calculated worth of any withdrawn increment of any in-sequence control rod which is not electically disarmed shall be less than 1.3% delta k.

4.3 SURVEILLANCE REQUIREMENTS

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

   (a) The Reactor Engineer shall verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

   (b) The Rod Worth Minimizer diagnostice test shall be performed.

   (c) Out-of-sequence control rods in each distinct RWM group shall be selected and the annunciation of the selection errors verified.

   (d) An out-of-sequence control rod shall be withdrawn no more than three notches and the rod block function verified.

4. The control rod pattern and sequence of withdrawal or insertion shall be verified to comply with Specification 3.3.B.4.

3.3 (cont'd)

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, can not occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 10% power the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. The control rod insertion and withdrawal sequences are established to assure that the maximum in sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 ΔK supercritical if they were to drop out of the core in the manner defined for the rod drop accident. The rod drop accident that is applicable to Vermont Yankee is discussed in reference (1). The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified delta k limit on in-sequence control rod or control rod segment worths. Each core reload will be analyzed to show conformance to the limiting parameters.

(1) Stein, R.C., Paone, C.J., Haun, J.M., "Rod Drop Accident Analysis for Large Boiling Water Reactors, Addendum 2 Exposed Cores," Supplement 2-NEDO 10527 January 1973.

3.5 LIMITING CONDITION FOR OPERATION

F. Automatic Depressurization System

1. Except as specified in Specification 3.5.F.2 below, the entire automatic depressurization relief system shall be operable at any time the reactor pressure is above 100 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the automatic depressurization subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible, provided that during such time the HPCI subsystem is operable.

4.5 SURVEILLANCE REQUIREMENT

F. Automatic Depressurization System

Surveillance of the automatic depressurization system shall be performed as follows:

1. During each operating cycle each relief valve shall be manually opened with the reactor at low pressure until the thermocouples downstream of the valve indicates fluid is flowing from the valve.

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

3.5 (cont'd)

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less that 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference Section 14.6.3.3.2 FSAR.

The containment cooling subsystem consists of two sets of 2 RHR service water pumps, 1 heat exchanger and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger and 1 RHR pump has sufficient capacity to perform the cooling function. Whenever one containment cooling subsystem becomes inoperable, the remaining subsystem will be tested daily.

D. Station Service Water and Alternate Cooling Tower Systems

The station service water subsystems and the alternate cooling tower system provide alternate heat sinks to dissipate residual heat after a shutdown or accident. Each station service water subsystem and the alternate cooling tower subsystem provides sufficient heat sink capacity to perform the required heat dissipation. The alternate cooling tower subsystem will provide the necessary heat sink in the event both station service water subsystems become incapacitated due to a loss of the Vernon Dam with subsequent loss of the Vernon Pond.

E. High Pressure Coolant Injection System

The high pressure coolant injection system (HPCIS) is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray cooling subsystems can protect the core.

The HPCIS meets this requirement without the use of outside power. For the pipe breaks for which the HPCIS is intended to function the core never uncovers and is continuously cooled; thus, no clad damage occurs and clad temperatures remain near normal throughout the transient. Reference Subsection 6.5.2.2 of the FSAR.

F. Automatic Depressurization System

The relief valves of the automatic depressurization system are a backup to the HPCIS. They enable the core spray cooling system or LPCIS to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCIS. Either of the two core spray cooling systems or LPCIS provide sufficient flow of coolant to prevent clad melting. All four relief valves are included in the automatic pressure relief system. Of these four, only two are required to provide sufficient capacity for the automatic depressurization system. (See VYNPS FSAR Vol. 4 Appendix B.) However, at least three valves are required by this section to provide an additional margin of redundancy. In addition specific recognition is made of the operability of the HPCI as an additional requirement.

LIMITING CONDITIONS FOR OPERATION

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

2. Both the sump and air sampling systems shall be operable during power operation. From and after the date that one of these systems is made or found inoperable for any reason, reactor operation is permissible only during succeeding seven days.

3. If these conditions cannot be met, initiate an orderly shutdown and the reactor shall be in the cold shutdown condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves shall be operable. The relief valves shall be operable, except that if one relief valve is inoperable, reactor power shall be immediately reduced to and maintained at or below 90% of rated power.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENTS

C. Coolant Leakage

Reactor coolant system leakage shall be checked and logged at least once per day.

D. Safety and Relief Valves

1. A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. Both valves shall be checked or replaced every two refueling outages. The lift point of the safety valves shall be set as specified in Specification 2.2.B.

2. A minimum of 1/2 of all relief valves shall be bench checked or replaced with a bench-checked valve each refueling outage. All four valves shall be checked or replaced every two refueling outages. The set pressures shall be as specified in Specification 2.2.B.

3.6 & 4.6 (cont'd)

When conductivity is in its proper normal range (approximately 10μmho/cm during reactor startup and 5μmho/cm during power operation), pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt, e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 5μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time when the conductivity exceed 5μmho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors", have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat