Entergy

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BVY 02-95

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 256
Administrative Changes to Position Titles and Miscellaneous Changes

Pursuant to 10CFR50.90, Vermont Yankee Nuclear Power Station (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed changes into the VY Technical Specifications (TS). The proposed changes consist of administrative changes to position titles as a result of Entergy's purchase of Vermont Yankee and miscellaneous changes. The proposed changes implement the use of generic position titles as provided by, or comparable to, ANSI N18.1-1971 or Regulatory Guide 1.8 in lieu of plant specific personnel titles. This change does neither eliminate nor compromise any of the qualifications, responsibilities or requirements for these positions, since the plant-specific personnel titles are identified in licensee controlled documents such as the Quality Assurance Manual and Technical Requirements Manual. In addition, a link between the generic titles and plant specific staff titles will be specified in the Technical Requirements Manual.

Additional proposed changes to the VY TS consist of editorial and administrative changes to the Table of Contents (TOC), Bases 3.5.G, 3.7.A and a minor change to Section 4.10.C.2 and Bases 4.10.C. The changes to the TOC are editorial as they are limited to correcting page numbers and titles to be consistent with the text and restructuring the format to reduce redundancy while ensuring completeness. The Bases 3.5.G and 3.7.A changes are editorial and necessary to improve clarity. VY is also proposing a change of diesel fuel specification ASTM D975-68 to ASTM D975-02 in Section 4.10.C.2 and Bases 4.10.C.

Attachment 1 to this letter contains supporting information for the proposed changes. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the marked-up version of the current Technical Specification pages showing the changes requested. Attachment 4 contains the retyped Technical Specification pages.

VY has reviewed the proposed Technical Specification changes in accordance with 10CFR50.92 and concludes that the proposed changes do not involve a significant hazards consideration.

The proposed changes will not increase the types and amounts of effluents that may be released off site, and therefore are eligible for categorical exclusion from the requirements for an

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environmental impact statement in accordance with 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued by June 30, 2003 for implementation within 60 days of its effective date.

If you have any questions concerning this transmittal, please contact Ronda Daflucas at (802) 258-4232.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi
Vice President, Operations

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Michael A. Balduzzi, who, being duly sworn, did state that he is Vice President, Operations of Vermont Yankee Nuclear Power Station, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Station, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2003

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS
Vermont Department of Public Service

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 256

Administrative Changes to Position Titles and Miscellaneous Changes

Supporting Information and Safety Assessment of Proposed Change

DESCRIPTION OF CHANGES

Vermont Yankee Nuclear Power Station (VY) proposes to revise the Technical Specifications (TS) Section 6 position titles to generic position titles as provided by, or comparable to, ANSI N18.1-1971 or Regulatory Guide 1.8 in lieu of plant specific personnel titles. Additionally, editorial revisions to the Table of Contents, removal of a redundant tab, administrative changes to Bases 3.5.G and 3.7.A and adoption of a more current revision of the diesel fuel specification in Surveillance 4.10.C.2 and Basis 4.10.C are proposed as described below.

Proposed Changes

On all pages of the Table of Contents, move page number from bottom center to bottom right justified position.

Current Page i, Table of Contents General. Delete the one-page General Table of Contents.

Proposed Page i, Table of Contents.
- Delete the text "(Continued)" since this page is now the first page of the Table of Contents.
- Insert "1.0 DEFINITIONS' which begins on page 1.
- Change 3.2.D. "Air Ejector Off-Gas System Isolation" to "Off-Gas System Isolation."
- Change 3.2.F. "Mechanical Vacuum Pump Isolation" to "Mechanical Vacuum Pump Isolation Instrumentation."
- Correct page number for Sections 3.2.G from 35 to 36.
- Correct page number for Sections 3.2.I, J and K from 36 to 37.

Proposed Page ii, Table of Contents (Continued).
- Change 3.4.C. "Liquid Poison Tank – Boron Concentration" to "Standby Liquid Control System Tank – Borated Solution."
- Correct page number for Section 3.6.C. from 119 to 118.
- Change 3.6.E. "Structural Integrity" to "Structural Integrity and Operability Testing."
- Correct page number for 3.6. Bases from 139 to 138.
- Change 3.7.B. "Standby Gas Treatment" to "Standby Gas Treatment System."
- Correct page number for 3.7.D. from 156 to 158.

Proposed Page iii, Table of Contents (Continued).
- Remove "4.9" listed under "Surveillance" since 3.9 and 4.9 have been deleted.
- Change 3.11.A. "Average Planar LHGR" to "Average Planar Linear Heat Generation Rate (APLHGR)."
- Change 3.11.B. "LHGR" to "Linear Heat Generation Rate (LHGR)."
- Change 3.11.C. "MCPR" to "Minimum Critical Power Ratio (MCPR)."

Page 111b, Bases 3.5.G. Change the word "rector" to "reactor."

Page 163, Bases 3.7.A. Change "1000 psig" to "normal operating pressure."

Page 218, Section 4.10.C.2. Change ASTM D975-68 to ASTM D975-02.

Page 223, Bases 4.10.C. Change ASTM D975-68 to ASTM D975-02 in two locations in paragraph.

Page 255, Section 6.1.A and B. Change "Plant Manager" to "plant manager."

Page 255, Section 6.1.C. Change "Shift Supervisor" to "shift supervisor" in three (3) locations throughout paragraph.

Page 255, Section 6.2.A.1. Add "The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Technical Requirements Manual."

Page 255, Section 6.2.A.2. Change "Plant Manager" to "plant manager."

Page 255, Section 6.2.A.3. Change "corporate executive with direct responsibility for the plant" to "site vice president."

Page 256, Section 6.2.A.4. Remove extra tab at the beginning of the paragraph.

Page 256, Section 6.2.B.7. Change "operations superintendent" to "operations manager" and "assistant operations superintendent" to "assistant operations manager."

Page 256, Section 6.2.B.8. Change "the Shift Engineer" to "a shift engineer." Change "Shift Supervisor (SS)" to "shift supervisor."

Page 257, Section 6.2.C. Change "Shift Engineer" to "shift engineer"

Page 258, Section 6.5.A.3. Change "Radiation Protection Manager" to "radiation protection manager."

Page 258, Section 6.5.B. Change "Shift Supervisor" to "shift supervisor" and change "Radiation Protection Manager" to "radiation protection manager."

Page 264, Section 6.7.B.1.b. Change "Plant Manager" to "plant manager."

REASON/BASIS FOR CHANGES

The current VY Technical Specifications contain titles for positions that existed in the Vermont Yankee Nuclear Power Station organization prior to the transfer of ownership to Entergy. The change to the use of generic position titles as provided by, or comparable to, ANSI N18.1-1971 or Regulatory Guide 1.8 is administrative and is being proposed as a result of an attempt to provide consistency between Vermont Yankee and the rest of Entergy Nuclear Operations, Inc. The change does not eliminate any of the qualifications, responsibilities or requirements for these positions, since the plant-specific personnel titles will be identified in licensee controlled documents such as the Technical Requirements Manual and Quality Assurance Manual. In addition, a link between the generic titles and plant specific staff titles will be specified in the Technical Requirements Manual. Specification 6.2.A.1 has been modified to include a sentence to provide that link.

The proposal includes deletion of the one-page General Table of Contents since the information is very high level, incomplete and otherwise redundant to the revised Table of Contents. The contents of the General Table of Contents have been incorporated into the Table of Contents for improvement. Page numbering and titles of sections have been corrected to be consistent with the titles throughout the text.

The change of the word "rector" to "reactor" on page 111b, Bases 3.5.G, is correcting a typographical error and is an administrative change.

The change of the text "1000 psig" to "normal operating pressure" on page 163, Bases 3.7.A, is proposed for clarity and correctness of the basis for the pressure suppression chamber water volume.

The change from ASTM D975-68 to ASTM D975-02 is reflected on page 218, Section 4.10.C.2. and page 223, Bases 4.10.C. The Standard Specification for Diesel Fuel Oils is issued under the fixed designation D975. The proposal is to adopt the current 2002 revision of ASTM D975 which provides more information and guidance than the 1968 standard.

The deletion of an extra tab at Page 256, Section 6.2.A.4 is correcting a typographical error and is an administrative change.

SAFETY ASSESSMENT

Revision of the position titles to generic titles to achieve consistency with titles used at other Entergy Nuclear Operations, Inc. plants is an administrative change. The proposed changes implement the use of generic position titles as provided by, or comparable to, ANSI N18.1-1971 or Regulatory Guide 1.8 in lieu of plant specific personnel titles. This change does not eliminate any of the qualifications, responsibilities or requirements for these positions, since the link between the generic and plant-specific personnel titles will be specified in the Technical Requirements Manual. The change to the use of generic titles is an administrative change, does not affect any system operation, function or safety analyses, and therefore has no impact on plant safety.

Revision of the Table of Contents and correction of typographical errors are administrative changes and have no impact on plant safety.

Revision of Bases 3.7.A clarifies that the pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released following a postulated rupture at normal operating pressure. Changing the text from "1000 psig" to "normal operating pressure" provides clarity and does not affect any system operation, function or safety analyses, and therefore has no impact on plant safety.

The proposed change to use a more current revision of the diesel fuel specification, specifically ASTM D975-02 rather than ASTM D975-68, has no impact on plant systems, structures or components and therefore has no impact on plant safety.

<u>Summary</u>

In summary, the proposed change of specific position titles to generic position titles, corrections to page numbers and section titles in the Table of Contents, and miscellaneous editorial changes are administrative in nature and do not affect any system operation, function or safety analyses, and therefore have no impact on plant safety. The proposed change to use diesel fuel specification ASTM D975-02 also does not affect any system operation, function or safety analyses, and therefore has no impact on plant safety.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 256

Administrative Changes to Position Titles and Miscellaneous Changes

Determination of No Significant Hazards Consideration

Determination of No Significant Hazards Consideration

Description of Amendment Request:

Vermont Yankee (VY) has determined that the proposed changes to Technical Specifications (TS), which change the position titles to generic position titles to provide consistency between Vermont Yankee and the rest of Entergy Nuclear Operations, Inc., do not involve a Significant Hazards Consideration. The Bases 3.7.A text change of "1000 psig" to "normal operating pressure" does not involve a Significant Hazards Consideration. The proposed change to use a more current revision of the diesel fuel specification, specifically ASTM D975-02, does not involve a Significant Hazards Consideration. The revisions to the Table of Contents, removal of a redundant tab and correction of typographical errors are editorial and do not involve a Significant Hazards Consideration. In support of this determination, each of the three (3) standards set forth in 10CFR50.92 is provided below.

Basis for No Significant Hazards Determination:

Pursuant to 10CFR50.92, VY has reviewed the proposed changes and concluded that they do not involve a significant hazards consideration since the proposed changes satisfy the criteria in 10CFR50.92(c).

1. The operation of the Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   VY has determined that the probability of occurrence of a previously evaluated accident is not increased because the proposed changes do not impact any accident initiating conditions. The proposed changes will have no significant impact on any safety related structures, systems or components. Additionally, the administrative changes do not affect any system operation or function.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   VY has determined that the proposed changes do not involve any physical alteration of plant equipment and do not change the method by which any safety-related system performs its function. No new or different types of equipment will be installed. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

   Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

VY has determined that the proposed changes do not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

The proposed changes do not impact design margins of any system to perform its intended safety functions. There is no physical or operational change being made which would alter the sequence of events, plant response, or margins in existing safety analyses. The proposed changes result in no impact on analyzed accident event precursors or effects. These proposed changes do not alter the physical design of the plant. There is no change in methods of operation.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 256

Administrative Changes to Position Titles and Miscellaneous Changes

Marked-up Version of the Current Technical Specifications

VYNPS

TABLE OF CONTENTS

GENERAL

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BASES: 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.

In accordance with specification 3.5.G.2, if the RCIC System is inoperable and the HPCI System is verified to be operable, the RCIC System must be restored to operable status within 14 days during reactor power operation. In this condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident. Operability of HPCI is therefore verified immediately when the RCIC System is inoperable during ~~rector~~ reactor power operation. This may be performed as an administrative check, by examining logs or other information, to determine if HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform surveillances needed to demonstrate the operability of the HPCI System. If the operability of the HPCI System cannot be verified, however, Specification 3.5.G.3 requires that an orderly shutdown be initiated and reactor pressure reduced to ≤ 150 psig within 24 hours. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the reactor water level. therefore, a limited time (14 days) is allowed to restore the inoperable RCIC System to operable status.

H. Minimum Core and Containment Cooling System Availability

The core cooling and 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 the core and containment cooling function is maintained with any combination of allowed inoperable components.

Operability of low pressure ECCS injection/spray subsystems is required during cold shutdown and refueling conditions to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of inadvertent draindown of the vessel. 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 in cold shutdown or refueling and there are no operations with a potential for draining the reactor vessel (OPDRV). However, if OPDRVs are in progress with irradiated fuel in the reactor vessel, operability of low pressure ECCS injection/spray subsystems is required to ensure capability to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. In this condition, at least 300,000 gallons of makeup water must be available to assure core flooding capability. In addition, only one diesel generator associated with one of the ECCS injection/spray subsystems is required to be operable in this condition since, upon loss of normal power supply, one ECCS subsystem is sufficient to meet this function.

BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than to those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from ~~1000 psig~~ normal operating pressure.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable internal design pressure for the pressure suppression chamber. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, the calculated peak accident containment pressure is approximately 44 psig, which is below the ASME design pressure of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humbolt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Internal design pressure is 62 psig.

3.10 LIMITING CONDITIONS FOR OPERATION

4. <u>480 V Uninterruptible Power Systems</u>

   From and after the date that one Uninterruptible Power System or its associated Motor Control Center are made or found to be inoperable for any reason, the requirements of Specification 3.5.A.4 shall be satisfied.

5. <u>RPS Power Protection</u>

   From and after the date that one of the two redundant RPS power protection panels on an in-service RPS MG set or alternate power supply is made or found to be inoperable, the associated RPS MG set or alternate supply will be taken out of service until the panel is restored to operable status.

C. <u>Diesel Fuel</u>

There shall be a minimum of 36,000 usable gallons of diesel fuel in the diesel fuel oil storage tank.

4.10 SURVEILLANCE REQUIREMENTS

4. <u>480 V Uninterruptible Power Systems</u>

   When it is determined that one Uninterruptible Power System or its associated Motor Control Center is inoperable, the requirements of Specification 4.5.A.4 shall be satisfied.

C. <u>Diesel Fuel</u>

1. The quantity of diesel generator fuel shall be logged weekly and after each operation of the unit.

2. Once a month a sample of diesel fuel shall be taken and checked for quality. The quality shall be within the applicable limits specified on Table I of ASTM D975~~-68~~ and logged.
   -02

BASES: 4.10 (Cont'd)

for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The Service Discharge Test (4.10.A.2.c) is a test of the batteries ability to satisfy the design requirements of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile (battery duty cycle).

Assurance that the diesels will meet their intended function is obtained by periodic surveillance testing and the results obtained from the pump and valve testing performed in accordance with the requirements of ASME Section XI and Specification 4.6.E. Specification 4.10.B.1.a provides an allowance to avoid unnecessary testing of the operable emergency diesel generator (EDG). If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.5.H.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, Specification 4.10.B.1.a is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of Surveillance Requirement (SR) 4.10.B.1.b suffices to provide assurance of continued operability of that EDG.

In the event the inoperable EDG is restored to operable status prior to completing either SR 4.10.B.1.a or SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1 or SR 4.10.B.3.b.2.

According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG.

Verification of operability of an off-site power source and Low Pressure Core and Containment Cooling Systems within one hour and once per eight hours thereafter as required by 4.10.B.3.b.1 may be performed as an administrative check by examining logs and other information to determine that required equipment is available and not out of service for maintenance or other reasons. It does not require performing the surveillance needed to demonstrate the operability of the equipment.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975~~-68~~ -02. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975~~-68~~ -02.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during absences.

B. The Plant Manager or designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

C. The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in plant startup or normal operation, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in cold shutdown or refueling with fuel in the reactor, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 ORGANIZATION

A. Onsite and Offsite Organizations

Organizations shall be established for unit operation and corporate management. These organizations shall include the positions for activities affecting safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Vermont Yankee Operational Quality Assurance Manual. Insert A

2. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those on-site activities necessary for safe operation and maintenance of the plant.

site vice president

3. The corporate executive with direct responsibility for the plant shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

Insert A

The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Technical Requirements Manual.

6.2 ORGANIZATION (Cont'd)

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate on-site manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Unit Staff

The unit staff organization shall include the following:

1. A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned during Plant Startup and Normal Operation.

2. At least one licensed Reactor Operator (RO) or one licensed Senior Reactor Operator (SRO) shall be present in the control room when fuel is in the reactor.

3. When the unit is in Plant Startup or Normal Operation, at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, shall be present in the control room.

4. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.B.1 and 6.2.B.8 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5. An individual qualified in radiation protection procedures shall be present on-site when there is fuel in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).

7. The operations ~~superintendent~~ manager or an assistant operations ~~superintendent~~ manager shall hold an SRO license.

8. While the unit is in Plant Startup or Normal Operation, ~~the~~ a shift engineer shall provide advisory technical support to the shift supervisor ~~(SS)~~.

6.2 ORGANIZATION (Cont'd)

C. Unit Staff Qualifications

Each member of the unit staff shall meet or exceed the minimum qualifications of the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants," except for the radiation protection manager who shall meet the qualifications of Regulatory Guide 1.8, Revision 1 (September 1975) and the Shift Engineer, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.3 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Applies to administrative action to be followed in the event a safety limit is exceeded.

If a safety limit is exceeded, the reactor shall be shutdown immediately.

6.4 PROCEDURES

Written procedures shall be established, implemented, and maintained covering the following activities:

A. Normal startup, operation and shutdown of systems and components of the facility.

B. Refueling operations.

C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected Primary System leaks and abnormal reactivity changes.

D. Emergency conditions involving potential or actual release of radioactivity.

E. Preventive and corrective maintenance operations which could have an effect on the safety of the reactor.

F. Surveillance and testing requirements.

G. Fire protection program implementation.

H. Process Control Program in-plant implementation.

I. Off-Site Dose Calculation Manual implementation.

6.5 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraphs 20.1601(a) and 20.1601(b) of 10 CFR 20:

A. Paragraph 20.1601, "Control of Access to High Radiation Areas. In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a), each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm, but less than 1000 mrem/hr at 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Radiation Protection personnel qualified in radiation protection procedures (e.g., radiation protection technicians) may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

3. A Radiation Protection individual qualified in radiation protection procedures (e.g., radiation protection technicians) with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and who will perform direct or remote (such as closed circuit TV cameras) periodic radiation surveillance at the frequency specified in the RWP. The surveillance frequency will be established by the Radiation Protection Manager.

B. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr at 30 cm, but less than 500 rad/hr at 1 meter. In addition, locked or continuously guarded entryways shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Radiation Protection Manager.

6.6 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4.

A. Occupational Radiation Exposure Report

An annual report covering the previous calendar year shall be submitted prior to April 30 of each year. The annual report shall implemented. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements.

2. System leakage inspections, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are: (1) Residual Heat Removal, (2) Core Spray, (3) Reactor Water Cleanup, (4) HPCI, (5) RCIC, and (6) Sampling Systems.

B. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

An Off-Site Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents for the purpose of demonstrating compliance with 10 CFR 50, Appendix I, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Radioactive Effluent Release Report and the Annual Radiological Environmental Operating Report required by Specification 6.6.D and Specification 6.6.E, respectively.

1. Licensee initiated changes to the ODCM:

   a. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

      i. Sufficient information to support the change together with appropriate analyses or evaluations justifying the change(s) and

      ii. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, and do not adversely impact the accuracy or reliability of effluent dose or setpoint calculations.

   b. Shall become effective upon review by PORC and approved by the plant manager.

   c. Shall be submitted to the Commission in the form of a legible copy of the affected pages of the ODCM as a part of or concurrent with the Radioactive Effluent Release

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 256

Administrative Changes to Position Titles and Miscellaneous Changes

Retyped Technical Specification Pages

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BASES: 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.

In accordance with specification 3.5.G.2, if the RCIC System is inoperable and the HPCI System is verified to be operable, the RCIC System must be restored to operable status within 14 days during reactor power operation. In this condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident. Operability of HPCI is therefore verified immediately when the RCIC System is inoperable during reactor power operation. This may be performed as an administrative check, by examining logs or other information, to determine if HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform surveillances needed to demonstrate the operability of the HPCI System. If the operability of the HPCI System cannot be verified, however, Specification 3.5.G.3 requires that an orderly shutdown be initiated and reactor pressure reduced to $\leq$ 150 psig within 24 hours. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the reactor water level. therefore, a limited time (14 days) is allowed to restore the inoperable RCIC System to operable status.

H. Minimum Core and Containment Cooling System Availability

The core cooling and 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 the core and containment cooling function is maintained with any combination of allowed inoperable components.

Operability of low pressure ECCS injection/spray subsystems is required during cold shutdown and refueling conditions to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of inadvertent draindown of the vessel. 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 in cold shutdown or refueling and there are no operations with a potential for draining the reactor vessel (OPDRV). However, if OPDRVs are in progress with irradiated fuel in the reactor vessel, operability of low pressure ECCS injection/spray subsystems is required to ensure capability to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. In this condition, at least 300,000 gallons of makeup water must be available to assure core flooding capability. In addition, only one diesel generator associated with one of the ECCS injection/spray subsystems is required to be operable in this condition since, upon loss of normal power supply, one ECCS subsystem is sufficient to meet this function.

BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than to those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from normal operating pressure.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable internal design pressure for the pressure suppression chamber. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, the calculated peak accident containment pressure is approximately 44 psig, which is below the ASME design pressure of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humbolt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Internal design pressure is 62 psig.

3.10 LIMITING CONDITIONS FOR OPERATION

4. 480 V Uninterruptible Power Systems

From and after the date that one Uninterruptible Power System or its associated Motor Control Center are made or found to be inoperable for any reason, the requirements of Specification 3.5.A.4 shall be satisfied.

5. RPS Power Protection

From and after the date that one of the two redundant RPS power protection panels on an in-service RPS MG set or alternate power supply is made or found to be inoperable, the associated RPS MG set or alternate supply will be taken out of service until the panel is restored to operable status.

C. Diesel Fuel

There shall be a minimum of 36,000 usable gallons of diesel fuel in the diesel fuel oil storage tank.

4.10 SURVEILLANCE REQUIREMENTS

4. 480 V Uninterruptible Power Systems

When it is determined that one Uninterruptible Power System or its associated Motor Control Center is inoperable, the requirements of Specification 4.5.A.4 shall be satisfied.

C. Diesel Fuel

1. The quantity of diesel generator fuel shall be logged weekly and after each operation of the unit.

2. Once a month a sample of diesel fuel shall be taken and checked for quality. The quality shall be within the applicable limits specified on Table I of ASTM D975-02 and logged.

BASES: 4.10 (Cont'd)

for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The Service Discharge Test (4.10.A.2.c) is a test of the batteries ability to satisfy the design requirements of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile (battery duty cycle).

Assurance that the diesels will meet their intended function is obtained by periodic surveillance testing and the results obtained from the pump and valve testing performed in accordance with the requirements of ASME Section XI and Specification 4.6.E. Specification 4.10.B.1.a provides an allowance to avoid unnecessary testing of the operable emergency diesel generator (EDG). If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.5.H.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, Specification 4.10.B.1.a is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of Surveillance Requirement (SR) 4.10.B.1.b suffices to provide assurance of continued operability of that EDG.

In the event the inoperable EDG is restored to operable status prior to completing either SR 4.10.B.1.a or SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1 or SR 4.10.B.3.b.2.

According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG.

Verification of operability of an off-site power source and Low Pressure Core and Containment Cooling Systems within one hour and once per eight hours thereafter as required by 4.10.B.3.b.1 may be performed as an administrative check by examining logs and other information to determine that required equipment is available and not out of service for maintenance or other reasons. It does not require performing the surveillance needed to demonstrate the operability of the equipment.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-02. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-02.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during absences.

B. The plant manager or designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

C. The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in plant startup or normal operation, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in cold shutdown or refueling with fuel in the reactor, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 ORGANIZATION

A. Onsite and Offsite Organizations

Organizations shall be established for unit operation and corporate management. These organizations shall include the positions for activities affecting safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Vermont Yankee Operational Quality Assurance Manual. The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Technical Requirements Manual.

2. The plant manager shall be responsible for overall unit safe operation and shall have control over those on-site activities necessary for safe operation and maintenance of the plant.

3. The site vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

6.2 ORGANIZATION (Cont'd)

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate on-site manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Unit Staff

The unit staff organization shall include the following:

1. A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned during Plant Startup and Normal Operation.

2. At least one licensed Reactor Operator (RO) or one licensed Senior Reactor Operator (SRO) shall be present in the control room when fuel is in the reactor.

3. When the unit is in Plant Startup or Normal Operation, at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, shall be present in the control room.

4. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.B.1 and 6.2.B.8 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5. An individual qualified in radiation protection procedures shall be present on-site when there is fuel in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).

7. The operations manager or an assistant operations manager shall hold an SRO license.

8. While the unit is in Plant Startup or Normal Operation, a shift engineer shall provide advisory technical support to the shift supervisor.

6.2 ORGANIZATION (Cont'd)

C. Unit Staff Qualifications

Each member of the unit staff shall meet or exceed the minimum qualifications of the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants," except for the radiation protection manager who shall meet the qualifications of Regulatory Guide 1.8, Revision 1 (September 1975) and the shift engineer, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.3 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Applies to administrative action to be followed in the event a safety limit is exceeded.

If a safety limit is exceeded, the reactor shall be shutdown immediately.

6.4 PROCEDURES

Written procedures shall be established, implemented, and maintained covering the following activities:

A. Normal startup, operation and shutdown of systems and components of the facility.

B. Refueling operations.

C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected Primary System leaks and abnormal reactivity changes.

D. Emergency conditions involving potential or actual release of radioactivity.

E. Preventive and corrective maintenance operations which could have an effect on the safety of the reactor.

F. Surveillance and testing requirements.

G. Fire protection program implementation.

H. Process Control Program in-plant implementation.

I. Off-Site Dose Calculation Manual implementation.

6.5 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraphs 20.1601(a) and 20.1601(b) of 10 CFR 20:

A. Paragraph 20.1601, "Control of Access to High Radiation Areas. In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a), each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm, but less than 1000 mrem/hr at 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Radiation Protection personnel qualified in radiation protection procedures (e.g., radiation protection technicians) may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

3. A Radiation Protection individual qualified in radiation protection procedures (e.g., radiation protection technicians) with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and who will perform direct or remote (such as closed circuit TV cameras) periodic radiation surveillance at the frequency specified in the RWP. The surveillance frequency will be established by the radiation protection manager.

B. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr at 30 cm, but less than 500 rad/hr at 1 meter. In addition, locked or continuously guarded entryways shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or the radiation protection manager.

6.6 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4.

A. Occupational Radiation Exposure Report

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1. Provisions establishing preventive maintenance and periodic visual inspection requirements.

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1. Licensee initiated changes to the ODCM:

    a. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

        i. Sufficient information to support the change together with appropriate analyses or evaluations justifying the change(s) and

        ii. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, and do not adversely impact the accuracy or reliability of effluent dose or setpoint calculations.

    b. Shall become effective upon review by PORC and approved by the plant manager.

    c. Shall be submitted to the Commission in the form of a legible copy of the affected pages of the ODCM as a part of or concurrent with the Radioactive Effluent Release