FEB - 7 1974

Docket Nos. 50-254 and 50-265

Commonwealth Edison Company
ATTN: Mr. Byron Lee, Jr.
Vice President
Post Office Box 767
Chicago, Illinois 60690

Change No. 12
Licenses Nos. DPR-29
and DPR-30

Gentlemen:

In response to our letter dated August 9, 1972, regarding drywell-suppression chamber vacuum breakers, you submitted Quad-Cities Special Report No. 4 by letter dated October 23, 1972. Our letter, dated November 8, 1972, related to meetings held with Commonwealth Edison Company (CECo) on October 5 and 6, 1972, in which CECo advised of corrective measures taken to assure vacuum breaker closure. This letter also stipulated additional surveillance required until all required corrective measures were completed. As a result of our review of Quad-Cities Special Report No. 4, Change No. 4 to Quad-Cities Technical Specifications, issued December 14, 1972, included revised Limiting Condition for Operation and Surveillance Requirements relating to the drywell-suppression chamber vacuum breakers.

Subsequent discussions with CECo personnel and additional evaluation by the Regulatory staff have identified the need for additional changes to Sections 3.7.A and 4.7.A of the Technical Specifications regarding containment vacuum breakers. These changes are summarized below:

1. Pressure Suppression Chamber - Reactor Building Vacuum Breakers:

   Addition of Limiting Condition for Operation (LCO) and Surveillance Requirement (SR) for all four valves in this system.

2. Drywell - Pressure Suppression Chamber Vacuum Breakers:

   a. LCO for allowable bypass opening of the vacuum breakers is changed from 1/8" opening at the bottom of the disk to 1/16" at all points along the seating surface of the disk.

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The position indication and alarm system LCOs are changed to be consistent with this requirement. The change in the description of the allowable vacuum breaker bypass opening is made for clarification and does not change the required performance of the vacuum breaker.

b. The LCO for Differential Pressure decay rate during reactor operation is revised to be consistent with the allowable opening rather than reference to a 1-inch orifice.

c. Surveillance Requirements include inspection and calibration of vacuum breaker systems and a revision of the pressure decay rate tests.

d. The bases are revised to be consistent with these changes.

The addition of and changes to the Limiting Condition for Operation and Surveillance Requirements to the Technical Specifications enhance the safety of plant operation. These changes have been discussed with CECo.

We have concluded that the above changes in Sections 3.7.A and 4.7.A of the Quad-Cities Technical Specifications do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner accorded by these changes.

Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Licenses Nos. DPR-29 and DPR-30 are hereby changed by replacing the present pages 148 through 150, 166, 167, 171, 172, and 173 with the enclosed revised pages.

These changes to the Quad-Cities Technical Specifications shall become effective following the next refueling outage of each unit when the necessary equipment modifications are made but no later than the end of March 1974 for Unit 1 and the end of October 1974 for Unit 2.

A copy of our Safety Evaluation relating to the Technical Specifications change is also enclosed.

Sincerely,

/s/

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

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SURNAME➤						

Enclosures and cc: See next page

Enclosures:
1. Revised pages 148 through 150, 166, 167, 171, 172, and 173
2. Safety Evaluation

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UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NOS. 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY

INTRODUCTION

In a letter to Commonwealth Edison Company (CECo) from D. J. Skovholt dated August 9, 1972, we advised that additional technical specifications for the suppression chamber - drywell vacuum breakers were required basically consisting of (1) additional testing, (2) means to alert the operator if a vacuum breaker is open, (3) analysis of results of bypass leakage in the event of a LOCA, and (4) reliability of operability of vacuum breakers. On August 24, 1972, during an inspection of Quad-Cities Unit 2 vacuum breakers, 6 of the 12 were found open. CECo submitted on September 19, 1972, plans to modify the vacuum breakers such that additional closure force would be exerted on the vacuum breaker to hold the disk against its seal surface. Modifications to the balancing level arm to provide the necessary closing force were made on Unit 2 prior to startup of that unit following the inspection. A change to the vacuum breaker Technical Specifications was made on December 14, 1972, however, new information developed since that time requires additional changes in the Technical Specifications.

EVALUATION

In the event of a LOCA, the vacuum breakers are required to be closed sufficiently to limit steam bypass from the drywell to suppression chamber void above the water level and to open when the suppression chamber pressure exceeds the drywell pressure by no more than 0.5 psi (Reference: Section 5.2, FSAR). The reactor building to suppression chamber vacuum breakers are required to open if the reactor building pressure exceeds the suppression chamber pressure by no more than 0.5 psi. Increased surveillance requirements were imposed by our letter dated November 8, 1972, to verify that the vacuum breakers will operate as intended in the event they are required to function.

Due to the time required to make the required changes to the containment vacuum breaker systems, including design, material procurement and installation, the Commission issued a letter to CECo, dated November 8, 1972, advising that increased surveillance should be performed until the proposed changes were installed and operational.

On October 23, 1972, CECo submitted Quad-Cities Special Report No. 4, "Drywell to Suppression Chamber Vacuum Breakers Modifications", dated October 20, 1972, in response to our August 9, 1972 letter. Evaluation of this report resulted in Change No. 4 to Sections 3.7.A.4 and 4.7.A.4 of the Technical Specifications of Quad-Cities Units 1 and 2, along with other changes, on December 14, 1972. The evaluation also led to our issuing a letter dated January 12, 1973, to all licensees with BWR Mark I Containments requesting certain inspections, information, analyses, and proposed technical specifications relating to the drywell to suppression chamber vacuum breakers. Review of responses received together with discussions with licensees, including CECo, resulted in additional changes required in the technical specifications. A summary of the required changes are shown below:

1. The reactor building to suppression chamber air-operated and self-actuating vacuum breakers are required to open with no more than a differential pressure of 0.5 psi. Additional surveillance requirements are specified to verify operability. The self-actuating vacuum breakers were designed and installed in such a manner that accessibility for surveillance tests is not possible, resulting in CECo considering changes to these vacuum breakers. CECo is presently performing a study to determine the extent of changes needed to meet the surveillance requirements and will advise us of their proposed changes in about one month.

2. An alarm system to alert the operator if a vacuum breaker is open more than allowable is being installed to meet single failure criteria. The position switches have been replaced to provide a signal if the vacuum breaker is open more than the equivalent of 1/16 inch at all points along the seal surface of the disk. The analysis for this allowable opening is shown in Special Report No. 4.

3. Differential pressure decay rate tests, in addition to vacuum breaker operability tests, have been included to verify that total bypass area between the drywell and suppression chamber does not exceed calculated allowable with a factor of 4 margin.

4. The maximum allowable inoperative and locked closed vacuum breaker is 25% of the total. This number is based on Bodega Bay tests performed by General Electric Company and provides adequate margin as shown in the FSAR.

The changes being made to the containment vacuum breaker systems and to the applicable sections of the technical specifications have been reviewed. It is concluded that when the changes to the vacuum breaker systems are completed and the technical specifications are in force there is reasonable assurance that the systems will function as intended by the SAR and will result in no significant hazard to the safety of the public.

The changes required to meet the revised technical specifications result in time needed to design, procure materials and equipment, and install equipment. In recognition of the time needed to make the changes and the additional surveillance imposed by the Commission letter dated November 8, 1972, the technical specification changes will not become effective until after the forthcoming refueling outage for each unit.

CONCLUSION

Based on the above, we have concluded that this action does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered.

John I. Riesland
Operating Reactors Branch #2
Directorate of Licensing

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

Date: February 7, 1974

3.7 LIMITING CONDITION FOR OPERATION	4.7 SURVEILLANCE REQUIREMENT
	(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.
3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers
a. Except as specified in Specification 3.7.A.3.b below, two pressure suppression chamber - reactor building vacuum breakers in each line shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber - reactor building air operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk. (12)	a. The pressure suppression chamber - reactor building vacuum breakers and associated instrumentation, including setpoint shall be checked for proper operation every three months.
b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the procedure does not violate primary containment integrity.	b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements. (12)

3.7 LIMITING CONDITION FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
4. Pressure Suppression Chamber - Drywell Vaccum Breakers	4. Pressure Suppression Chamber - Drywell Vacuum Breakers
4 a. When primary containment is required, all pressure suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.4.b, c and d., below, pressure suppression chamber - drywell vacuum breakers shall be considered operable if:	a. Periodic Operability Tests
	(1) Once each month each pressure suppress chamber - drywell vacuum breaker shall be exercised. Operability of valves, position switches, and position indicators and alarms shall be veri
	(2) A drywell to suppression chamber differential pressure decay rate test shall be conducted at least every 3 months.
12 (1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding the equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.	12 b. During each refueling outage:
	(1) The pressure suppression chamber - drywell vacuum breakers shall be teste to determine the force required to ope each valve from fully closed to fully open.
(2) The valve can be closed by gravity when released after being opened by remote or manual means, to within the equivalent of 1/16" at all points along the seal surface of the disk.	(2) Vacuum breakers position indication and alarm systems shall be calibrat and functionally tested.
(3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 1/16" at all points along the seal surface of the disk.	(3) At least 25% of the vacuum breakers shall be inspected such that all vacuum breakers shall have been inspected following every fourth refueling outage. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected.

7 LIMITING CONDITION FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
b. Any drywell-suppression chamber vacuum breaker may be non-fully closed as indicated by the position indication and alarm systems provided that drywell to suppression chamber differential pressure decay rate is demonstrated to be not greater than 25% of the differential pressure decay rate for all vacuum breakers open the equivalent of 1/16" at all points along the seal surface of the disk.	(4) A drywell to suppression chamber leak test shall demonstrate that with initial differential pressure of not less than 1.0 psi, the differential pressure decay rate does not exceed the rate which would occur through a 1-inch orifice without the addition of air or nitrogen.
12 c. Reactor operation may continue provided that no more than one quarter of the number of pressure suppression chamber - drywell vacuum breakers are determined to be inoperable provided that they are secured or known to be in the closed position.	12
d. If failure occurs in one of the two position alarm systems for one or more vacuum breakers, reactor operation may continue provided that a differential pressure decay rate test is initiated immediately and performed every 15 days thereafter until the failure is corrected. The test shall meet the requirements of Specification 3.7.A.4.b.	
5. Oxygen Concentration	5. Oxygen Concentration
a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.5.b.	The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the structural integrity of the containment is maintained.

The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% flow pipes each containing two vacuum relief breakers. Operation of either flow pipe will maintain the pressure differential less than 2 psig, the external design pressure of the primary containment. Redundancy of lines justifies reactor operation with one valve out of service for repairs for a period of seven days.

The capacity of the pressure suppression chamber - drywell vacuum breakers are designed to limit the pressure differential between the suppression chamber and drywell to not greater than 0.5 psi during post-accident drywell cooling. They are sized on the basis of the Bodega Bay pressure suppression system test.

12 Based on these tests, design flow from the suppression chamber to the drywell can be obtained with 25% of the vacuum breakers closed without exceeding the 0.5 psi differential pressure limit.

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16" at all points along the seal surface of the disk (see Quad-Cities Special Report No. 4).

12 Reactor operations is not permitted if differential pressure decay rate is demonstrated to exceed 25% of allowable, thus providing a margin of safety for the primary containment in the event of a small break in the primary system.

Each drywell-suppression chamber vacuum breaker is fitted with redundant pairs of position switches which provide signals of disk position to panel mounted indicators and annunicate an alarm in the control room if the disk is open more than allowabl The alarm systems meet the intent of IEEE 279 standards. The quality of the alarm system justifies continued reactor operation for 15 days between differential pressure decay rate tests if one alarm system is inoperable for one or more vacuum breakers.

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of Zirconium in the core are such that the occurrence of a very limited (a per cent or so) reaction of the Zirconium and steam during a loss of coolant accident would lead to the liberation of sufficient hydrogen to result in a flammable concentration in the containment. Subsequent ignition of the hydrogen, if it is present in sufficient quantities to result in excessively rapid recombination, could result in a loss of containment integrity.

The 5% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems did not sufficiently cool the core.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure.

The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen

intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods. 3

The data reduction methods of ANSI N45.4-1972 will be applied for integrated leak rate tests.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

The results of the loss of coolant accident analysis referenced in Section 5.2.4.3 of the SAR indicate that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves 4

due to holdup in the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out of service time will be kept to a practical minimum.

Surveillance of the reactor building - pressure suppression chamber vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgment 12

Pressure suppression chamber - drywell vacuum breakers monthly operability tests are performed to check capability of the disks to open and close and to verify that the position indication and alarm circuits function properly. The disk open during accident conditions and during transient additions of energy through relief valves. This periodic operation of the disks and the quality of equipment justify the frequency of operability tests of this equipment.

Following each quarterly operability test, a differential pressure decay rate test is performed to verify leakage from the drywell to the suppression chamber is within specified limits.

Measurement of force to open, calibration of position switches, inspection of equipment and functional testing are performed during each refueling outage. This frequency 12 is based on equipment quality, experience and judgment. Also a more stringent differential pressure decay rate test is performed during refueling outages than is performed monthly. This test is performed to verify that total leakage paths between the drywell and suppression chamber are not in excess of the equivalent to a 1-inch orifice.

This small leakage path is only a small fraction of the allowable, thus integrity of the containment system is assured prior to startup following each refueling outage (see Quad-Cities Special Report No. 4).

When a suppression chamber-drywell vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panel are designed to function as follows:

Full Closed (Closed to ≤1/16" open)	2 Green - On 2 Red - Off
Intermediate Position (≥1/16" open to < full open)	2 Green - On 2 Red - On
Full Open	2 Green - Off 2 Red - On

The remote test panel consists of a push button to actuate the air cylinder for testing, two red lights, and two green lights for each of the twelve valves. The two switches controlling the red lights are adjusted to provide indication and alarm if a disk opening occurs that is equivalent to one-sixteenth of an inch (1/16") at all points around the circumference of the valve disk. The physical characteristics of the valve 12 and the positioning of the limit switches permits one setting of the limit switches to satisfy the criteria. The two switches controlling the green lights are adjusted to provide indication of the disk very near the full open position. The control room alarm circuits for each vacuum breaker are redundant and fail safe. This assures that no single failure will defeat alarming the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or that system degradation has occurred and additional testing is required immediately. The frequency of testing the alarms is based on experience and quality of the equipment.

B. Standby Gas Treatment System and
C. Secondary Containment - Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the NS Savannah (ORNL 3726). Pressure drop tests across filter sections are performed to detect gross plugging or leak paths through the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval of once per operating cycle is reasonable. Duct heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient.

The in-place testing of charcoal filters is performed using Freon-112 or equivalent, which
3 is injected into the system upstream of the charcoal filters. Measurements of the Freon concentration upstream and downstream of the charcoal filters are made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding
3 capacity for elemental iodine and/or organic iodine, the test also gives an indication of the relative efficiency of the installed system.

The fission product source term defined in TID 14844 was used in evaluating the charcoal filters.

High-efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This