DOCKET FILE

January 11, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS RELATED TO 10 CFR PART 50, APPENDIX J, OPTION B (TAC NOS. M94061, M94062, M94065, AND M94066)

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. DPR-19 and Amendment No. 142 to Facility Operating License No. DPR-25 for Dresden, Units 2 and 3, and Amendment No. 169 to Facility Operating License No. DPR-29 and Amendment No. 165 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated November 14, 1995 and supplemented January 4, 1996. The November 14, 1995, and January 4, 1996, applications also requested the same changes for the LaSalle facility. Those requests will be addressed separately.

The amendments revise the Technical Specifications to incorporate 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original signed by:
John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254, 50-265
Enclosures:
1. Amendment No. 148 to DPR-19
2. Amendment No. 142 to DPR-25
3. Amendment No. 169 to DPR-29
4. Amendment No. 165 to DPR-30
5. Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated November 14, 1995, and supplemented January 4, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 148, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 148

FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-5	1-5
3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3
3/4.7-4	3/4.7-4
3/4.7-7	3/4.7-7
B 3/4.7-1	B 3/4.7-1
B 3/4.7-2	B 3/4.7-2
B 3/4.7-3	B 3/4.7-3
B 3/4.7-4	B 3/4.7-4
B 3/4.7-5	B 3/4.7-5
B 3/4.7-6	B 3/4.7-6
B 3/4.7-7	B 3/4.7-7
B 3/4.7-8	B 3/4.7-8
-	6-12a

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. Perform required visual examinations and leakage rate testing except for primary containment air lock testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.

2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. DELETED

4.7 - SURVEILLANCE REQUIREMENTS

B. DELETED

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3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

1. With one primary containment air lock door inoperable:
   a. Maintain at least the OPERABLE air lock door closed[b] and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.
   b. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By performing required primary containment air lock leakage testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program[c][d].
2. At least once per 6 months, by verifying that only one door in each air lock can be opened at a time[e].

---

a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

d Results shall be evaluated against acceptance criteria applicable to Specification 4.7.A.1.

e Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

    a. The inoperable valve is restored to OPERABLE status, or

    b. The instrument line is isolated and the associated instrument is declared inoperable.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is $\leq$11.5 scfh.

BASES

3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting leakage to less than 1.0 La except prior to the first startup after performing a required leakage test. At this time, Primary Containment Leakage Rate Testing Program leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (La) is 1.6% by weight of the containment atmosphere per day at the design basis LOCA maximum peak containment pressure (Pa) of 48 psig.

Surveillance requirements maintain PRIMARY CONTAINMENT INTEGRITY by requiring compliance with visual examinations and leakage rate test requirements and test frequencies of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing or main steam isolation valve leakage does not necessarily result in a failure of the surveillance requirement. The impact of the failure to meet such surveillance requirements must be evaluated against the Type A, Type B and Type C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be less than 0.60 La for combined Type B and Type C leakage, and 0.75 La for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of less than or equal to 1.0 La. At less than or equal to 1.0 La, the off-site dose consequences are bounded by the assumptions of the safety analysis.

3/4.7.B DELETED

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3.7.A. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The surveillance requirements reflect the leakage rate testing requirements with respect to air lock leakage (Type B

leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. The surveillance requirements have been annotated such that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and Type C primary containment leakage.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. G. Eisenhut (NRC) to Mr. L. DelGeorge (CECo) dated June 25, 1982.)

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers

are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

BASES

3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I Primary Containment Nitrogen System

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of

a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on 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 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the low pressure coolant injection (LPCI)/containment cooling system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression chamber

BASES

and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the LPCI/containment cooling system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the LPCI/containment cooling system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, containment cooling service water (CCSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

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3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. This unique arrangement requires that special allowed out-of-service times be provided for combinations of subsystem and diesel generator power source inoperability that may occur. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

5. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.6% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 142
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated November 14, 1995, and supplemented January 4, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 142 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 142

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-5	1-5
3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3
3/4.7-4	3/4.7-4
3/4.7-7	3/4.7-7
B 3/4.7-1	B 3/4.7-1
B 3/4.7-2	B 3/4.7-2
B 3/4.7-3	B 3/4.7-3
B 3/4.7-4	B 3/4.7-4
B 3/4.7-5	B 3/4.7-5
B 3/4.7-6	B 3/4.7-6
B 3/4.7-7	B 3/4.7-7
B 3/4.7-8	B 3/4.7-8
-	6-12a

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. Perform required visual examinations and leakage rate testing except for primary containment air lock testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.
2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.
3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.
4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. DELETED

4.7 - SURVEILLANCE REQUIREMENTS

B. DELETED

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3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

1. With one primary containment air lock door inoperable:

   a. Maintain at least the OPERABLE air lock door closed[b] and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

   b. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By performing required primary containment air lock leakage testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program[c][d].

2. At least once per 6 months, by verifying that only one door in each air lock can be opened at a time[e].

---

a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

d Results shall be evaluated against acceptance criteria applicable to Specification 4.7.A.1.

e Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

    a. The inoperable valve is restored to OPERABLE status, or

    b. The instrument line is isolated and the associated instrument is declared inoperable.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is $\leq$11.5 scfh.

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3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting leakage to less than 1.0 La except prior to the first startup after performing a required leakage test. At this time, Primary Containment Leakage Rate Testing Program leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (La) is 1.6% by weight of the containment atmosphere per day at the design basis LOCA maximum peak containment pressure (Pa) of 48 psig.

Surveillance requirements maintain PRIMARY CONTAINMENT INTEGRITY by requiring compliance with visual examinations and leakage rate test requirements and test frequencies of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing or main steam isolation valve leakage does not necessarily result in a failure of the surveillance requirement. The impact of the failure to meet such surveillance requirements must be evaluated against the Type A, Type B and Type C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be less than 0.60 La for combined Type B and Type C leakage, and 0.75 La for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of less than or equal to 1.0 La. At less than or equal to 1.0 La, the off-site dose consequences are bounded by the assumptions of the safety analysis.

3/4.7.B DELETED

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3.7.A. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The surveillance requirements reflect the leakage rate testing requirements with respect to air lock leakage (Type B

leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. The surveillance requirements have been annotated such that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and Type C primary containment leakage.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. G. Eisenhut (NRC) to Mr. L. DelGeorge (CECo) dated June 25, 1982.)

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers

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are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

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3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I Primary Containment Nitrogen System

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of

a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on 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 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

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Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the low pressure coolant injection (LPCI)/containment cooling system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the LPCI/containment cooling system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the LPCI/containment cooling system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, containment cooling service water (CCSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

BASES

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. This unique arrangement requires that special allowed out-of-service times be provided for combinations of subsystem and diesel generator power source inoperability that may occur. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

5. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.6% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the combined Type B and Type C tests, and ≤ 0.75 L_a for Type A tests.

b. Air lock testing acceptance criteria is the overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By letters dated November 14, 1995, and January 4, 1996, Commonwealth Edison Company (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30) for the Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station Units 1 and 2. The proposed changes would revise the technical specifications to reflect the approval for the licensee to use 10 CFR Part 50, Appendix J, Option B for the Dresden and Quad Cities Stations containment leakage rate test programs. The Commission has made an initial no significant hazards consideration determination regarding this request. That determination was published in the Federal Register on December 7, 1995 (60 FR 62896). The January 4, 1996, supplement only requested a change in the implementation schedule for the amendment. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The November 14, 1995, and January 4, 1996, letters also requested similar changes for the LaSalle County Station. These requested changes will be addressed separately.

2.0 BACKGROUND

Compliance with Appendix J provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate values specified in the technical specifications and bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. 10 CFR Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program", was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that Regulatory Guide 1.163 or another implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant technical specifications.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed technical specifications for implementing Option B. After some discussion, the staff and NEI agreed on a set of model technical specifications which were transmitted to NEI in a letter dated November 2, 1995. These technical specifications are to serve as a model for licensees to develop plant-specific technical specifications in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, Regulatory Guide 1.163 provides that a licensee establish an administrative leakage limit. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not technical specifications requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's November 14, 1995, letter to the NRC proposes to establish a "Primary Containment Leakage Rate Program" and proposes to add this program to the technical specifications. The program references Regulatory Guide 1.163 which specifies methods acceptable to the NRC for complying with Option B. This requires a change to existing Technical Specifications 3/4.7.A, 3/4.7.C, and 3/4.7.D, the deletion of 3/4.7.B, and the addition of the program to Section 6.8 of the technical specifications.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B, and C testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis. The licensee has committed to a Primary Containment Leakage Rate Testing Program in accordance with the guidelines contained in Regulatory Guide 1.163. Technical specifications consistent with those transmitted to NEI in a letter dated November 2, 1995, except as noted below, were also proposed.

The technical specification changes proposed by the licensee differ with the model technical specifications developed by the NRC staff in cooperation with NEI on one item. The generic surveillance for secondary containment integrity requires verifying that the leakage rate for all secondary containment bypass leakage meets certain criteria at a frequency in accordance with the Primary Containment Leakage Rate Testing Program. The licensee, however, has chosen to retain its existing surveillance which requires verifying once per 24 hours that the pressure within the secondary containment is $\geq$ 0.25 inch of vacuum water gauge, verifying once per 31 days that appropriate doors and penetrations are closed, and verifying once per 18 months that each standby gas treatment train can produce adequate secondary containment vacuum at a specified flow rate. The current specifications provide adequate assurance of secondary containment, were previously approved by the staff, and are acceptable. Based on the above, the licensee's proposed changes implementing Option B of Appendix J are acceptable.

Option B states that specific existing exemptions to Option A are still applicable unless specifically revoked by the NRC. Both Dresden and Quad Cities currently have approved exemptions to 10 CFR Part 50, Appendix J, that were issued by the NRC on June 25, 1982, and June 12, 1984, respectively. These exemptions, which focus on testing methodology aspects of Appendix J, are unaffected by the change to the Option B testing frequency requirements and are not affected by this amendment.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 62896). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Lobel (by precedent)
D. Wigginton (by precedent)
J. Hickman

Date: January 11, 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated November 14, 1995, and supplemented January 4, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 169 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 169

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-5	1-5
3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3
3/4.7-4	3/4.7-4
3/4.7-7	3/4.7-7
B 3/4.7-1	B 3/4.7-1
B 3/4.7-2	B 3/4.7-2
B 3/4.7-3	B 3/4.7-3
B 3/4.7-4	B 3/4.7-4
B 3/4.7-5	B 3/4.7-5
B 3/4.7-6	B 3/4.7-6
B 3/4.7-7	B 3/4.7-7
B 3/4.7-8	B 3/4.7-8
-	6-12a

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. Perform required visual examinations and leakage rate testing except for primary containment air lock testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.
2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.
3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.
4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION	4.7 - SURVEILLANCE REQUIREMENTS
B. DELETED	B. DELETED

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3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

1. With one primary containment air lock door inoperable:
   a. Maintain at least the OPERABLE air lock door closed[b] and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.
   b. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By performing required primary containment air lock leakage testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program[c][d].
2. At least once per 6 months, by verifying that only one door in each air lock can be opened at a time[e].

---

a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

d Results shall be evaluated against acceptance criteria applicable to Specification 4.7.A.1.

e Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is $\leq$11.5 scfh.

3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting leakage to less than 1.0 La except prior to the first startup after performing a required leakage test. At this time, Primary Containment Leakage Rate Testing Program leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (La) is 1.0% by weight of the containment atmosphere per day at the design basis LOCA maximum peak containment pressure (Pa) of 48 psig.

Surveillance requirements maintain PRIMARY CONTAINMENT INTEGRITY by requiring compliance with visual examinations and leakage rate test requirements and test frequencies of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing or main steam isolation valve leakage does not necessarily result in a failure of the surveillance requirement. The impact of the failure to meet such surveillance requirements must be evaluated against the Type A, Type B and Type C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be less than 0.60 La for combined Type B and Type C leakage, and 0.75 La for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of less than or equal to 1.0 La. At less than or equal to 1.0 La, the off-site dose consequences are bounded by the assumptions of the safety analysis.

3/4.7.B DELETED

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3.7.A. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The surveillance requirements reflect the leakage rate testing requirements with respect to air lock leakage (Type B

leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. The surveillance requirements have been annotated such that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and Type C primary containment leakage.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. B. Vassallo (NRC) to Mr. D. L. Farrar (CECo) dated June 12, 1984.)

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers

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are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

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3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I Primary Containment Nitrogen Concentration

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional nitrogen supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of

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a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on 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 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the containment cooling mode of the residual heat removal (RHR) system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression

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chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the RHR system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the RHR system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, residual heat removal service water (RHRSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

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3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. This unique arrangement requires that special allowed out-of-service times be provided for the combinations of subsystem and diesel generator power source inoperability that may occur. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

5. Primary Containment Leakage Rate Testing Program

   A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

   The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

   The maximum allowable primary containment leakage rate, L_a, at P_a, is 1% of primary containment air weight per day.

   Leakage rate acceptance criteria are:

   a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

   b. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

   The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated November 14, 1995, and supplemented January 4, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang For

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 11, 1996

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

ATTACHMENT TO LICENSE AMENDMENT NO. 165

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-5	1-5
3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3
3/4.7-4	3/4.7-4
3/4.7-7	3/4.7-7
B 3/4.7-1	B 3/4.7-1
B 3/4.7-2	B 3/4.7-2
B 3/4.7-3	B 3/4.7-3
B 3/4.7-4	B 3/4.7-4
B 3/4.7-5	B 3/4.7-5
B 3/4.7-6	B 3/4.7-6
B 3/4.7-7	B 3/4.7-7
B 3/4.7-8	B 3/4.7-8
-	6-12a

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. Perform required visual examinations and leakage rate testing except for primary containment air lock testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.
2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.
3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.
4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION	4.7 - SURVEILLANCE REQUIREMENTS
B. DELETED	B. DELETED

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3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

1. With one primary containment air lock door inoperable:

   a. Maintain at least the OPERABLE air lock door closed[b] and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

   b. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By performing required primary containment air lock leakage testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program[c][d].

2. At least once per 6 months, by verifying that only one door in each air lock can be opened at a time[e].

---

a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

d Results shall be evaluated against acceptance criteria applicable to Specification 4.7.A.1.

e Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

    a. The inoperable valve is restored to OPERABLE status, or

    b. The instrument line is isolated and the associated instrument is declared inoperable.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is $\leq$11.5 scfh.

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3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting leakage to less than 1.0 La except prior to the first startup after performing a required leakage test. At this time, Primary Containment Leakage Rate Testing Program leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (La) is 1.0% by weight of the containment atmosphere per day at the design basis LOCA maximum peak containment pressure (Pa) of 48 psig.

Surveillance requirements maintain PRIMARY CONTAINMENT INTEGRITY by requiring compliance with visual examinations and leakage rate test requirements and test frequencies of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing or main steam isolation valve leakage does not necessarily result in a failure of the surveillance requirement. The impact of the failure to meet such surveillance requirements must be evaluated against the Type A, Type B and Type C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be less than 0.60 La for combined Type B and Type C leakage, and 0.75 La for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of less than or equal to 1.0 La. At less than or equal to 1.0 La, the off-site dose consequences are bounded by the assumptions of the safety analysis.

3/4.7.B DELETED

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3.7.A. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The surveillance requirements reflect the leakage rate testing requirements with respect to air lock leakage (Type B

leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. The surveillance requirements have been annotated such that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and Type C primary containment leakage.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. B. Vassallo (NRC) to Mr. D. L. Farrar (CECo) dated June 12, 1984.)

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers

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are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

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3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I Primary Containment Nitrogen Concentration

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional nitrogen supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of

a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on 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 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the containment cooling mode of the residual heat removal (RHR) system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression

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chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the RHR system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the RHR system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, residual heat removal service water (RHRSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. This unique arrangement requires that special allowed out-of-service times be provided for the combinations of subsystem and diesel generator power source inoperability that may occur. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

5. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 1% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.