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. 157
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (ComEd, the licensee) dated February 17, 1997, as supplemented February 27, March 12, March 26, April 2, and April 10, 1997, 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, amending paragraphs 2.C.(2) and 2.C.(6), and adding paragraph 2.C.(7). Facility Operating License No. DPR-19* is hereby amended to read as follows:

---

*License pages 3, 3a and 5 are provided, for convenience, for the composite license to reflect this change.

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 157, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

2.C.(6) Surveillance Requirements

The Surveillance Requirements contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 150:

a. Surveillance Requirement 4.1.A.2 - RPS Logic System Functional Test

b. Surveillance Requirement 4.2.A.2 - Primary & Secondary Containment Logic System Functional Test

c. Surveillance Requirement 4.2.J.2 - Feedwater Pump Trip Logic System Functional Test

d. Surveillance Requirement 4.6.F.1.b - Relief Valve Logic System Functional Test

e. Surveillance Requirement 4.9.A.9 - Simultaneous Diesel Generator Start

f. Surveillance Requirement 4.9.A.10 - Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fifteenth refueling outage (D2R15).

2.C.(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 157, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

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

Attachments:
1. License pages 3, 3a and 5
2. Changes to the Technical Specifications

Date of Issuance: April 30, 1997

(5) ComEd, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2527 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 157, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

(4) The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

(5) The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

(6) Surveillance Requirements

The Surveillance Requirements contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 150:

a. Surveillance Requirement 4.1.A.2 - RPS Logic System Functional Test

b. Surveillance Requirement 4.2.A.2 - Primary & Secondary Containment Logic System Functional Test

c. Surveillance Requirement 4.2.J.2 - Feedwater Pump Trip Logic System Functional Test

d. Surveillance Requirement 4.6.F.1.b - Relief Valve Logic System Functional Test

e. Surveillance Requirement 4.9.A.9 - Simultaneous Diesel Generator Start

f. Surveillance Requirement 4.9.A.10 - Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fifteenth refueling outage (D2R15).

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 157, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to the licensee in letters dated February 2, 1983, September 28, 1987, July 6, 1989, and August 15, 1989.

In addition, the facility has been granted certain exemptions from Sections II and III of Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for test of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and set to the licensee in a letter dated June 25, 1982.

I. This license is effective as of the date of issuance and shall expire at midnight on January 10, 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A - Technical Specifications
Appendix B - Additional Conditions

Date of Issuance: February 20, 1991

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-19

Commonwealth Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
157	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Effective as of the issuance of Amendment No. 157 and shall be implemented within 30 days.

Time (seconds)	Containment Pressure (PSIG)
0-240	9.5
240-480	2.9
480-6000	1.9
6000-accident end	2.5

Amendment Number	Additional Condition	Implementation Date
157	The EOPs shall be changed to alert operators to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 157.

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. 152
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (ComEd, the licensee) dated February 17, 1997, as supplemented February 27, March 12, March 26, April 2, and April 10, 1997, 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, amending paragraph 3.B., and adding paragraph 3.O. Facility Operating License No. DPR-25* is hereby amended to read as follows:

---

*License page 6 is provided, for convenience, for the composite license to reflect this change.

3.B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 152, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.0 Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 152, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to Unit 3 returning to Mode 3 from the current refueling outage D3R14.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

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

Attachments:
1. License page 6
2. Changes to the Technical Specifications

Date of Issuance: April 30, 1997

L. Deleted. [Amdt. 87, 7-24-86]

M. Deleted. [Amdt. 85, 12-12-85]

N. By Amendment No. 144, the license is amended to allow, on a one time temporary basis, operation of Dresden, Unit 3, with the corner room structural steel members in the Low Pressure Coolant Injection Corner Rooms outside the Updated Final Safety Analysis Report (UFSAR) design parameters. Operation under these conditions is allowed up to and including the next scheduled refueling outage (D3R14).

The repairs to Dresden, Unit 3, corner room structural steel shall restore the steel design margins to the current UFSAR (updated through Revision 1A) design criteria. The design of the modifications to the Dresden, Unit 3, corner room structural steel members will be based on use of elastic section modules and the structural steel stresses will be limited to 1.6 of the American Institute of Steel Construction (AISC allowables). The modifications to Dresden, Unit 3, corner room structural steel will be implemented during the upcoming D3R14 refueling outage.

During this interim period of operation, should vibratory ground motion exceeding the UFSAR Operating Basis Earthquake (OBE) design parameters, Dresden, Unit 3, will be shut down for inspection and will not start up without prior NRC approval.

O. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 152, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

4. This license is effective as of the date of issuance and shall expire at Mid-night January 12, 2001.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed By:

Peter A. Morris, Director
Division of Licensing

Enclosures:
Appendix A - Technical Specifications
Appendix B - Additional Conditions

Date of Issuance: January 12, 1971

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-25

Commonwealth Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
152	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.

Time (seconds)	Containment Pressure (PSIG)
0-240	9.5
240-480	2.9
480-6000	1.9
6000-accident end	2.5

Amendment Number	Additional Condition	Implementation Date
152	The licensee shall complete the evaluation of the torus attached piping.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.
152	The EOPs shall be changed to alert operators to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 152.

ATTACHMENT TO LICENSE AMENDMENT NOS. 157 AND 152

FACILITY OPERATING LICENSE NO. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 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 numbers and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4.7-16	3/4.7-16
3/4.7-17	3/4.7-17
B 3/4.7-5	B 3/4.7-5
B 3/4.7-6	B 3/4.7-6
3/4.8-5	3/4.8-5

3.7 - LIMITING CONDITIONS FOR OPERATION

K. Suppression Chamber

The suppression chamber shall be OPERABLE with:

1. The suppression pool water level between 14' 6.5" and 14' 10.5",

2. A suppression pool maximum average water temperature of ≤95°F during OPERATIONAL MODE(s) 1 or 2, except that the maximum average temperature may be permitted to increase to:

    a. ≤105°F during testing which adds heat to the suppression pool.

    b. ≤110°F with THERMAL POWER ≤1% of RATED THERMAL POWER.

    c. ≤120°F with the main steam line isolation valves closed following a scram.

3. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1 inch diameter orifice at a differential pressure of 1.0 psid.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the suppression pool water level outside the above limits, restore the water level to within the limits

4.7 - SURVEILLANCE REQUIREMENTS

K. Suppression Chamber

The suppression chamber shall be demonstrated OPERABLE:

1. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

2. At least once per 24 hours by verifying the suppression pool average water temperature to be ≤95°F, except:

    a. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature to be ≤105°F.

    b. At least once per hour when suppression pool average water temperature is ≥ 95°F, by verifying:

        1) Suppression pool average water temperature to be ≤ 110°F, and

        2) THERMAL POWER to be ≤ 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 95°F for more than 24 hours.

    c. At least once per 30 minutes with the main steam isolation valves closed following a scram and suppression pool average water temperature >95°F, by verifying suppression pool average water temperature to be ≤120°F.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. Deleted.

4. Deleted.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

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 safety/relief valve discharges or from Design Basis Accidents (DBAs). This is the essential mitigative feature of a pressure-suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from the High Pressure Coolant injection turbine system exhaust lines. Suppression pool average temperature, in conjunction with suppression pool water level is a key indication of the capacity of the suppression pool to fulfill these requirements. 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 once per 24 hours 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.

BASES

A limitation of the suppression pool average temperature is required to ensures that the containment conditions assumed in the safety analyses are met. This limitation subsequently ensure that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heat-up of the suppression pool. The postulated DBA against which the primary containment performance is evaluated in the entire spectrum of postulated pipe breaks within the primary containment. Input to the safety analyses include initial suppression pool water volume and suppression pool temperature. An initial pool temperature of 95°F is assumed for these analyses. Reactor shutdown at 110°F and vessel de-pressurization at a pool temperature of 120°F are also assumed for these analyses. The limit of 105°F at which testing is terminated, is not used in the safety analyses because DBAs are assumed not to initiate during plant testing.

The suppression pool is also designed to quench the energy from safety/relief valve discharges. Thus, the safety analyses related to the suppression pool must consider all accident scenarios that involve safety/relief valve actuations. The limit for the suppression pool average temperature is set low enough to preclude local boiling due to safety/relief valve discharge via the T-quencher devices. In accordance with GE NEDO-30832, local suppression pool temperature limits are not required because the emergency core cooling system pump inlets are located below the elevation of the quenchers.

The available net positive suction head may be less than that required by the emergency core cooling system pumps, thus there is dependency on containment over pressure during the accident injection phase.

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.

3.8 - LIMITING CONDITIONS FOR OPERATION

C. Ultimate Heat Sink

The ultimate heat sink shall be OPERABLE with:

1. A minimum water level at or above elevation 500 ft Mean Sea Level, and

2. An average water temperature of ≤95°F.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, 5 and *.

ACTION:

With the requirements of the above specification not satisfied:

1. In OPERATIONAL MODE(s) 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. In OPERATIONAL MODE(s) 4 or 5 declare the diesel generator cooling water system inoperable and take the ACTION required by Specification 3.8.B.

3. In OPERATIONAL MODE *, declare the diesel generator cooling water system inoperable and take the ACTION required by Specification 3.8.B. The provisions of Specification 3.0.C are not applicable.

4.8 - SURVEILLANCE REQUIREMENTS

C. Ultimate Heat Sink

The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

---

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential to drain the reactor vessel.