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NRC PDR (4) JRBuchanan
Local PDR (2)
ORB #2 Reading
VStello
KRGoller
RMDiggs
MGrotenhuis
PWO'Connor
OELD
OI&E (5)
BJones (8)
SCharf (10)
JMMcGough
BHarless
DEisenhut
ACRS (16)
OPA (CMiles)
DRoss

MAY 2 1977

Docket Nos. 50-237/249
50-254/265

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your request dated March 8, 1977, the Commission has issued the enclosed Amendment Nos. 30 and 28 to Facility Operating License Nos. DPR-19 and DPR-25 for Unit Nos. 2 and 3 of the Dresden Nuclear Station and Amendment Nos. 40 and 38 to Facility Operating License Nos. DPR-29 and DPR-30 for Unit Nos. 1 and 2 of the Quad Cities Nuclear Station.

These amendments incorporate changes to the surveillance frequency and testing methods utilized to assure the operability of the relief valves used in the automatic pressure relief subsystem. During our review we discussed several changes to your proposed Technical Specifications. Your staff has agreed to these changes and they have been incorporated in the enclosed amendments.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment Nos. 30/28 to Licenses DPR-19/25
2. Amendment Nos. 40/38 to Licenses DPR-29/30
3. Safety Evaluation
4. Notice

OFFICE	DOR:ORB #2 PWOC	DOR:ORB #2 PWOC	DOR:ORB #2	DOR:STS	OELD	DOR:ORB #2
SURNAME	PWO'Connor:ah	MGrotenhuis	RMDiggs	JMMcGough	D SWANSON	DKDavis
DATE	4/6/77	4/6/77	4/5/77	4/14/77	4/29/77	4/2/77

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc w/enclosures:
Mr. Charles Whitmore
President and Chairman
Iowa-Illinois Gas and
Electric Company
206 East Second Avenue
Davenport, Iowa 52801

Mr. John W. Rowe
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Anthony Z. Roisman, Esquire
Roisman, Kessler and Cashdan
1025 15th Street, N. W., 5th Floor
Washington, D. C. 20005

Moline Public Library
504 - 17th Street
Moline, Illinois 61265

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Mr. William Waters
Chairman, Board of Supervisors
of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Mr. Marcel DeJaegher, Chairman
Rock Island County Board
of Supervisors
Rock Island County Court House
Rock Island, Illinois 61201

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

Mr. N. Kalivianakas
Plant Superintendent
Quad Cities Nuclear Power Station
22710 206th Avenue, North
Cordova, Illinois 61242

cc w/enclosures and cy of CECo's
filing dtd. 3/8/77:
Department of Public Health
ATTN: Chief, Division of
Radiological Health
535 West Jefferson
Springfield, Illinois 62706

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 30
License No. DPR-19

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 8, 1977, 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 License No. DPR-19 is hereby amended to read as follows:

   "B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 30, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 30

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following existing pages of the Technical Specifications with the attached revised pages. Changed areas on the revised pages are shown by a marginal line.

REMOVE PAGES	INSERT PAGES
78	78
86	85D
	86

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
D. Automatic Pressure Relief Subsystems	D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:
1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.	1. During each operating cycle the following shall be performed: a. A simulated automatic initiation which opens all pilot valves, and b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure. c. A logic system functional test shall be performed each refueling outage.
2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.	2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter
3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.	3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately

Bases:

4.5.A.-4.5.F.

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were simulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

H. <u>Maintenance of Filled Discharge Pipe</u>

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 28
License No. DPR-25

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 8, 1977, 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 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. 28, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Replace the following existing pages of the Technical Specifications with the attached revised pages. Changed areas on the revised pages are shown by a marginal line.

REMOVE PAGES	INSERT PAGES
78	78
86	85D
	86

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

   c. A logic system functional test shall be performed each refueling outage.

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

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

Bases:

4.5.A.-4.5.F.

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were simulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
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 March 8, 1977, 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 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. 40, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following existing pages of the Appendix A portion of the Technical Specifications with the attached revised pages. Changed areas on the revised pages are shown by a marginal line.

REMOVE PAGES	INSERT PAGES
3.5/4.5-5	3.5/4.5-5
3.5/4.5-16	3.5/4.5-16
	3.5/4.5-16A

provided that during such 7 days all active components of the automatic pressure relief subsystems, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable.

3. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated, and the reactor pressure shall be reduced to 90 psig within 24 hours.

operable immediately. The automatic pressure relief and RCIC systems shall be demonstrated to be operable daily thereafter.

D. Automatic Pressure Relief Subsystems

1. The automatic pressure relief subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 7 days unless repairs are made and provided that during such time the HPCI subsystem is operable.

3. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

D. Automatic Pressure Relief Subsystems

Surveillance of the automatic pressure relief subsystems shall be performed as follows:

1. The following surveillance shall be carried out on a six-month surveillance interval:
   a. A simulated automatic initiation which opens all pilot valves.
   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.
2. A logic system functional test shall be performed each refueling outage.
3. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by themocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for the relief valve opening. This test method may be performed over a wide range of reactor pressures greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems is filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Instrumentation has been provided to monitor the presence of water in the discharge piping between the monthly intervals at which the lines are vented and alarm the control room if it is not. This instrumentation will be calibrated on the same frequency as the safety system instrumentation and the alarm system tested monthly. This testing ensures that, during the interval between the monthly venting checks, the status of the discharge piping is monitored on a continuous basis.

An alarm point of ≥ 40 psig for the low pressure of the fill system has been chosen because, due to elevations of piping within the plant, 39 psig is required to keep the lines full. The shutoff head of the fill system pumps is 74 psig and therefore will not defeat the low-pressure cooling pump discharge press interlock of ≥ 75 psig as shown in Table 3.2-2.

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls and ceilings have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test an electrical penetration, compressed air is supplied to a test connection and the space between the fittings is pressurized to approximately 15 psig. The outer faces are then tested for leaks using a soap bubble solution.

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
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 March 8, 1977, 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 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. 38, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Replace the following existing pages of the Appendix A portion of the Technical Specifications with the attached revised pages. Changed areas on the revised pages are shown by a marginal line.

REMOVE PAGES	INSERT PAGES
3.5/4.5-5	3.5/4.5-5
3.5/4.5-15	3.5/4.5-15
	3.5/4.5-15A

provided that during such 7 days all active components of the automatic pressure relief subsystems, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable.

3. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated, and the reactor pressure shall be reduced to 90 psig within 24 hours.

D. Automatic Pressure Relief Subsystems

1. The automatic pressure relief subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 7 days unless repairs are made and provided that during such time the HPCI subsystem is operable.

3. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

operable immediately. The automatic pressure relief and RCIC systems shall be demonstrated to be operable daily thereafter.

D. Automatic Pressure Relief Subsystems

Surveillance of the automatic pressure relief subsystems shall be performed as follows:

1. The following surveillance shall be carried out on a six-month surveillance interval:
   a. A simulated automatic initiation which opens all pilot valves.
   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.
2. A logic system functional test shall be performed each refueling outage.
3. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by themocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for the relief valve opening. This test method may be performed over a wide range of reactor pressures greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems is filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Instrumentation has been provided to monitor the presence of water in the discharge piping between the monthly intervals at which the lines are vented and alarm the control room if it is not. This instrumentation will be calibrated on the same frequency as the safety system instrumentation and the alarm system tested monthly. This testing ensures that, during the interval between the monthly venting checks, the status of the discharge piping is monitored on a continuous basis.

An alarm point of ≥ 40 psig for the low pressure of the fill system has been chosen because, due to elevations of piping within the plant, 39 psig is required to keep the lines full. The shutoff head of the fill system pumps is 74 psig and therefore will not defeat the low-pressure cooling pump discharge press interlock of ≥ 75 psig as shown in Table 3.2-2.

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls and ceilings have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test an electrical penetration, compressed air is supplied to a test connection and the space between the fittings is pressurized to approximately 15 psig. The outer faces are then tested for leaks using a soap bubble solution.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 30, 28, 40, and 38, TO FACILITY LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30

COMMONWEALTH EDISON COMPANY AND IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DRESDEN NUCLEAR POWER STATION UNIT NOS. 2 AND 3

QUAD CITIES NUCLEAR POWER STATION UNIT NOS. 1 AND 2

DOCKET NOS. 50-237/249/254/265

INTRODUCTION

By letter dated March 8, 1977, Commonwealth Edison requested amendments to the Dresden Unit Nos. 2 and 3 and Quad Cities Unit Nos. 1 and 2 Technical Specifications which would modify the existing method of testing the operability of the relief valves used in the automatic pressure relief subsystem at these facilities. The modified surveillance techniques require observation of turbine bypass closure or control valve closure as verification of relief valve operation during testing. The previous method of testing required detection of an increase in temperature at the relief valve discharge caused by steam exiting from the relief valve.

BACKGROUND

The staff has recently become aware of a potential deficiency in the method being used to confirm valve operability during periodic testing of Boiling Water Reactor (BWR) safety-relief valves. This deficiency concerns the use of the safety-relief valve temperature indication as a positive method of confirmation that a safety-relief valve is open when manually actuated during surveillance testing.

We have found that an increased temperature indication may be obtained at the safety-relief valve exit with the safety-relief valve closed. This indicated temperature increase is the result of steam vented through the valve actuation mechanism during the surveillance test. In view of this finding, we have concluded that a temperature increase at the valve exit, by itself, does not provide a positive means of verification that the safety-relief valve has opened.

When we became aware of the above deficiency in some BWR Technical Specifications, we requested Commonwealth Edison (CECo), by our letter dated January 5, 1977, to propose a change to the technical specifications of their boiling water reactors. We provided examples of acceptable methods that CECo could propose to provide assurance of satisfactory operation of the relief valves. In response to our request, CECo submitted their March 8, 1977 request for amendments to Licenses DPR-19, -25, -29, and -30.

EVALUATION

We have reviewed CECo's proposed change to the Technical Specifications for Dresden Unit Nos. 2 and 3 and Quad Cities Unit Nos. 1 and 2. CECo has proposed the deletion of the unreliable specification based on relief valve exit temperature and replaced it with a specification based upon the observation of the turbine bypass valve or control valve closure that occurs concurrent with the actual opening of the relief valve and compensates for the diversion of steam to the suppression pool.

We have concluded that the method proposed by CECo provides assurance that the relief valve has actually opened during surveillance testing and is therefore acceptable to the NRC staff.

In addition to the changes proposed by the licensee, the NRC staff has concluded that the surveillance frequency at Quad Cities Station should be increased from once per operating cycle to once per six months. The NRC staff has determined that the increased testing frequency is necessary to assure early detection of relief valve malfunctions that have been occurring more frequently at Quad Cities Station than at any other nuclear power station. The Commonwealth Edison Company has agreed to the staff's suggested shortening of the surveillance interval for the relief valves at Quad Cities Station.

ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendments do not involve a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSIONS

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 2, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 30, 28, 40 and 38 to Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30 (respectively), issued to the Commonwealth Edison Company (and, in the matter of License Nos. DPR-29 and DPR-30, the Iowa-Illinois Gas and Electric Company), which revised Technical Specifications for operation of Unit Nos. 2 and 3 of the Dresden Nuclear Power Station (located in Grundy County, Illinois) and Unit Nos. 1 and 2 of the Quad Cities Nuclear Power Station (located in Rock Island County, Illinois). These amendments are effective as of their date of issuance.

The amendments incorporated changes in the Technical Specifications surveillance frequency and testing methods utilized to assure the operability of the relief valves used in the automatic pressure relief subsystem.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendment dated March 8, 1977, (2) Amendment Nos. 30 and 28 to License Nos. DPR-19 and DPR-25, and Amendment Nos. 40 and 38 to License Nos. DPR-29 and DPR-30, and (3) the Commission's concurrently issued related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and for those items relating to Dresden Unit Nos. 2 and 3 at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60450, and for those items relating to Quad Cities Unit Nos. 1 and 2 at the Moline Public Library, 504 - 17th Street, Moline, Illinois 60625. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this second day of May, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors