MAY 24 1974

Docket Nos. 50-249 and 50-254

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

Your application dated March 29, 1974, supplemented by letters of April 18 (50-249 only), April 24 (50-254 only), and April 22, 1974, requested authorization to replace one reactor coolant system electromatic relief valve with a combination safety/relief valve and requested approval of several changes to technical specifications related to this valve modification and to scram reactivity considerations. Our Safety Evaluation of your proposal is enclosed.

We have concluded that the proposed modification and changes to Technical Specifications do not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, Amendment Nos. 3 and 8 to Facility Operating Licenses Nos. DPR-25 and DPR-29, respectively, are enclosed revising the Technical Specifications thereto to authorize replacement of an electromatic relief valve with a combination safety/relief valve and associated changes. A copy of a notice which is being forwarded to the Office of the Federal Register for publication relating to this action also is enclosed for your information.

Sincerely,

Original signed by
Dennis L. Ziemann

~~Karl~~ R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing

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Enclosures and cc: See next page

Enclosures:
1. Safety Evaluation
2. Amendment No. 3 to License No. DPR-25
3. Amendment No. 8 to License No. DPR-29
4. Federal Register Notice

cc w/enclosures:
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President and Chairman
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Supervisors
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Chairman, Board of Supervisors
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Grundy County Courthouse
Morris, Illinois 60450

Morris Public Library

Moline Public Library

cc w/enclosures and cy of CECo ltrs
dtd 3/29, 4/18, and 4/22/74:
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SVarga, L:RP

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SURNAME ➤	RDSilver:sjh	JIRiesland	RMDiggs	DLZiemann	R. Kinsey	KRGo'
DATE ➤	5/2/74 ~~4/26/74~~	5/2/74 ~~4/26/74~~	5/2/74 ~~4/26/74~~	5/3/74	5/14/74 5/24	

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 18 81465-1 520-284

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NOS. 3 AND 8 TO LICENSE NOS. DPR-25 AND DPR-29

(CHANGE NOS. 20 AND 17 TO APPENDIX A OF TECHNICAL SPECIFICATIONS)

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT 3 (DOCKET NO. 50-249)

AND

QUAD-CITIES UNIT 1 (DOCKET NO. 50-254)

INTRODUCTION

By application dated March 29, 1974, and supplements dated April 18 (50-249 only), April 24 (50-254 only), April 22 and May 20, 1974, Commonwealth Edison requested authorization to replace one reactor coolant system electromatic relief valve with a combination safety/relief valve and requested approval of several changes to Technical Specifications. The changes to Technical Specifications include requirements for the modified valve, increased pressure set points for the spring-loaded safety valves, and more rapid scram times for the control rods. The purpose of the modification and changes is to provide greater margin between the calculated pressure rise in the relief valve sizing transient and the lowest setting of the spring-loaded safety valves. The need for the change is related to scram reactivity considerations.

DISCUSSION

The set point and capacity of reactor coolant system relief and safety valves ~~is~~ are determined from design codes and from comparisons of calculated pressure increases resulting from postulated abnormal and accident conditions with design criteria. Because a pressure increase also causes a power increase due to collapse of coolant voids, fuel element thermal-hydraulic margins for abnormal operational transients are also compared to design criteria and considered in determining the adequacy of relief valve design.

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One factor in the magnitude of the pressure and power transients following certain abnormal occurrences is the rate at which the reactor is shut-down; i.e., the rate at which reactivity is decreased by control rods following a scram signal. Analyses performed since the initial design evaluation of the acceptability of the relief and safety valve system show that the scram reactivity, or worth of the control rods as a function of vertical position, changes with core exposure. The change is such that the rate of shutdown following a scram signal is slower than postulated in performing the initial design evaluations of the relief and safety valves. The effects of the slower rate of shutdown on pressure and thermal-hydraulic design margins can be compensated by reducing reactor power level. Reactor power level had to be procedurally limited during the course of the last cycle at Dresden 3 and Quad-Cities 1 and, without modifications, limitations will be necessary in future cycles.

Commonwealth Edison has proposed the valve modification and Technical Specification changes as a step in minimizing power restrictions needed to compensate for revised scram reactivity curves. Commonwealth Edison estimates that without the proposed changes power restrictions early and late in the next fuel cycle would be 97% and 86% of licensed power, respectively. At full power, the margin between the lowest spring-loaded safety valve setting in the previous technical specifications (1210 psig) and the peak pressure from an assumed transient involving turbine trip without bypass would be very close to the pressure setting of the lowest safety valve and the design criteria minimum margin of 25 psi could not be assured. To assure that such margin is preserved, the applicant proposes to raise the settings on the spring-loaded safety valves and proposes to replace one of the electromagnetic relief valves with a Target Rock combination safety/relief valve. With the proposed changes, the allowable power early and late in the cycle would be 100% and 93%. The analyses performed to arrive at the allowable power were done utilizing methods and design criteria previously approved. The assumptions used were modified to account for proposed technical specification revisions to scram time limits and safety valve set points, and to account for core average exposures and the control rod management program through the next cycle. The analyses for exposures early in the cycle were performed using the "generic B" scram reactivity curve. Analyses performed for exposures beyond the point where the "generic B" curve is applicable were performed with an end of cycle, all rods out scram reactivity curve ("C" curve). These curves are selected to provide an envelope of actual scram reactivity worths for calculational purposes. Commonwealth Edison's analyses show that the limiting transient for relief valve design continues to be a postulated turbine trip without bypass. Using the "B" curve, the

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calculated pressure resulting from turbine trip without bypass is 1185 psig. The pressure margin to the lowest setting of a spring-loaded safety valve (1240 psig) is 55 psi. Using the "C" curve, the calculated pressure margin is 42 psi. This margin is greater than the design criteria minimum margin of 25 psi and is acceptable. The thermal-hydraulic limit, which is the minimum critical heat flux ratio (MCHFR), remains well above the minimum design criteria value of 1.0 in both cases.

These changes do not adversely affect the margins involved in the limiting accident assumed for establishing safety valve requirements, which involves closure of the main steam isolation valves with indirect scram from high neutron flux. The limiting accidents were analyzed assuming operation of the eight spring-loaded safety valves at the higher set points and operation of the relief/safety valve.

Using the "B" curve, the calculated peak pressure at the bottom of the reactor vessel is 78 psi below the 1375 psig allowed by ASME Boiler and Pressure Vessel Code Section III, the code section used and approved for vessel design. Using the "C" curve, the calculated pressure margin is 74 psi. These margins are approximately the same as those calculated in the initial Safety Analysis Report and are acceptable.

Commonwealth also presented the results of an analysis using the "C" curve and assuming operation of only the eight spring-loaded safety valves at the higher settings. The calculated peak pressure using these assumptions is only nine psi above that calculated using nine valves. Accordingly, even without the addition of the Target Rock safety/relief valve, the margin is still at least 65 psi and is not significantly different than that originally approved in the initial Safety Analysis Report for this facility. The added relief/safety valve which relieves at 1125 psig, through existing relief valve piping to the torus, fulfills a requirement of the ASME Boiler and Pressure Vessel Code which requires that the first safety valve relieve at a pressure corresponding to a peak reactor vessel pressure below design pressure.

Additional safety related concerns addressed by GE include the acceptability of the safety/relief valve and the structural adequacy of the piping and supports for the valve. The change from one electromagnetic relief valve to a Target Rock safety/relief valve does not involve safety considerations except as to pressure settings as discussed above. The proposed safety/relief valve is identical to that approved by the staff for use at other boiling water reactors, except that the flow capacity has been restricted to match that of the electromatic relief valve which it replaces. The structural adequacy of the piping and supports has been analyzed using dynamic analysis methods to assure that there is no adverse effect from the change.

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The proposed changes to the Technical Specifications include revised requirements on control rod scram times and safety valve settings. These revised requirements are consistent with the assumptions used in the design bases analyses and are, therefore, acceptable.

CONCLUSION

Based on the above, we have concluded that the proposed modification and amendment do not involve significant new safety information of a type not considered by any previous Commission safety review of the facility; potentially involve a significant increase in the probability or consequence of an accident considered in a previous Commission safety review of the facility; or involve a potentially significant decrease in the margin of safety during normal plant operations, anticipated operational occurrence, or postulated accidents considered in any previous Commission safety review of the facility and, therefore, do not involve a significant hazards consideration. We have further concluded that there is reasonable assurance that the health and safety of the public will not be endangered.

Richard D. Silver
Operating Reactors Branch #2
Directorate of Licensing

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

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

Date: MAY 24 1974

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COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

(DRESDEN UNIT 3)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-25

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated March 29, 1974, as supplemented April 18 and 22, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. 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;

   C. 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

   D. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, 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, attached to Facility Operating License No. DPR-25, are revised as indicated in the attachment to this license amendment. The Technical Specifications, as revised,

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are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

3. This license amendment is effective as of the date of its issuance.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

Karl R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing

Attachment:
Change No. 20 to Appendix A
Technical Specifications

Date of Issuance: MAY 24 1974

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ATTACHMENT TO LICENSE AMENDMENT NOS. 3 AND 8

CHANGE NOS. 20 AND 17 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NOS. DPR-25 AND DPR-29

1. The following change applies to Dresden Unit 3, License No. DPR-25:

   Replace existing pages 12, 18, 19, 20, 21, 58, 63, 78, and 90 with the attached revised pages bearing the same numbers.

2. The following change applies to Quad-Cities Unit 1, License No. DPR-29:

   Replace existing pages 14, 21, 24, 25, 26, 75, 76, 83, 100, 101, and 119 with the attached revised pages bearing the same numbers.

NOTE

On all of the revised pages, the changed areas are reflected by marginal lines.

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in detail (3). In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analysis show that even if the bypass system fails to operate, the design limit of MCHFR = 1.0 is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided with Dresden Units 2 and 3 has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

The proposed fuel operating conditions for Unit 3 reflect linear power generation rates and exposures higher than those experienced previously in BWR plants. Additional experimental data beyond that presented in Amendment 15 of the SAR will be obtained to further support the proposed combinations of fuel linear power generation rates and exposures, considering both normal and anticipated transient modes of operation. To develop these data for further assurance of fuel integrity under all modes of plant operation, a surveillance program on BWR fuel which operates beyond current production fuel experience will be undertaken. The schedule of inspections will be contingent on the availability of the fuel as influenced by plant operating and facility requirements. The program, as outlined in Amendment 17 of the SAR, will include surveillance of reactor plant off-gas activity, relevant plant operating data and fuel inspection.

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(3) SAR, Section 4.4.3 for turbine trip and load reject transients, Section 4.3.3 for flow control full coupling demand transient, and Section 11.3.3 for maximum feedwater flow transient.

also: "Dresden Second Reload License Submittal", transmitted on September 14, 1973, from Commonwealth Edison to J.F. O'Leary, U.S. Atomic Energy Commission.

also: "Dresden Station Special Report No. 29 Supplement B"

E. Turbine Stop Valve Scram - The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting at 10% of valve closure the resultant increase in surface heat flux is the same as for the load rejection and thus adequate margin exists. No perceptable change in MCHFR occurs during the transient. Ref. Section 11.2.3 SAR, "Dresden 3 Second Reload License Submittal," 9/14/73, and"Dresden Station Special Report No.29 Supplement B".

F. Generator Load Rejection Scram - The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCHFR from becoming less than 1.0 for this transient. For the load rejection from 100% power, the heat flux increases to only 106.5% of its rated power value which results in only a small decrease in MCHFR. Ref. Section 4.4.3 SAR., "Dresden 3 Second Reload license Submittal," 9/14/73, and "Dresden Station Special Report No. 29 Supplement B".

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel.

Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure there is no increase in neutron flux.

1.2 SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be ≤1060 psig.

B. Primary System Safety Valve Nominal Settings shall be as follows:

≤1 valve at 1125 psig*
≤2 valves at 1240 psig
≤2 valves at 1250 psig
≤2 valves at 1260 psig
≤2 valves at 1260 psig

The allowable setpoint error for each valve shall be ±1%.

*Target Rock combination safety/relief valve

Bases:

1.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, coolant system piping and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1175 psig at 560°F, and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and isolation condenser and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the isolation condenser and primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram, together with the turbine bypass system, limit the pressure to approximately 1100 psig (4). In addition, pressure relief valves have been provided to reduce the probability of the open discharge safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to a value (5) which is at least 25 psi below the setting of the first open discharge safety valve. Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves or turbine bypass system. Credit is taken for the neutron flux scram however.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

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(4) SAR Section 11.2.2. } also: "Dresden 3 Second Reload License Submittal", 9/14/73
(5) SAR Section 4.4.3. } also: "Dresden Station Special Report No. 29 Supplement B".

Bases:

2.2 In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 100% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high pressure scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is actually a backup protection to the high flux scram which was analyzed in Section 4.4.3 of the SAR, re-examined in the Dresden 3 Second Reload License submittal, September 14, 1973, and reanalyzed in "Dresden Station Special Report No.29 Supplement B". failure of the turbine stop valve closure scram, failure of the bypass system to actuate and failure of the relief valves to open) the pressure would rise rapidly due to void reduction in the core. A high pressure scram would occur at 1060 psig. The pressure at the bottom of the vessel is about 1163 psig when the first safety valve opens and about 1290 psig when the last valve opens. Both values are clearly within the code requirements. Vessel dome pressure reaches less than 1277 psig with the peak at the bottom of the vessel less than 1301 psig. Therefore, the pressure scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

3.3 LIMITING CONDITION FOR OPERATION

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3 SURVEILLANCE REQUIREMENT

C. Scram Insertion Times

1. After each refueling outage and prior to power operation with reactor pressure above 800 psig, all control rods shall be subject to scram-time tests from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. 25 of the operable control rods, selected to be uniformly distributed throughout the core, shall be scram-time tested at full reactor pressure at the time intervals listed below following any outage exceeding 72 hours in duration: 1 week, 2 weeks, 4 weeks, 8 weeks, 16 weeks and continuing at 16 week intervals:
   a) If the mean 90% insertion time of the tested control rod drives increases by more than 0.25 seconds or if the mean insertion time exceeds 3.5 seconds, then an additional sample of 25 control rods, selected to be uniformly distributed throughout the core, shall be scram tested. If the mean 90% insertion time of the 50 selected control rod drives exceeds 4.25 seconds, then all operable drives will be tested. Subsequent testing shall revert to the original 25 control rods at the 1 week, 2 week, etc., sequence interval; and

operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other then limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.0. Figure 3.5.2 of the SAR (1) & (2) shows the control rod scram reactivity used in analyzing the transients. Figure 3.5.2 (1) & (2) should not be confused with the total control rod worth, 18%Δk, as listed in some amendments to the SAR. The 18%Δk value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Figure 3.5.2 of the SAR (1) & (2) represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum amount of reactivity to be inserted during is controlled by permitting no more than 10% of the operable rods to have long scram times in the analytical treatment of the transients. 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds.

(1) For Cycle 2 of Dresden 3 and Cycle 3 of Dresden 2 Figure I-1 of Special Report No. 29

(2) For Cycle 3 of Dresden 3, Figure 1 of Dresden Station Special Report No. 29 Supplement B

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 thirty 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 are 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 low pressure each relief valve shall be manually opened until thermocouples downstream of the valve indicate fluid is flowing from the valve.

   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.

3.6 LIMITING CONDITION FOR OPERATION	4.6 SURVEILLANCE REQUIREMENT

an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320°F within 24 hours.

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (psig)
1	≤ 1125*
2	≤ 1240
2	≤ 1250
2	≤ 1260
2	≤ 1260

The allowable set point error for each valve is ±1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (psig)
1	≤ 1125*
2	≤ 1130
2	≤ 1135

*Target Rock combination safety/relief valve

COMMONWEALTH EDISON COMPANY AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

(QUAD-CITIES UNIT 1)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
License No. DPR-29

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee - acting for itself and as agent for Iowa-Illinois Gas and Electric Company) dated March 29, 1974, as supplemented April 22 and 24, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. 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;

   C. 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

   D. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, 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, attached to Facility Operating License No. DPR-29, are revised as indicated in the attachment to this license amendment. The Technical Specifications, as revised,

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DATE ➤						

are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

3. This license amendment is effective as of the date of its issuance.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

Karl R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing

Attachment:
Change No. 17 to Appendix A
Technical Specifications

Date of Issuance: MAY 24 1974

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SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NOS. 3 AND 8

CHANGE NOS. 20 AND 17 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NOS. DPR-25 AND DPR-29

1. The following change applies to Dresden Unit 3, License No. DPR-25:

   Replace existing pages 12, 18, 19, 20, 21, 58, 63, 78, and 90 with the attached revised pages bearing the same numbers.

2. The following change applies to Quad-Cities Unit 1, License No. DPR-29:

   Replace existing pages 14, 21, 24, 25, 26, 75, 76, 83, 100, 101, and 119 with the attached revised pages bearing the same numbers.

NOTE

On all of the revised pages, the changed areas are reflected by marginal lines.

1.1 Safety Limit Bases (cont'd)

psig or 5% flow. In general, Specification 1.1.B will only be applicable during startup, hot stand-by, or shutdown of the plant. A review of all the applicable low pressure and low flow data[1,2] has shown the lowest data point for transition boiling to have a heat flux of 144,000 Btu/hr/ft^2. To assure applicability to the Quad-Cities fuel geo-metry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., criti-cal heat flux was assumed to occur for these con-
3 ditions at 72,000 Btu/hr/ft^2. Assuming a peaking factor of 3.06 this is equivalent to a core average power of 460 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow con-ditions. For any greater pressure or flow conditions, there is increased margin.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail[3]. In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a signi-ficant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this speci-fication a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of MCHFR = 1.0 is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided with the Quad-Cities units has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will pro-vide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, con-sideration must also be given to water level

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(1) E.Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M.Becker, "Burnout Conditions for Flow of of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) SAR, Section 4.4.3 for turbine trip and load reject transients, Section 4.3.3 for flow control full coupling demand transient, and Section 11.3.3 for maximum feedwater flow transient.
also: "Dresden Station Special Report No. 29 Supplement B"

2.1 Limiting Safety System Setting Bases (cont'd)

subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would increase the capacity requirement for each of the ECCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of ECCS capacity requirements.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could lead to a loss of effective core cooling. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Scram - The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting at 10% of valve closure the resultant increase in surface heat flux is the same as for the load rejection and thus adequate margin exists. No perceptable change in MCHFR occurs during the transient. Ref. Section 11.2.3 SAR and Dresden Station Special Report No. 29 Supplemet

F. Turbine Control Valve Fast Closure Scram -
3 The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCHFR from becoming less than 1.0 for this transient. For the load rejection from 100% power, the heat flux increases to only 106.5% of its rated power value which results in only a small decrease in MCHFR. Ref. Section 4.4.3 SAR and Dresden Station Special Report No. 29 Supplement B.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs in the run mode when
4 the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation to Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided

1.2 SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor coolant high pressure scram shall be $\leq$1060 psig.

B. Primary system safety valve Nominal settings shall be as follows:

Unit 1	Unit 2
1 valve at 1125 psig*	
2 valves at 1240 psig	2 valves at 1210 psig
2 valves at 1250 psig	2 valves at 1220 psig
2 valves at 1260 psig	2 valves at 1230 psig
2 valves at 1260 psig	2 valves at 1240 psig

The allowable setpoint error for each valve shall be ± 1%.

*Target Rock combination safety/relief valve

1.2 Safety Limit Bases

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and coolant system piping. The respective design pressures are 1250 psig at 575°F, and 1175 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure
3 Vessel Code, Section III for the pressure vessel and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

3 The relationships of stress levels to yield strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram, together with the turbine bypass system limit the pressure to approximately 1100 psig[4]. In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram
13 limit the reactor pressure to a value (5,6,7 & 8) which is at least 25 psi below the setting of the first safety valve. Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves or turbine bypass system. Credit is taken for the neutron flux scram, however.

(4) SAR Section 11.2.2.

(5) SAR Section 4.4.3.

(6) Dresden 3 Special Report No. 29, "Transient Analysis for Cycle 2".

(7) Letter to D.J. Skovholt from J.S. [illegible], dtd 10/18/73, subj: Scram Reactivity Limitations for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2.

(8) Dresden Station Special Report No. 29 Supplement B.

1.2 Safety Limit Bases (cont'd)

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

2.2 Limiting Safety System Setting Bases

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high pressure scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is actually a backup protection to the high flux scram which was analyzed in Section 4.4.3 of the SAR, and re-examined for Unit 1 fuel cycle 2 in "Dresden Station Special Report No.29 Supplement B". If the high flux scram were to fail during a maximum pressure transient (also assuming failure of the turbine stop valve closure scram, failure of the bypass system to actuate and failure of the relief valves to open), the pressure would rise rapidly due to avoid reduction in the core. A high pressure scram would occur at 1060 psig.

Unit 1

The pressure at the bottom of the vessel is about 1163 psig when the first safety valve opens and about 1290 psig when the last valve opens. Both valves are clearly within code requirements. Vessel done pressure reaches less than 1277 psig with a peak at the bottom of vessel less than 1301 psig. Therefore, the pressure scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Unit 2

The pressure at the bottom of the vessel is about 1240 psig when the first vessel safety valve opens and about 1280 psig when the last valve opens. Both values are clearly within the code requirements.

Vessel dome pressure reaches about 1305 psig with the peak at the bottom of the vessel near 1330 psig. Therefore, the pressure scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

3.3 LIMITING CONDITION FOR OPERATION

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. Both RBM channels shall be operable; or .

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

13

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)	
5	Unit 1	Unit 2
20	0.375	0.375
50	0.900	0.900
90	2.00	2.00
	3.50	5.00

4.3 SURVEILLANCE REQUIREMENT

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

3

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3.3 LIMITING CONDITION FOR OPERATION

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)	
	Unit 1	Unit 2
5	0.398	0.398
20	0.954	0.954
50	2.120	2.120
90	3.800	5.300

13

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

10

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

4.3 SURVEILLANCE REQUIREMENT

10

2. Following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 week intervals, 50% of the control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the semiannual operating report to the AEC.

3.3 Limiting Condition for Operation Bases (cont'd)

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.
4 Section 7.4.5.3 of the SAR references the effects of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCHFR remains greater than 1.0. Figure 3.5-2 (1) of the SAR shows the control rod scram reactivity used in analyzing the transients. Figure 3.5-2 (1) should not be confused with the total control rod worth, 18%Δk, as listed in some amendments to the SAR. The 18%Δk value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Figure 3.5-2 (1) of the SAR represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the

(1) For Quad-Cities Unit 1 fuel cycle 2 Figure 1 of "Dresden Station Special Report No. 29 Supplement B"

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

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 is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding thirty days unless repairs are made and provided that during such time the HPCI subsystem is operable.

4 | D. Automatic Pressure Relief Subsystems

Surveillance of the automatic pressure relief subsystems 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 low pressure each relief valve shall be manually opened until thermocouples downstream of the valve indicate fluid is flowing from the valve.

   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.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem are made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, continued 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 4. 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.

E. Reactor Core Isolation Cooling System

1. The RCIC system shall be operable whenever the reactor pressure is greater than 150 psig, irradiated fuel is in the reactor vessel, and prior to startup from a Cold Condition.

4.5 SURVEILLANCE REQUIREMENT

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.

E. Reactor Core Isolation Cooling System

4 Surveillance of the RCIC System shall be performed as follows:

1. RCIC system testing shall be as specified in Specification 4.5.A.1.a, b, c, and d, except that the RCIC pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of 1150 psig to 150 psig, and a logic system functional test shall be run during each refueling outage.

3.6 LIMITING CONDITION FOR OPERATION	4.6 SURVEILLANCE REQUIREMENT

3.6 LIMITING CONDITION FOR OPERATION

E. Safety and Relief Valves

4 1. Prior to reactor startup for power operation and during reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

4 2. If Specification 3.6.E.1 is not met, the reactor shall remain shutdown until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (psig) Unit 1	Unit 2
1	1125*	
2	1240	1210
2	1250	1220
2	1260	1230
2	1260	1240

The allowable set point error for each valve is ±1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (psig)
1	1125*
2	1130
2	1135

* Target Rock combination safety/relief valve on Unit 1

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NOS. 50-249 AND 50-254

COMMONWEALTH EDISON COMPANY AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF FACILITY LICENSE AMENDMENTS

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment Nos. 3 and 8 to Facility Operating License Nos. DPR-25 and DPR-29 (respectively) to the Commonwealth Edison Company (and in the matter of License No. DPR-29, the Iowa-Illinois Gas and Electric Company) which revised Technical Specifications for operation of the Dresden Unit 3 (located in Grundy County, Illinois) and Quad-Cities Unit 1 (located in Rock Island County, Illinois).

The amendments permit replacement of one reactor coolant system electromatic relief valve with a combination safety/relief valve and other related changes.

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, and 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.

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For further details with respect to these actions, see (1) the application for amendment dated March 29, 1974, and supplements thereto dated April 18 (50-249 only), April 24 (50-254 only), and April 22, 1974, (2) Amendment Nos. 3 and 8 to License Nos. DPR-25 and DPR-29, with any attachments, and (3) the Commission's 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 at the Morris Public Library at 604 Liberty Street in Morris, Illinois 60451 (for those items relating to Dresden Unit 3), and at the Moline Public Library at 504 - 17th Street in Moline, Illinois 60265 (for those items relating to Quad-Cities Unit 1).

A copy of items (2) and (3) may be obtained upon request addressed to the Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this MAY 24 1974

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

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

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