MAR 20 1975

Docket No. 50-237

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 4 to Facility License No. DPR-19. This amendment includes Change No. 30 to the Technical Specifications and is in response to your request dated October 11, 1974.

The amendment allows operation with a combination safety/relief valve in place of an electromatic relief valve. Copies of the related Safety Evaluation and the Federal Register Notice relating to this action also are enclosed.

As at Dresden 3, you will limit power to 93% of full licensed level when the scram reactivity insertion rate is less than that of curve B on Figure 1 of "Dresden Station Special Report 29, Supplement B," dated March 29, 1974. The reduced power will assure that you maintain the design 25 psi minimum margin between the peak pressure and the safety valve settings during certain system transients. In this regard, we have imposed a limitation on power level which is prescribed in Paragraph 3.F of your facility license.

Sincerely,

Original signed By
Dennis L. Ziemann

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

C/P

Enclosures:
1. Amendment No. 4
   w/Change No. 30
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: see next page

FOR CONCURRENCES SEE PREVIOUS YELLOW

OFFICE	RL:ORB #2	RL:ORB #2	RL:ORB #2	OELD	RL:AD/ORs
SURNAME	RDSilver:aw/tc	RMDiggs	DLZiemann	Fred Bray	KRGoller
DATE	3/13/75	3/7/75	3/12/75	3/19/75	3/20/75

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

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The amendment allows operation with a combination safety/relief valve in place of an electromatic relief valve. Copies of the related Safety Evaluation and the Federal Register Notice relating to this action also are enclosed.

At 3950 MWd/T into this cycle as for Dresden 3, you will limit power levels to 93% of full licensed level. The reduced power will assure that you maintain the design 25 psi minimum margin between the peak pressure and the safety valve settings during certain system transients. In this regard, we have imposed a limitation on power level which is prescribed in Paragraph 3.F of your facility license.

Sincerely,

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

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1. Amendment No. 4
   w/Change No. 30
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

OFFICE ➤	L:ORB #2	L:ORB #2	L:ORB #2	OGC	L:AD/OR	
SURNAME ➤	RDiggs	RDSilver/tc	DLZiemann		KRGoller	
DATE ➤	1/2/75	1/24/75	1/30/75	1/7/75	1/ /75	

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

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of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

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Illinois Department of Public Health
Springfield, Illinois 62706

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
1 N. Wacker Drive, Room 822
Chicago, Illinois 60606

OFFICE ➤						
SURNAME ➤						
DATE ➤						

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

(DRESDEN UNIT 2)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-19

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

   A. The filing by the Commonwealth Edison Company (the licensee) dated October 11, 1974, 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, and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 3.B and 3.F of Facility License No. DPR-19 are hereby amended and added (respectively) to read as follows:

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"3.B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 33."

"3.F. Restrictions

At the point in operating cycle 4 when the reactivity insertion rate during a scram is less than that of curve B on Figure 1 of "Dresden Station Special Report 29, Supplement B," dated March 29, 1974, the reactor power level shall be restricted to 93% of rated power at 100% of rated core flow.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 30 to the
Technical Specifications

Date of Issuance: MAR 20 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 30 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-19

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

(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. See also NEDO-20547, General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Dresden Nuclear Power Station Unit 2 and Dresden Special Report No. 29 Supplement B.

E. Turbine Stop Valve Scram - The turbine stop valve scram like the lead 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 scarm 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 perceptible change in MCHFR occurs during the transient. Refer
30 to Section 11.2.3 SAR and Ref. (1) and (2).

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.
30 Refer to Section 4.4.3, SAR and Ref. (1) and (2).

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.

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(1) NEDO-20547, General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Dresden Nuclear Power Station Unit 2.

(2) Dresden Station Special Report No. 29 30
Supplement B.

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 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
30 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%.

30 *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 safe 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 safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to 1185 psig (5)-(7) which is 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.

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.

(5) SAR, Section 4.4.3.

(6) Special Report No. 29 and Supplement B thereto. 30

(7) NEDO-20547, General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Dresden Nuclear Power Station Unit 2.

Bases:

2.2 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 References (8) and (9). 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 void reduction in the core. A high pressure scram would occur at 1060 psig.
The pressure at the bottom of the vessel is
30 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
30 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.

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(8) SAR, Section 4.4.3.

(9) NEDO-20547, General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Dresden Nuclear Power Station Unit 2.

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT

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

30

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

30

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

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 levels. 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
30 3.5.2 of the SAR (1) & (2) shows the control rod scram reactivity used in analyzing the transients.
30 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
30 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.

30 (1) For Cycle 3, Fig. I-1 of Special Report No. 29.

(2) For Cycle 4, Fig. 1 of Dresden Station Special Report No. 29, Supplement B.

3.5 LIMITING CONDITION FOR OPERATION

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
30 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
30 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.

4.5 SURVEILLANCE REQUIREMENT

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.

30 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.

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

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 eight 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
30	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*
30	2	≤ 1130
	2	≤ 1135

30 *Target Rock combination safety/relief valve

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 TO LICENSE NO. DPR-19

(CHANGE NO. 30 TO THE TECHNICAL SPECIFICATIONS)

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT 2

DOCKET NO. 50-237

INTRODUCTION

By application dated October 11, 1974, Commonwealth Edison proposed a change in the Technical Specifications for Dresden Unit 2 to allow operation with a combination safety/relief valve in place of an electromatic relief valve. Related information was submitted in a letter of August 27, 1974, with enclosed report NEDO-20547, which requested authorization for operation during Cycle 4.

DISCUSSION

The scram reactivity curves for Dresden 2 at the end of fuel Cycle 4 will change such that the reactivity insertion rate will be slower than that at the end of the first fuel cycle as analyzed in the Final Safety Analysis Report. The change in scram reactivity insertion rate is the same as that previously evaluated for Dresden 3 in Reference 3. The change in scram reactivity insertion rate results in an increase in the peak pressure during pressurization events. The analysis of a turbine trip, assuming failure of the bypass system, is used to evaluate the adequacy of the relief valve system capacity. The analysis of turbine trip without bypass has shown that to maintain acceptable peak pressure margins, reduction in power level at the end of a fuel cycle or plant modifications are necessary. A plant modification has been proposed which reduces but does not eliminate the power level restrictions needed to maintain acceptable peak pressure margins. The proposed modification is the replacement of an electromatic relief valve with a combination safety relief valve. The transient following turbine trip with failure of bypass has been reanalyzed with the assumption that this plant modification has been completed. The results show that acceptable peak pressure margins are maintained for plant operation at 100 percent of rated power until the scram reactivity decreases to the generic B curve which has been

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calculated to occur at a fuel exposure increase of about 3950 MWD/T. Thereafter, acceptable margins are maintained for operation at 93 percent of rated power until the end of the fuel cycle. The power level reduction to 93 percent of rated at 100 percent flow reduces the steam flow rate, the average void fraction, and the average heat generation rate of the fuel. The reduction in average void fraction reduces the reactivity increase during the collapse of the voids during the turbine trip with failure of the bypass. All of these factors reduce the peak power and peak pressure sufficiently to compensate for the increases which would otherwise occur at the end of the fuel cycle because of the slower negative reactivity insertion of the control rods.

Therefore, the power level of Dresden Unit 2 will be restricted as a license condition to 93 percent of rated power at 100 percent of rated flow when the scram reactivity insertion rate is less than that of the generic B curve as presented in Reference 1. The power level at Dresden 2 was restricted during Cycle 3 for the same reason (Reference 2).

In the submittal dated October 11, 1974, CE proposed changes in the Technical Specifications for Dresden Unit 2 to allow operation with a combination relief/safety valve in place of an electromatic relief valve. The changes to the Technical Specifications include requirements for the modified valve, increased pressure setpoints for the spring-loaded safety valves, and more rapid scram times for the control rods. These changes were previously authorized for Dresden Unit 3 by Amendment No. 3 to Facility Operating License No. DPR-25, issued May 24, 1974. Since Dresdens 2 and 3 are of identical design in features relevant to the evaluation of the changes to the Technical Specifications and since the proposed changes to the Technical Specifications are identical to the Dresden 3 changes, the staff evaluation for Dresden 3 Amendment No. 3 is applicable to Dresden 2. The staff evaluation supporting Amendment No. 3 to DPR-25 is enclosed.

CONCLUSION

Based on the considerations discussed in our evaluation, we have concluded that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Enclosure: SER dated 5/24/74 (Reference 3).

Date: MAR 20 1975

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References

1. Dresden Station Report No. 29, Supplement B, Transient Analyses for Dresden 3 Cycle 3 and Quad Cities 1 Cycle 2. (March 29, 1974).

2. Change No. 25 to License No. DPR-19, Docket 50-237, issued December 27, 1973.

3. Safety Evaluation by the Directorate of Licensing Supporting Amendment Nos. 3 and 8 to License Nos. DPR-25 and DPR-29, issued May 24, 1974.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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 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.

One factor in the magnitude of the pressure and power transients following certain abnormal occurrences is the rate at which the reactor is shutdown; 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 85% 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 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.

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

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John I. Riesland

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Dennis L. Ziemann

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

Date: May 24, 1974

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF FACILITY LICENSE AMENDMENT

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 4 to Facility Operating License No. DPR-19 to the Commonwealth Edison Company which revised Technical Specifications for operation of the Dresden Nuclear Power Station Unit 2 located in Grundy County, Illinois. The amendment is effective as of its date of issuance.

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

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

For further details with respect to these actions, see (1) the application for amendment dated October 11, 1974, (2) Amendment No. 4 to License No. DPR-19 with Change No. 30, and (3) the Commission's concurrently issued related Safety Evaluation and the Safety Evaluation dated May 24, 1974, in Docket 50-249 on the same subject. All of these items

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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.

A single copy of items (2) and (3) may be obtained upon request addressed to the Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 20th day of March 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

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