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Regulatory File Cy.

April 18, 1974

Mr. J. F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Dresden Station Unit 3 - Proposed Change to Facility Operating License DPR-25, AEC Docket 50-249

Dear Mr. O'Leary:

Pursuant to Section 50.59 of 10 CFR 50 and Paragraph 3.B of Facility Operating License DPR-25, Commonwealth Edison submits a proposed change to Appendix A of DPR-25. The purpose of this proposed change is to incorporate into the Technical Specifications the changes discussed in Section 6 of "Dresden Station Special Report No. 29, Supplement B." The bases for this change are the analyses in Dresden Station Special Report No. 29, Supplement B. The proposed change is indicated on the attached revised pages 12, 18, 19, 20, 21, 58, 63, 78 and 90 of DPR-25 Appendix A (Technical Specifications).

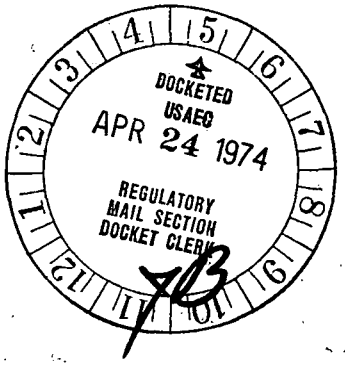
Three (3) signed originals and thirty-seven (37) copies are submitted for your review and approval.

Very truly yours,

J. S. Abel
Nuclear Licensing Administrator
Boiling Water Reactors

SUBSCRIBED and SWORN to
before me this 18th day
of April, 1974.

Notary Public



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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.

(Revised with Change 16 issued 3/25/74)

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.

(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".

(Revised with Change 16 issued 3/25/74)

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	2.2 LIMITING SAFETY SYSTEM SETTING

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 valves 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.

(Revised with Change 16 issued 3/25/74)

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 a value (5) 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.

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.

also: "Dresden 3 Second Reload License Submittal", 9/14/73

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 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, 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	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

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.

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

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
D. Automatic Pressure Relief Subsystems	D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:
1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel	1. During each operating cycle the following shall be performed: a. A simulated automatic initiation which opens all pilot valves, and b. With the reactor at 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. 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.	2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.
3. From and after the date that more than one of five **relief valves of the automatic** pressure relief subsystem 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.	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	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.	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
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.	*Target Rock combination safety/relief valve