Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

REGULATORY DOCKET FILE COPY

RECEIVED
SEP 21 1977
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

September 12, 1977

Mr. Edson G. Case, Deputy Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Unit 2
Proposed Amendment to Appendix A,
Technical Specifications, for
Facility Operating License DPR-19
to Support Reload No. 3
NRC Docket No. 50-237

Dear Mr. Case:

Pursuant to 10 CFR 50.59, Commonwealth Edison Company proposes to amend Appendix A Technical Specifications to Facility Operating License No. DPR-19 to support core Reload No. 3 at Dresden Station Unit 2. These changes are based on the plant specific analyses summarized in Enclosure (1) to this letter.

For Reload No. 3, it is planned to load 192 8D250 bundles having a bundle average enrichment of 2.50 wt. % U-235.

The plant specific analyses reveal that for 8 x 8 fuel, the most limiting transient is the turbine trip without bypass valves at EOC and for the 7 x 7 fuel, the most limiting transient is the rod withdrawal error. The Minimum Ciritcal Power Ratio (MCPR) proposed for Cycle 4 is 1.27 for 8 x 8 fuel and 1.39 for 7 x 7 fuel.

Revised wording in the bases for the safety valve sizing transient are proposed for greater clarity and to provide consistency with other utilities.

These proposed changes have received off-site and on-site review and approval.

For the purposes of your schedule, the projected startup date for this unit is December 7, 1977.

Please address any questions on this matter to this office.

7726400033

Three (3) signed originals and 37 copies of this letter are provided for your use.

Very truly yours,

M.S. Turbak
for R. L. Bolger
Assistant Vice President

SUBSCRIBED and SWORN to
before me this 13th day
of Sept., 1977.

Notary Public

Enclosures: (1) General Electric Boiling Water Reactor Reload - 3 Licensing Submittal for Dresden Nuclear Power Station Unit 2 (NEDO-24034, 77NED110, June 1977)

(2) Amended page 3 of license DPR-19 and pages 20, 21, 42, 81D and 85B

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 30, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

Commonwealth Edison shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Commonwealth Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. For the purpose of repairing a crack in the recirculation bypass line in the "A" loop, the licensee may perform the repair program as described in a report entitled "Commonwealth Edison Company Dresden Station 2A Recirculation Pump 4 Equalizing Line Repair Program" transmitted by letter dated September 23, 1974.

F. Restrictions

Reactor power level shall be limited to maintain pressure margin to the safety valve setpoints during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 3 licensing submittal for Dresden Unit 2 (NEDO-24034). Plant operation shall be limited to the operating plan described therein with subsequent operation in the coastdown mode permitted to 40% power.

G. Equalizer Valve Restriction

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

4. This license is effective as of the date of issuance and shall expire eighteen (18) months from said date, unless extended for good cause shown, or upon the earlier issuance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris

Peter A. Morris, Director

...ment
...dix A - Technical Specifications

...of Issuance: DEC 2 2 1969

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 a value 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 during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram however. The pressure at the bottom of the vessel peaks at less than 1325 psig. The indirect flux scram and safety valve actuation, therefore, provide adequate margin below the peak allowable vessel pressure of 1375 psig.

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

---

(4) SAR, Section 11.2.2.

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 neutron flux 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 available as a backup protection to the high flux scram which was analyzed in References (5) and (6) and is reexamined in the Reload Licensing Submittal for each subsequent cycle. If the high flux scram were to fail, a high pressure scram would occur at 1060 psig.

(5) SAR, Section 4.4.3.

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

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias)(7)	$\le [.65W + 43] \left[\frac{LTPF}{TPF}\right]$ (2)
* 1	APRM upscale (refuel and Startup/Hot Standby mode)	$\le$12/125 full scale
2	APRM downscale (7)	$\ge$3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	$\le [.65W + 40]$ (2)
1	Rod block monitor downscale (7)	$\ge$5/125 full scale
3	IRM downscale (3)	$\ge$5/125 full scale
3	IRM upscale	$\le$108/125 full scale
* 3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\le 10^5$ counts/sec

K. Minimum Critical Power Ratio (MCPR)

During steady state operation, MCPR shall be greater than or equal to -

Unit 2

1.39 (7 x 7 fuel)

1.27 (8 x 8 fuel)

at rated power and flow. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of K_f, where K_f is as shown in Figure 3.5-2.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, the MCPR shall be 1.32 times K_f where K_f is as shown in Figure 3.5-2.

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

heat generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on that presented in Ref. (2) and assumes a linearly increasing variation in axial gaps between core bottom and top, and assumes with 95% confidence, that no more than one fuel rod exceeds the design LHGR due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining $\Delta P/P$ in accordance with Refs. (3) and (4).

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this Specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, a MCPR of 1.18, is satisfied. For any of the special set of transients or disturbance caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. The results apply with increased conservatism while operating with MCPR's greater than specified.

The most limiting transients with respect to MCPR are generally:

(a) Rod withdrawal error
(b) Turbine trip without bypass
(c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles Reload Licensing Submittal specifies the limiting transient for each fuel type.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the Specification. This assure that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

---

(2) Fuel Densification Effects on General on General Electric Boiling Water Reactor Fuel," Section 3.2.1, Supplement 6, Aug. 1973.

(3) USAEC Report, "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Dec. 14, 1973.

(4) GE Planning and Development Memorandum #45, "Length Growth of BWR Fuel Elements", R. A. Proebsthe, October 1, 1973 (Proprietary).