MAR 25 1974

Docket No. 50-249

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

Change No. 16
License No. DPR-25

Gentlemen:

By application dated September 14, 1973, as supplemented, you requested authorization to operate Dresden 3 with Reload 2 fuel bundles and proposed related changes to the Technical Specifications. The supplements reviewed as part of this action are your letters dated November 27, December 6 (2 letters), December 17, 1973 (2 letters), January 9 (2 letters), January 18 and January 23, 1974.

The use of 8 x 8 fuel in reloads has been reviewed on a generic basis by the licensing staff and the Advisory Committee on Reactor Safeguards (ACRS). The reports based on these reviews were transmitted to you by letters dated February 11 and February 20, 1974. The staff Safety Evaluation of the use of 8 x 8 fuel and the additional matters specifically related to your request ~~was~~ were transmitted to you by letter dated March 15, 1974. Based on these reviews, we have concluded that the health and safety of the public will not be endangered by the proposed refueling and subsequent operation with Reload 2 and with the proposed modifications to the technical specifications. Your submittal states that at an exposure corresponding to approximately four months operation after startup, the maximum steady state power will be limited to 97% power and approximately five months later the power will be further reduced. The power reductions with increasing exposure will be made to acceptably limit calculated primary system pressure increase from a postulated operational transient. We request that you notify us upon reaching the exposure at which power will be limited to 97%. Prior to reaching an exposure requiring a further power reduction, approximately nine months after startup, we request that you submit a supplemental analysis which will be applicable for the remainder of the cycle.

A notice of proposed issuance of these changes to the technical specifications was published in the Federal Register on February 13, 1974, for a 30-day notice period. The notice period has expired without intervention. Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License No. DPR-25 are hereby changed by replacing pages 12, 18, 20, 21, 42, 48, 81B, 81C, 85A, 85B, 85C and 157 with the revised pages 12, 18, 20, 21, 42, 48, 81B, 81C, 85A, 85B and 157 appended hereto.

As required by 10 CFR Part 2, the enclosed notice relating to the issuance of this change is being filed with the Office of the Federal Register for publication.

Sincerely,

/S/

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Revised pages
2. Federal Register Notice

cc w/enclosures:

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WOMiller
PCollins
DLZiemann
RDSilver
RMDiggs
NDube
MJinks
BScharf (15)
SKari
J. Riesland
A. Giambusso

I informed Jim Abel of CE 4:30 PM on 3/25 that this was signed. R. Silver

OFFICE ➤	L:ORB #2	L:ORB #2	L:ORB #2	L:RS	OGC	L:OR
SURNAME ➤	RDSilver:rwg	RMDiggs	DLZiemann	VStello OK for	Gallo	DJSkovholt
DATE ➤	3/22/74	3/22/74	3/22/74	3/22/74	3/25/74	3/[illegible]/74

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.

(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.

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.

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.

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.

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 safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to 1180 psig (5) which is 30 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

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

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 and re-examined in the Dresden 3 Second Reload License submittal, September 14, 1973.

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 1240 psig when the first 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.

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

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias)(7)	$\leq$ 0.650W + 43(2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$ 12/125 full scale
2	APRM downscale (7)	$\geq$ 3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	$\leq$ 0.650W + 42 (2)
1	Rod block monitor downscale (7)	$\geq$ 5/125 full scale
3	IRM downscale (3)	$\geq$ 5/125 full scale
3	IRM upscale	$\leq$ 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	$\leq 10^5$ counts/sec

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

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32" of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves; i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCHFR does not decrease to 1.0. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCHFR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCHFR is maintained greater than 1.0

The APRM rod block which is set at 12% of rated power is functional in the refuel and Startup/Hot Standby mode. This control rod block provides the same type of protection in the refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the Run mode; i.e., it prevents MCHFR from decreasing below 1.0 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked when MCHFR is ~1.16, thus allowing adequate margin. Ref. Section 7.4.5.3 SAR. Below ~70% power the worst case withdrawal of a single control rod results in a MCHFR >1.0 without rod block action, thus below this level it is not required.

I. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1.

J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{LT}\right)\right]$$

$LHGR_d$ = Design LHGR = 17.5 KW/ft, 7 x 7 fuel
= 13.4, 8 x 8 fuel

$\left(\Delta P/P\right)_{max}$ = Maximum power spiking penalty = 0.036 for 7 x 7 fuel and 0.026 for 8 x 8 fuel

LT = Total core length = 12 ft

L = Axial position above bottom of core

I. Average Planar LHGR

Daily during reactor power operation, the average planar LHGR shall be checked.

J. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.

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

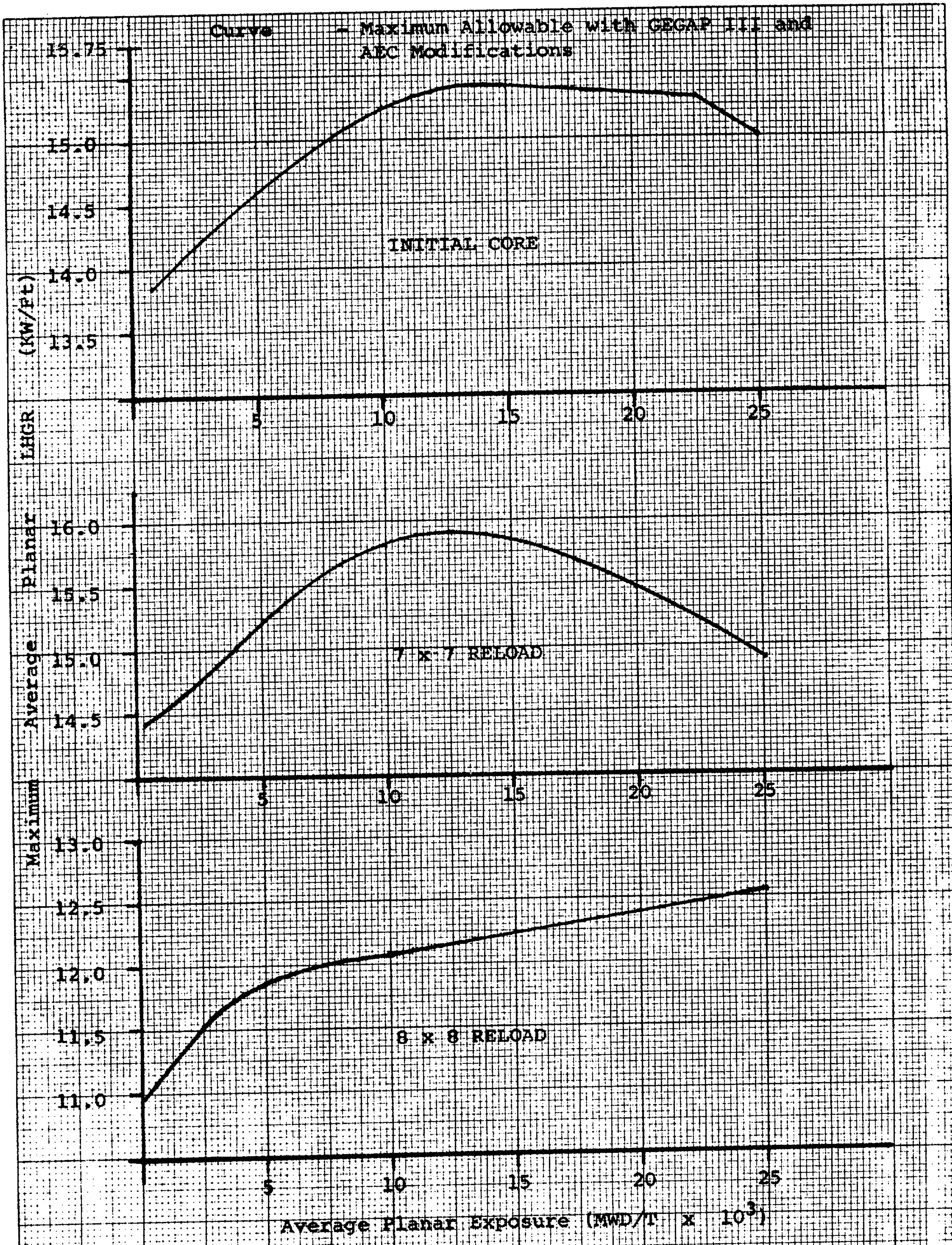

FIGURE 3.5.1 - MAXIMUM ALLOWABLE PLANAR LHGR

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

3.5.I Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300^{o}F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribtuion within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20^{o}F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the IAC limit.

The maximum average planar LHGR's shown in Figure 3.5.1 are based on calculations employing the models described in the General Electric Report NEDM-10735 as modified by General Electric Report NEDO-20181 including modifications made by the Atomic Energy Commission transmitted to Commonwealth by letter dated December 5, 1973.

3.5.J Local LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE Topical Report NEDM-10735 Supplement 6, and assumes a linearly increasing variation

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

in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. An irradiated growth factor of 0.25% was used as the basis for determining $\Delta P/P$ in accordance with General Electric Development and Planning Memorandum #45, "Length Growth of BWR Fuel Elements," R. A. Proebsthe, October 1, 1973 and U.S. Atomic Energy Commission report, "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," December 14, 1973.

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

5.0 DESIGN FEATURES

5.1 Site

Dresden Unit 3 is located at the Dresden Nuclear Power Station which consists of a tract of land of approximately 953 acres located in the northeast quarter of the Morris 15-minute quadrangle (as designated by the United States Geological Survey), Goose Lake Township, Grudy County, Illinois. The tract is situated in portions of Sections 25, 26, 27, 34, 35, and 36 of Township 34 North, Range 8 East of the Third Principal Meridian.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies.

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.90.

5.6 Seismic Design

The reactor building and all contained engineered safegards are designed for the maximum credible earthquake ground motion with an acceleration of 20 per cent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

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

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-249

COMMONWEALTH EDISON COMPANY
DRESDEN NUCLEAR POWER STATION UNIT 3

NOTICE OF ISSUANCE OF CHANGES TO
TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSE

No request for a hearing or petition for leave to intervene having been filed following publication of the notice of proposed action in the Federal Register on February 13, 1974 (39 F.R. 5527), the Atomic Energy Commission (the Commission) has issued Change No. 16 to the Technical Specifications of Facility Operating License No. DPR-25 to the Commonwealth Edison Company (the licensee). This change, effective immediately, authorizes the licensee to operate the Dresden Nuclear Power Station Unit 3 (the facility) using 8 x 8 fuel (containing uranium 235) and changes the limiting conditions for operation associated with fuel densification for the 8 x 8 and 7 x 7 fuels. The licensee is presently authorized to possess and operate its facility located in Grundy County, Illinois, at power levels up to 2527 MWt using a full core of 7 x 7 fuel (containing uranium 235).

The Commission has found that the application for the above action dated September 14, 1973, as supplemented by filings dated November 27, 1973, December 6 and 17, 1973, and January 9, 18 and 23, 1974, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission regulations published in 10 CFR Chapter I. On March 15, 1974, the Commission's Directorate of Licensing completed its evaluation of the action and issued a Safety Evaluation concluding that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the facility with the 8 x 8 fuel and the related changes to the Technical Specifications as authorized by Change No. 16.

A copy of Change No. 16 to the Technical Specifications of Facility Operating License No. DPR-25, the Directorate of Licensing's Safety Evaluation dated March 15, 1974, the Technical Report on the General Electric Company 8 x 8 assembly by the Directorate of Licensing dated February 5, 1974, and the Report of the Advisory Committee on Reactor Safeguards dated February 12, 1974, on the subject of operation of boiling water reactors with 8 x 8 fuel bundles are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C., and at the Morris Public Library at 604 Liberty Street in Morris, Illinois 60670. Single copies of these items may be obtained upon request sent to the Deputy Director for Reactor Projects, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D. C. 20545.

FOR THE ATOMIC ENERGY COMMISSION

/S/

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

Dated at Bethesda, Maryland,
this MAR 25 1974