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DEC 26 1974

Docket No. 50-237

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. 7 to Facility License No. DPR-19 for Dresden Unit 2. This amendment includes Change No. 33 to the Technical Specifications and is in response to your request dated August 27, 1974, and supplements thereto dated October 10 and 22, November 7 and 22 and December 5, 1974.

This amendment authorizes operation of Dresden Unit 2 using a partial reload containing 7 x 7 and 8 x 8 fuel assemblies, deletes the restriction imposed by Amendment 5, Change 31 for operation with 8 x 8 fuel and approves technical specification changes related to (1) the reload, (2) the core thermal safety limit, and (3) limiting safety system settings, limiting conditions of operation, and surveillance requirements related to fuel cladding integrity.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

original signed by Fredric D. Anderson

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

Enclosures:
1. Amendment No. 7
   w/Change No. 33
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See attached

OFFICE ▶	L:ORB-2	L:ORB-2	L:ORB-2	OGC	L:AD/ORs	L:RP
SURNAME ▶	x7403:esp RSilver	RMDiggs	DLZiemann		for G. Lear KRGoller	AGiambusso
DATE ▶	12/23/74	12/23/74	12/23/74	12/2/74	12/24/74	12/26/74

Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43-16-81465-1 445-678

cc w/encls:
John W. Rowe, Esquire
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza
Chicago, Illinois 60670

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

Chairman, Board of Supervisors
of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

cc w/encls and filings dtd 10/22/74, 11/7 & 22/74 and 12/5/74:
Mr. Leroy Stratton
Bureau of Radiological Health
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

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COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-19

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated August 27, 1974 and supplemented by filings dated October 10 and 22, November 7 and 22 and December 5, 1974, comply 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;

   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; and

   E. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on October 30, 1974, (39 FR 38274).

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Facility License No. DPR-19 is hereby amended by deleting Paragraph 3.F, and by changing Paragraph 3.B to read as follows:

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

FOR THE ATOMIC ENERGY COMMISSION
Original signed by
Roger S. Boyd
for A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Change No. 33 to the
Technical Specifications

Date of Issuance: DEC 26 1974

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ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 33 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-19

Delete pages 2, 4, 5, 6, 7, 9, 11, 12, 14, 16, 16A, 17, 18, 20, 21, 22, 34, 48, 81B, 81C, and 85B from the Technical Specifications and insert the attached replacement pages bearing the same number(s) and the additional page 5A. The changed areas on the revised pages are shown by a marginal line.

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and abnormal situations can be safely controlled.

I. Limiting Safety System Setting (LSSS) – The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

J. Limiting Total Peak Factor – The Limiting Total Peaking Factor (LTPF) is the lowest Total Peaking Factor which
33 limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the operating limit at 100% power.

K. Logic System Functional Test – A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

L. Minimum Critical Heat Flux Ratio (MCHFR) – The lowest in-core ratio of critical heat flux (that heat flux which results in transition boiling) to the actual heat flux.

M. Mode – The reactor mode is that which is established by the mode-selector-switch.

N. Operable – A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

O. Operating – Operating means that a system or component is performing its intended functions in its required manner.

P. Operating Cycle – Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

Q. Primary Containment Integrity – Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

R. Protective Instrumentation Definitions

1. Instrument Channel – An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

Y. Secondary Containment Integrity – Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or are secured in the isolated position.

Z. Shutdown – The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

AA. Simulated Automatic Actuation – Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

BB. Total Peaking Factor - The Total Peaking Factor (TPF) is the highest product of radial, axial, and local peaking factors simultaneously operative at any segment of fuel rod. 33

CC. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is greater than 600 psig the combination of recirculation flow and reactor thermal power-to-water shall not exceed the limit shown in Figure 1.1.1. The safety limit is exceeded when the recirculation flow and thermal power-to-water conditions result in a point above or to the left of the limit line.

B. When the reactor pressure is less than 600 psig or recirculation flow is less than 5% of design, the reactor thermal power-to-water shall not exceed 460 MW(t).

C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds

2.1 FUEL CLADDING INTEGRITY

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:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - When the reactor mode switch is in the run position, the APRM flux scram setting shall be as shown in Figure 2.1.1 unless the combination of power and 33
peak LHGR is above the curve in Figure 2.1.2. When the combination of power and peak LHGR is above the 33
curve in Figure 2.1.2 a scram setting(s) as given by:

$$S \leq [.65W + 55] \left[\frac{LTPF}{TPF}\right]$$ 33

where:

S = setting in per cent of rated power
W = recirculation loop flow in per cent of rated flow

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING
	TPF = LTPF unless the combination of power and peak LHGR is above the curve in Figure 2.1-2 at which point the actual peaking factor value shall be used. 33 LTPF = 3.05 for 7 x 7 fuel LTPF = 3.01 for 8 x 8 fuel 2. APRM - When the reactor mode switch is in the start-up/hot standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

the scram setting established by Specification 2.1.A and a control rod scram does not occur.

D. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.

3. IRM - The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.1.1 unless the combination of power and peak LHGR is above the curve in Figure 2.1.2. When the combination of power and peak LHGR is above the curve in Figure 2.1.2 a rod block trip setting (S_{RB}) as given by:

$$S_{RB} \leq [.65W + 43] \left[\frac{LTPF}{TPF}\right]$$ 33

where:

the definitions used for the APRM scram trip apply.

C. Reactor Low Water Level Scram setting shall be ≥143" above the top of the active fuel at normal operating conditions.

D. Reactor Low Low Water Level ECCS initiation shall be 83" $\left(^{+4''}_{-0''}\right)$ above the top of the active fuel at normal operating conditions.

E. Turbine Stop Valve Scram shall be ≤10% valve closure from full open.

F. Generator Load Rejection Scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main Steamline Isolation Valve Closure Scram shall be ≤10% valve closure from full open.

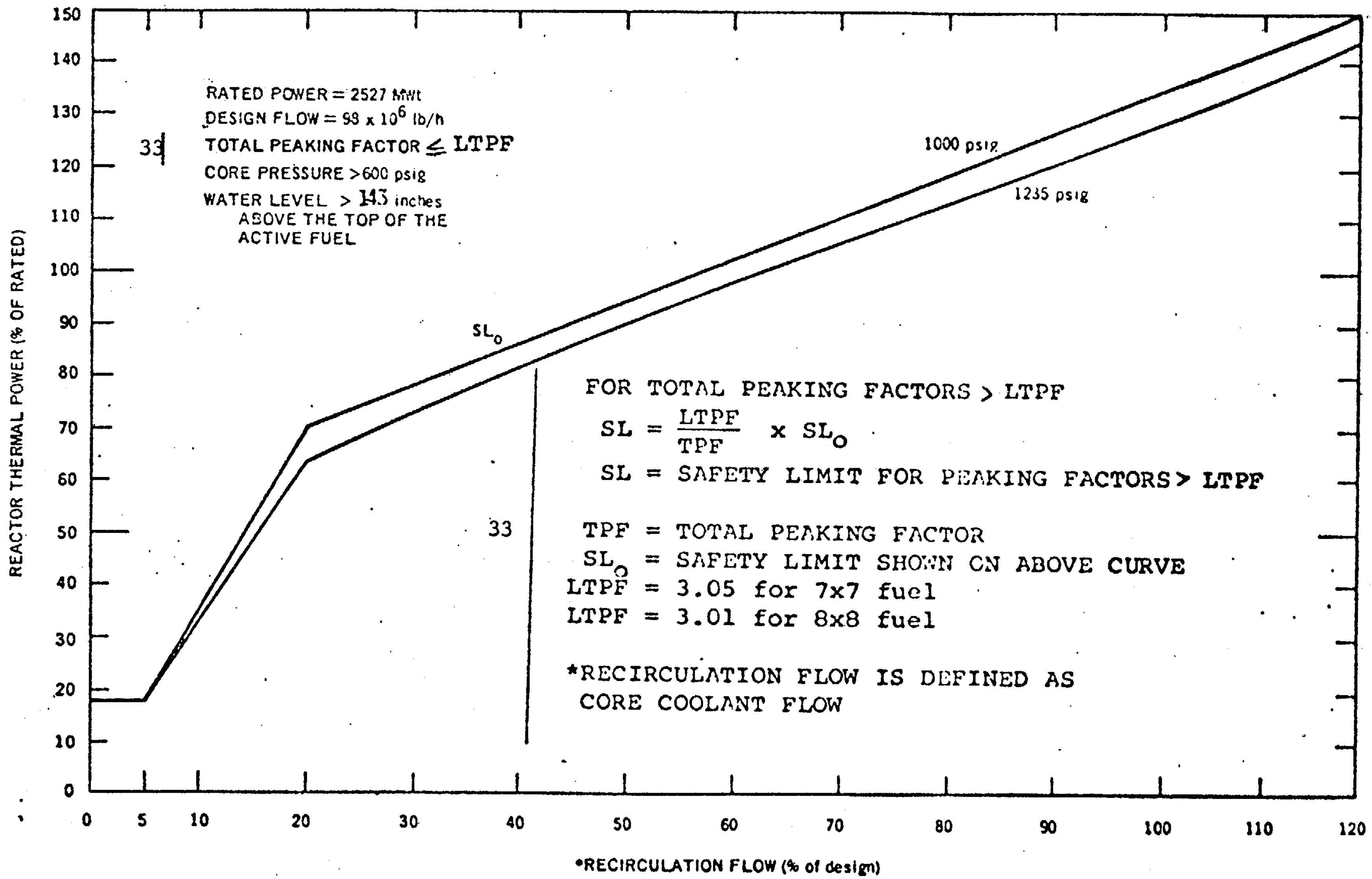

Figure 1.1.1. Core Thermal Safety Limit

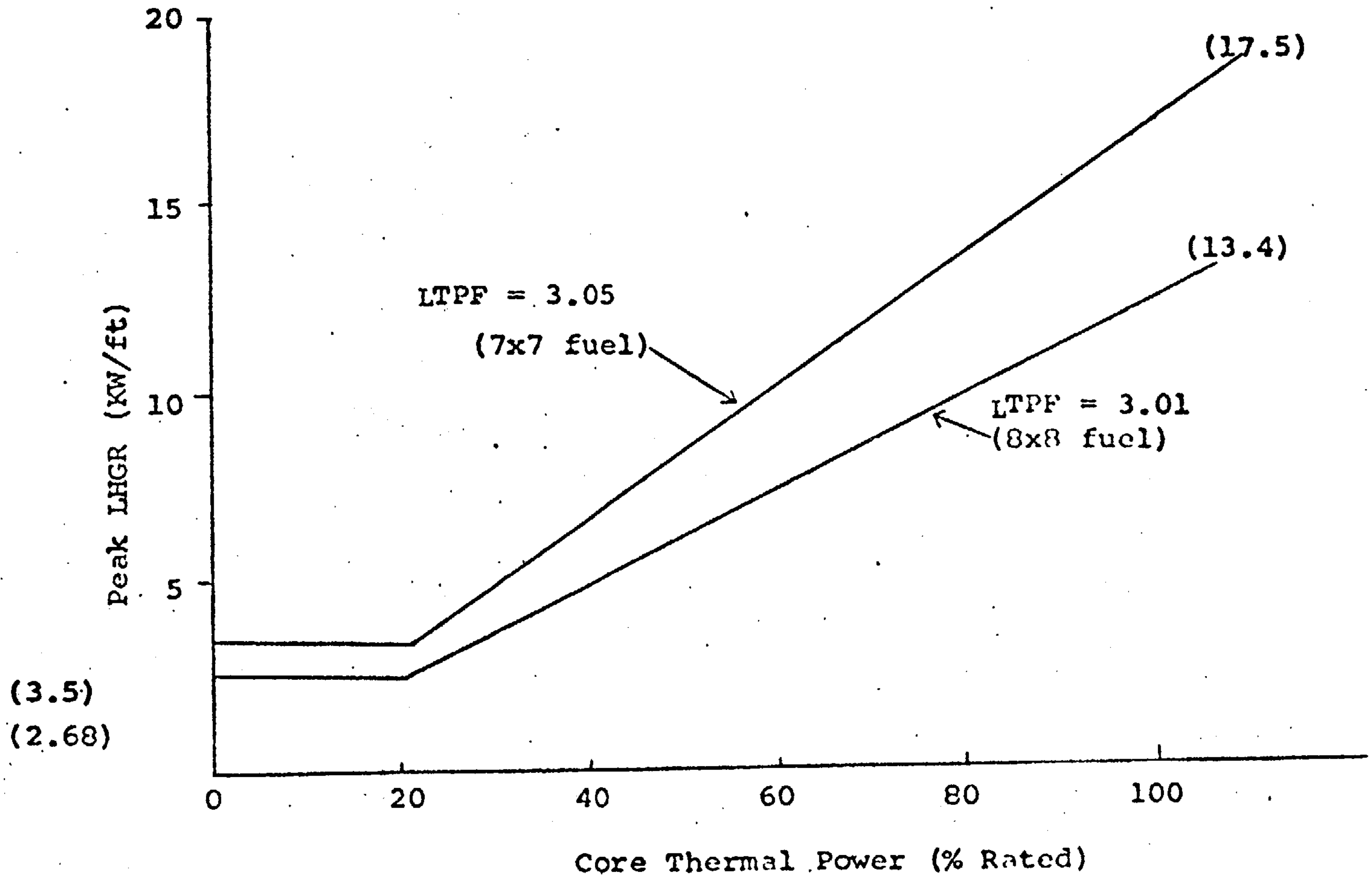

Figure 2.1-2 Peak LHGR Versus Core Thermal Power for a Limiting Total Peaking Factor (LTPF)

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is based on a pressure of 1235 psig. In no case is reactor pressure ever expected to exceed 1250 psig, and therefore, the curves will cover all operating conditions with mere interpolation. If reactor pressure should ever exceed 1250 psig during power operation, it would be assumed that the safety limit has been violated. For pressures between 600 psig, which is the lowest pressure used in the critical heat flux data, and 1000 psig, the upper curve is applicable with increased margin.

The power shape assumed in the calculation of these curves was based on design limits and results
33 in a Limiting Total Peaking Factor (LTPF). For any peaking of smaller magnitude, the curves are conservative. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core Local Power Range Monitor (LPRM) System. However, to maintain applicability of the safety limit curve, the safety limit will be lowered according to the equation given on Figure 1.1.1 in the rate event of power operation with a
33 total peaking factor in excess of the Limiting Total Peaking Factor.

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows which is 348°F for rated thermal power. For any lower feedwater temperature, subcooling is increased and the curves are conservative.

The water level assumed in the calculation of the safety limit was that level corresponding to the bottom of the steam separator skirt (0" on the level instrument and approximately 12' above the top of the active fuel). This point is below the water level scram setpoint. As long as the water level is above this point the safety limit curves are applicable; i.e., the amount of steam carry under would not be increased and therefore the core inlet enthalpy and subcooling would not be influenced.

The values of the parameters involved in Figure 1.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the Average Power Range Monitor (APRM) in-core nuclear instrumentation is calibrated to read in terms of percent rated power.

The range in pressure and flow used for Specification 1.1.A was 600 psig to 1250 psig and 5% to 100%, respectively. Specification 1.1.B provides a requirement on power level when operating below 600 psig or 5% flow. In general, Specification 1.1.B will only be applicable during startup, hot standby, or shutdown of the plant. A review of all the applicable low pressure and low flow data (1, 2) has shown the lowest data point for transition boiling to have a heat flux of 144,000 $Btu/hr/ft^2$. To assure applicability to the Dresden 3 fuel geometry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., critical heat flux was assumed to occur for these conditions at 72,000 $Btu/hr/ft^2$. Assuming a peaking factor of 3.0, this is equivalent to a core average power of 460 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions, there is increased margin.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed

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

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

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.

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actually conducted from rated power but with the conservative void coefficient.

Inherent in these analyses is the fact that steady-state operation without forced recirculation flow will not be permitted except during startup testing.

In summary, the transients presented in the SAR were analyzed only up to the design flow control line and not above because:

1. The licensed maximum steady-state power level is 2527 MWt.

2. The units cannot physically be brought above 2527 MWt unless abnormal operation is employed.

3. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

4. The analysis model itself is demonstrated to be conservative.

5. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher strating power, which has been shown above to be unrealistic, than using values for the parameters.

A. Neutron Flux Scram — The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent power. Since fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients with an APRM scram setting as shown in Figure 2.1.1, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analysis reported in the SAR demonstrates that, even with a fixed 120% scram trip setting, none of the postulated transients result in violation of the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin. See page 15 for further comparison.

An increase in the APRM scram setting to greater than that shown in Figure 2.1.1 would decrease the margin present before the thermal hydraulic safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. A reduction in this operating margin would increase the frequency of spurious scrams which have an adverse affect on reactor safety because of unnecessary thermal stress which it causes. Thus, the APRM setting was selected because it provides adequate margin from the thermal hydraulic safety limit yet allows operating margin which minimizes unnecessary scrams.

The thermal hydraulic safety limit of Specification 1.1 was based on the Limiting Total Peaking Factor. A factor has been included on Figure 1.1.1 to adjust the safety limit in
the event peaking factor exceeds the Limiting 33
Total Peaking Factor. Likewise, the scram setting should also be adjusted to assure MCHFR does not become less than 1.0 in this degraded situation. This has been accomplished by use of Figure 2.1.2. If the combination of power and LHGR is greater than that shown by the curve, the APRM scram setting is adjusted downward by formula given in the specification. The scram setting as given by

the equation will prevent MCHFR from becoming less then 1.0 for the given heat flux condition for the worst expected transients. If the APRM scram setting should require a change due to an abnormal peaking condition, it will be done by changing the intercept point and thus, the entire flow bias curve will be shifted down. Below 20% power the peak LHGR normally will be less than or equal to 20% power value. However, if the peak LHGR below 20% power exceeds the 20% power value, the APRM scram and rod block settings shall be lowered by the formula in the specifications. The above safety margins are not significantly
33 reduced because power maneuvers below 20% power are restricted to control rod movements due to the protective interlocks limiting recirculation pump operation to minimum speed. During this period flow increases inherently occur with power increases, even with no recirculation pumps operating. Pump operation enhances this phenomenon at minimum pump speed. Since TPF improves with nearly every rod withdrawal, and power ascension must be accomplished by slow rod withdrawal, the specification provides operational flexibility while still maintaining adequate margin to the Safety Limit.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 18% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern.

Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analyses of transients from this operating condition are less severe than the same transients from the two pump operation.

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels, arranged in the core as shown in Figure 7.4.4 of the FSAR. The IRM is a 5 decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being 1/2 of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise, if the instrument were on range 5, the scram setting would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. In the start-up/hot standby mode, a scram at 120 divisions on the instrument is less than 15% power, except for range 10 on the instrument. Thus, the scram setting on the IRM is also less than the 15% scram on the APRM, except in the 10th range. The IRM scram provides protection for changes which occur, both locally and over the entire core. The IRM, because of the scram arrangement discussed above, thus provides additional or back-up protection to the APRM 15% scram in the start-up and hot standby mode. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit or the APRM 15% scram occurred. For the case of a single control rod withdrawal error this transient has been analyzed in Section 7.4.4.3 of the FSAR. In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in Section 7.4.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining heat flux within those values specified in the safety limit for this condition of plant operation. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides back-up protection for the APRM.

APRM Control Rod Block Trips — Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against exceeding a MCHFR of unity. This rod block setpoint, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable setpoint provides substantial margin from fuel damage, assuming steady-state operation at the setpoint, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship; therefore, the worst case MCHFR during steady-state operation is at 108% of rated power. Peaking factors as specified in Section 3.2.5 of the SAR were considered. The total peaking factor was 3.0. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors are greater than the Limiting
3 Total Peaking Factor. This assures rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by changing the intercept point of the flow bias curve; thus, the entire curve will be shifted downward.

C. Reactor Low Water Level Scram — The reactor low water level scram is set at a point which will assure that the water level used in the safety limit is maintained. The bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

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

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

E. Turbine Stop Valve Scram - The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a 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
33 to Section 11.2.3 SAR and Ref. (1).

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

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 providin 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 b the IRM high neutron flux scram. Thus, the combi of main steam line low pressure isolation and isolati 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 33
Submittal for Dresden Nuclear Power Station Unit 2.

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 33
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 fro 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.

(7) NEDO-20547, General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Dresden Nuclear Power Station Unit 2. 33

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 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.

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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.

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The response times of the individual functions shall not exceed 0.10 second.

B. During operation with greater than a Limiting Total Peaking Factor, either:

   a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B; or

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   b. The power distribution shall be changed such that a total peaking factor greater than the Limiting Total Peaking Factor no longer exists.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Daily during reactor power operation, the peak LHGR shall be determined and the core power distribution shall be checked 33
for Limiting Total Peaking Factor.

a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Nework, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Tank, Main Steam Line Isolation Valve Closure, Generator Load Rejection, and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving, e.g. the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The peak LHGR shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the peak LHGR is adequate.

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.i., 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 33 rod withdrawl is blocked when MCHFR is >1.0, 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.

3.5 LIMITING CONDITIONS FOR OPERATION

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.

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

$LHGR_d$ = Design LHGR

= 17.5 kw/ft, 7X7 fuel assemblies

33 = 13.4 kw/ft, 8X8 fuel assemblies

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= .037 initial core fuel

= .026 reload 1, 7X7 fuel

33 = .021 reload 1, 8X8 fuel

L_T = Total Core Length = 12 ft

L = Axial distance from bottom of core

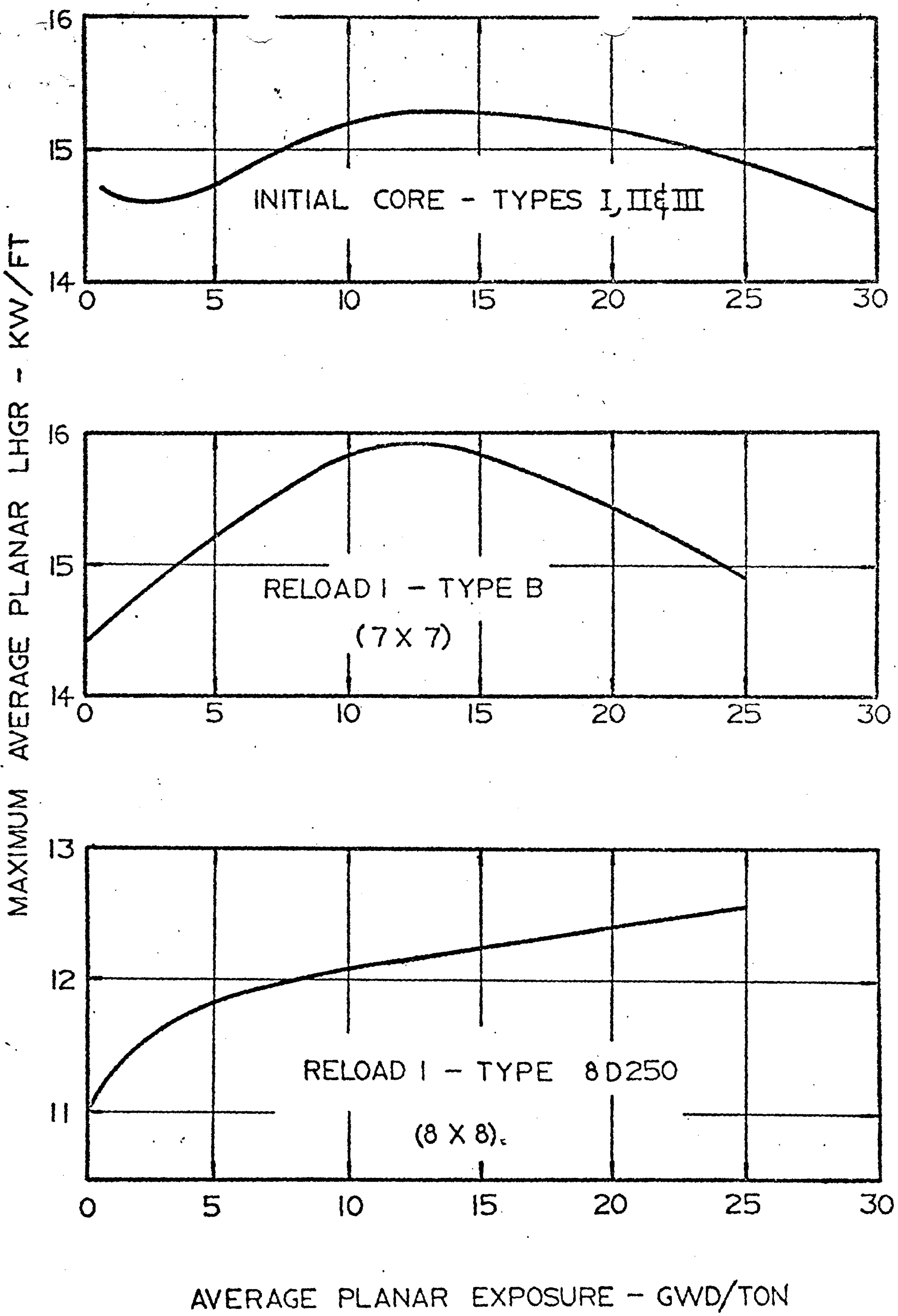
4.5 SURVEILLANCE REQUIREMENTS

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.


MAXIMUM AVERAGE PLANAR LHGR - KW/FT
16
15
14
0
5
10
15
20
25
30
INITIAL CORE - TYPES I, II & III
16
15
14
RELOAD I - TYPE B
(7 X 7)
13
12
11
RELOAD I - TYPE 8D250
(8 X 8)
AVERAGE PLANAR EXPOSURE - GWD/TON

81C

33

3.5 Limiting Condition for Operation Bases (Cont'd)

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on an assumed linearly increasing variation 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. For the initial core fuel, the power spike penalty specified is based on that presented in Reference 9. An irradiation growth factor of .25% was used as the basis for determining the maximum gap size in
3 accordance with References 10 and 11.

For the 7X7 reload fuel, the maximum gap size and evaluation of the power spiking penalty was conducted in accordance with the models specified in References 12-14. For the 8X8 reload fuel, the maximum gap size and evaluation of the power spiking penalty was conducted in accordance with the models specified in References 13-15.

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(9) NEDM-10735, Supplement 6, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Section 3.2.1, Aug. 1973.

(10) J.A. Hinds (GE) Letter to V.A. Moore (USAEC), "Plant Evaluation with GE GEGAP-III," Dec. 12, 1973.

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

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(12) NEDO-20547, "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Dresden Nuclear Power Station Unit 2, 33
Supplement A," Section 3.2.2.1.

(13) J.A. Hinds, Letter to V. Stello, "Power Spiking and Linear Heat Generation Rate Models," 101-344-73, December 10, 1973.

(14) V.A. Moore, Letter to I.S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 22, 1974.

(15) NEDO-20360, "General Electric Boiling Water Reactor Generic Reload Application for 8X8 Fuel, Section 3.3.4.1, April 1974.

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. DPR-19

(CHANGE NO. 33 TO THE TECHNICAL SPECIFICATIONS)

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT 2

DOCKET NO. 50-237

INTRODUCTION

By application dated August 27, 1974, Commonwealth Edison Company (CE) requested authorization to operate Dresden Unit 2 with reload fuel assemblies in the core. According to CE's plan, approximately 156 reload fuel assemblies will replace an equal number of fuel assemblies presently in the core. The reference reload is to consist of forty 7 x 7 fuel assemblies similar to fuel presently in the core and one hundred sixteen 8 x 8 fuel assemblies. The application also includes a request for approval of proposed Technical Specifications related to the reload and to the core thermal safety limit and Limiting Safety System Setting (APRM Flux Trip and Control Rod Block). Supplements to the application were submitted by letters dated October 10, October 22, November 7, November 22 and December 5, 1974.

The acceptability of the neutronic, thermal-hydraulic, and mechanical design of the 8 x 8 fuel assemblies during normal operation, operational transients and postulated accidents was evaluated by the Regulatory staff in a previous report[1]. The use of 8 x 8 fuel assemblies for reloads was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974[2]. The use of 8 x 8 reload fuel assemblies in the Dresden 3 reactor (which is essentially identical to the Dresden 2 reactor) was evaluated and approved by Change No. 16 to the Technical Specifications of Facility Operating License No. DPR-25 dated March 25, 1974.

The 7 x 7 reload fuel assemblies are identical to the 7 x 7 reload fuel assemblies previously approved for operation in the Dresden 3 core by Technical Specification Change No. 16 to Facility Operating License No. DPR-25 dated March 25, 1974.

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With one exception, the evaluations of the acceptability of the reload fuel for the Dresden 3 core by Technical Specification Change No. 16 to Facility Operating License No. DPR-25 are applicable to the Dresden 2 reload fuel. A principle design change for this reload 8 x 8 fuel is the use of leaf springs to minimize the bypass flow area between the fuel assembly shroud and the lower end fitting. The effect of this design change is discussed below.

Our safety evaluation of this reload (Reload No. 1) for the Dresden 2 core is based on the licensee's application as amended, and on information contained in a GE topical report, NEDO-20360[3] referred to in the application. The NEDO-20360 report is still being evaluated by the staff for use as a topical. Our use of that report in this analysis was limited to considerations applicable to Dresden and does not imply acceptability of its use for other facilities.

Authority to load but not operate with Reload 1 fuel was approved by Amendment No. 5, Change No. 31 to Facility Operating License No. DPR-19 dated December 5, 1974.

EVALUATION

The reference core Dresden 2 Reload 1 consists of 508 initial 7 x 7 fuel assemblies and forty 7 x 7 reload fuel assemblies and one hundred sixteen 8 x 8 reload fuel assemblies which are scatter loaded throughout the core. Four fuel assemblies surrounded by control blades will contain only one 8 x 8 reload fuel assembly. This loading scheme assures that, in the core interior, the higher enrichment 8 x 8 reload fuel assembly will be "paired" with three lower powered exposed 7 x 7 fuel assemblies. No significant fuel loading asymmetries will exist.

The Regulatory staff's review[1] of the mechanical design of the 8 x 8 reload fuel assemblies concludes that the background of experience compiled by the General Electric Company is sufficient to enable GE to design fuel rods of new design with confidence in their durability. In addition the 8 x 8 fuel assemblies for Dresden 2 are of similar design to the 8 x 8 reload fuel assemblies approved for use in Dresden 3 by Technical Specification Change 16 to Facility Operating License No. DPR-25. Because Dresden 2 and 3 operate at identical conditions and the fuels used are nearly identical, the evaluation of mechanical design discussed in our safety evaluation for Technical Specification Change No. 16 to Facility Operating License No. DPR-25 is applicable to the 8 x 8 reload fuel assemblies for Dresden 2. The 8 x 8 fuel assemblies for Dresden 2 are of similar design and material to the 7 x 7 fuel assemblies which have successfully been operated at Dresden 2. Both the 8 x 8 and 7 x 7 fuel assemblies will operate at the same pressure and temperature and the fluid velocity and quality will be nearly identical and, therefore, the new 8 x 8 fuel assemblies are expected to exhibit the same operational characteristics as the previously operated 7 x 7 fuel assemblies.

Accident induced loads and stresses have been calculated for both the 7 x 7 and 8 x 8 fuel assemblies using the same methods. The limiting accident loads results from a steam line break. The pressure difference following a steam line break are less than 10% greater than normal operating pressure differences. As in normal operation, the pressure differences in an 8 x 8 assembly following a steam line break are 5 to 10% greater than in a 7 x 7 assembly. The loads following a steam line break are well below the allowable loads.

Based upon the above, the staff concludes that the mechanical design of the 8 x 8 reload fuel assemblies for Dresden 2 is adequate to assure the mechanical integrity of the fuel assemblies. Additional assurance of acceptable fuel performance of the new fuel is provided by the radiological surveillance performed on the reactor primary coolant and off-gas to provide an early indication of incipient fuel failure caused by mechanical deterioration of the fuel assemblies.

We have also reviewed the nuclear design of the reload fuel. The CE submittal indicates that the nuclear characteristics of the Reload 1 fuel assemblies are similar to those previously loaded. Thus the reactivity coefficients and total control system worths of the reconstituted core will not differ significantly from those values which were previously reported for Dresden 2. In addition, the nuclear characteristics of the Dresden 2, Cycle 4 with Reload 1 fuel assemblies are quite similar to the previously approved Reload 2 fuel assemblies for Dresden 3.

The application also indicates that the shutdown margin of the reconstituted core meets the technical specification requirement that the core be at least 0.25% Δk subcritical in the most reactive operating state with the strongest control rod fully withdrawn and with all other control rods fully inserted. The report predicts that, at a core average exposure of 9880 MWD/T at the end of Cycle 3, the shutdown margin is 3.45% Δk with the strongest control rod fully withdrawn and all other rods fully inserted. The analysis applies to control blades with non-inverted boron filled tubes. However, the analysis indicates adequate shutdown margin for Cycle 4 in the event there are a number of blades with inverted control tubes and, therefore, is acceptable.

The application states that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by 3% Δk at 20°C, xenon free. This margin is acceptable.

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The basic criterion for the storage of fuel for Dresden 2 is that k_{eff} of the fuel as stored in the fuel pool is ≤ 0.90. This is achieved if the uncontrolled infinite multiplication factor of a single fuel assembly is limited to 1.26 at 65°C. The reload fuel has an infinite multiplication factor of <1.26, and therefore, meets the fuel storage requirements for Dresden 2.

Based on our evaluation as reported(1), we conclude that a mixed 8 x 8 and 7 x 7 core will be nearly identical, neutronically, to a 7 x 7 core and that the nuclear design is acceptable.

Thermal-hydraulic methods used to analyze assembly flow rates and MCHFR's are discussed in Reference 3. These methods are the same as those used to analyze reactor conditions previously and are acceptable. To provide adequate thermal margin during normal steady-state operation, the reactor is limited to operating with maximum LHGR's of 17.5 Kw/ft for 7 x 7 fuel and 13.4 Kw/ft for 8 x 8 fuel. In addition the MCHFR for both fuel types is 1.9. These operating criteria areaacceptable to the staff to satisfy the criterion of no fuel damage during abnormal operating transients. General Electric has predicted an increase in bypass flow caused by channel wall deflections. The deflection model was developed from mreasurements of creep deformation of the shroud at operating conditions. To nullify this potential increase in flow area, leaf springs have been attached to each of the four sides of the lower end fittings of the reload fuel. The effect of this change on bypass flow and the different hydraulic characteristics of the 8 x 8 fuel assemblies are accounted for in the steady-state and transient analyses that are presented.

Based on a review of the information provided by the licensee, we conclude that:

1. The thermal-hydraulic design criteria and analysis methods, which are the same as those used previously for justifying plant safety, are acceptable.

2. The licensee has accounted for the different hydraulic characteristics of the reload fuel in an acceptable manner.

Abnormal operational transients were discussed in the staff report for 8 x 8 reload fuel(1). As previously discussed, the mechanical, nuclear, and thermal-hydraulic characteristics of the 7 x 7 and 8 x 8 fuel are similar and will respond to the transients similarly.

The application and supplements include analyses of the events which have limiting minimum critical heat flux ratios (MCHFR), including a seizure of one recirculation pump, a continuous withdrawal of a control

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rod, and misorientation of a fuel assembly. The calculated MCHFR's during a pump seizure accident are 1.08 and 1.17 for the 8 x 8 and 7 x 7 fuel assemblies respectively. Rod block monitors are used to maintain MCHFR above 1.05 in the event of a rod withdrawal error. These results are acceptable.

The rod withdrawal error is discussed in the application for Dresden 2 Reload 1 in terms of the worst case condition. The report shows that the local power range monitor subsystem (LPRMs) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will present rod movement at indicated set points and therefore will prevent fuel damage by maintaining MCHFR $\geq$ 1.0.

The application considers loading errors in which an 8 x 8 fuel assembly is placed in a 7 x 7 fuel assembly position and the 7 x 7 fuel assembly is placed in a 8 x 8 fuel assembly position. The report states that no fuel damage would be incurred during subsequent reactor operation with the misplaced fuel bundles at the maximum permitted power. In all cases, the results of these analyses show that the fuel damage limits, i.e., a MCHFR of unity and a cladding strain of one percent, are not reached.

On the basis of the above, we conclude that operation with the reload core will not result in exceeding fuel damage limits during anticipated transients for Dresden 2.

Transient analyses have also been evaluated to determine the effect of Reload 1 on calculated primary system pressure transients. The limiting transient for these analyses is the turbine trip without bypass. The application states that the transient analyses previously performed for the Dresden 3 reload core containing 7 x 7 and 8 x 8 fuel(4) are applicable to the Dresden 2 Reload 1 Cycle 4 core. We reviewed the parameters used for Dresden 3 in our safety evaluation for Technical Specification Change No. 16 to Facility Operating License No. DPR-25 and found no significant differences from the parameters applicable for Dresden 2. We conclude that our previous safety evaluation for Dresden 3 reload is also applicable to Dresden 2 Reload 1, and the Dresden 2 reload core is acceptable.

Accident Analysis

The generic re-evaluation of accidents to account for the effects of 8 x 8 fuel was discussed in the staff evaluation(1) and is applicable for Dresden 2. Plant specific aspects of the accident review were discussed in our evaluation for the Dresden 3 reload for Technical Specification Change No. 16 to Facility Operating License No. DPR-25. Because of the similarity of the reactors and reload fuel to Dresden 2, the Dresden 3 evaluation is applicable to Dresden 2. The ECCS evaluation discussed in our safety evaluation for Technical Specification Change No. 16 to Dresden 3 was based on the Interim Acceptance Criteria and is also applicable to Unit 2. Our evaluation of the ECCS with regard to 10 CFR 50.46 for the Dresden Units will be addressed in a subsequent report.

Proposed Changes to Technical Specifications

Although the performance characteristics of the Reload 1 fuel are similar to previously authorized loadings, certain changes to the technical specifications are necessary to accommodate this fuel. In addition, changes have been made to the limitations related to APRM flux scram and rod block and to the core thermal safety limit.

The changes consist of:

1. Changing LHGR limits related to effects of fuel densification in the 7 x 7 and 8 x 8 Reload 1 fuel assemblies.

2. Adding a maximum average planar LHGR curve related to the IAC for the 7 x 7 and 8 x 8 Reload 1 fuel assemblies.

3. Adding definitions for total peaking factor and limiting total peaking factor.

4. Stating values for limiting total peak factors for 7 x 7 and for 8 x 8 fuel assemblies.

5. Modifying the core thermal safety limit, Figure 1.1.1 to add a correction for high peaking factors for 8 x 8 fuel assemblies and to slightly increase the safety limit for 7 x 7 fuel assemblies with high peaking factors.

6. Modifying the form of the APRM flux scram and rod block for clarity, for conformity to the form presently used in other boiling water reactor technical specifications and to set limitations associated with 8 x 8 fuel assemblies.

The acceptability of these changes is discussed below:

The local LHGR limits have been changed to incorporate the effects of fuel densification on the operation of the reload 8 x 8 and 7 x 7 fuel assemblies. The methods used to calculate appropriate limits to account for fuel densification have been previously approved by the staff for both 8 x 8 and 7 x 7 fuel(1). The proposed specification assures that the maximum linear heat generation rate in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated.

The maximum average planar LHGR curves related to the Interim Acceptance Criteria for the 7 x 7 and 8 x 8 reload fuel assemblies have been added in accordance with the Commission's regulations. The methods used to calculate appropriate MAPLHGR limits related to the Interim Acceptance Criteria are the methods previously approved by the staff(1). As indicated above, we will subsequently address consideration of 10 CFR 50.46.

The proposed change in total peaking factor (TPF) recognizes that different TPF are used for the 7 x 7 and 8 x 8 fuel assembly designs. The limiting total peak factors (LTPF) of 3.05 and 3.01 have been proposed for 7 x 7 and 8 x 8 fuel assemblies respectively. The LTPFs are reference numbers used in calculating the core thermal safety limit, APRM flux scram settings and APRM rod block settings. The values of 3.05 and 3.01 are total peaking factors that would result in peak linear heat generation rates under 17.5 and 13.4 kw/ft in 7 x 7 and 8 x 8 fuel assemblies, respectively. If the peaking factors are above the LTPF, the limits must be reduced, as proposed. to assure that the LHGR's remain acceptable. The values of 3.05 for 7 x 7 fuel is higher than the 3.0 reference peaking factor used previously and results in slightly higher core thermal safety limit for high peaking factors. However, the former reference peaking factor was based on a nominal value of peaking while the proposed limit is based on the design LHGR of 17.5 kw/ft. Therefore, the change from 3.0 to 3.05 is consistent with the analyses previously reviewed and accepted.

The change in format of the APRM rod block and APRM flux trip limits is consistent with the limitations generally used in other boiling water reactors. The format updates the settings and surveillance to reflect use of both 7 x 7 and 8 x 8 fuel assemblies.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Richard D. Silver

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

Fredric D. Anderson

Fredric D. Anderson, Acting Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: DEC 26 1974

REFERENCES

1. Technical Report on the General Electric Company 8 x 8 Fuel Assembly dated February 5, 1974, by the Directorate of Licensing.

2. Report on General Electric 8 x 8 Fuel Design for Reload Use, Advisory Committee on Reactor Safeguards, February 12, 1974.

3. General Electric Boiling Water Reactor General Reload Application for 8 x 8 Fuel, NEDO-20360 (April 1974).

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

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO 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 October 30, 1974 (39 F.R. 38274), the Atomic Energy Commission (the Commission) has issued Amendment No. 7 to Facility Operating License No. DPR-19 to the Commonwealth Edison Company (the licensee) for Unit 2 of the Dresden Nuclear Power Station (the facility), a boiling water reactor located in Grundy County, Illinois, and currently authorized for operation at power levels up to 2527 MWt. The amendment is effective as of its date of issuance.

The license amendment authorizes operation of the facility using a partial reload containing 7 x 7 and 8 x 8 fuel assemblies, deletes the restriction imposed by Amendment 5, Change 31 for operation with 8 x 8 fuel and approves technical specification changes related to (1) the reload, (2) the core thermal safety limit, and (3) limiting safety system settings, limiting conditions of operation, and surveillance requirements related to fuel cladding integrity.

The Commission has found that the application for the amendment dated August 27, 1974, as supplemented, complies with the requirements

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of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations published in 10 CFR Chapter I. 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.

The Commission's Directorate of Licensing has completed its evaluation of the above action and a Safety Evaluation is being issued concurrently with this notice 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 changes to the Technical Specifications as authorized by Amendment No. 7 to License No. DPR-19.

Copies of (1) Amendment No. 7 with Change No. 33 to the Technical Specifications of Facility Operating License No. DPR-19, and (2) the Commission's concurrently issued Safety Evaluation 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, 604 Liberty Street, Morris, Illinois 60670. Single copies of items 1 and 2 may be obtained upon request addressed to the U. S. Atomic Energy Commission, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 26th day of December, 1974.

FOR THE ATOMIC ENERGY COMMISSION

original signed by
Fredric Anderson

~~Fredric D. Anderson, Acting Chief~~
Operating Reactors Branch #2
Directorate of Licensing

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With one exception, the evaluations of the acceptability of the reload fuel for the Dresden 3 core by Technical Specification Change No. 16 to Facility Operating License No. DPR-25 are applicable to the Dresden 2 reload fuel. A principle design change for this reload 8 x 8 fuel is the use of leaf springs to minimize the bypass flow area between the fuel assembly shroud and the lower end fitting. The effect of this design change is discussed below.

Our safety evaluation of this reload (Reload No. 1) for the Dresden 2 core is based on the licensee's application as amended, and on information contained in a GE topical report, NEDO-20360(3) referred to in the application. The NEDO-20360 report is still being evaluated by the staff for use as a topical. Our use of that report in this analysis ~~is as a supplement to the Dresden 2 submittal and our approval of the GE application does not constitute approval of NEDO-20360 for use as a topical~~. was limited to considerations applicable to Dresden and does not imply acceptability of its use for other facilities. Authority to load but not operate with Reload 1 fuel was approved by Amendment No. 5, Change No. 31 to Facility Operating License No. DPR-19 dated December 5, 1974.

was limited to considerations applicable to its use for other facilities.

EVALUATION

The reference core Dresden 2 Reload 1 consists of 508 initial 7 x 7 fuel assemblies and forty 7 x 7 reload fuel assemblies and one hundred sixteen 8 x 8 reload fuel assemblies which are scatter loaded throughout the core. Four fuel assemblies surrounded by control blades will contain only one 8 x 8 reload fuel assembly. This loading scheme assures that, in the core interior, the higher enrichment 8 x 8 reload fuel assembly will be "paired" with three lower powered exposed 7 x 7 fuel assemblies. No significant fuel loading asymmetries will exist.

The Regulatory staff's review(1) of the mechanical design of the 8 x 8 reload fuel assemblies concludes that the background of experience compiled by the General Electric Company is sufficient to enable GE to design fuel rods of new design with confidence in their durability. In addition the 8 x 8 fuel assemblies for Dresden 2 are of similar design to the 8 x 8 reload fuel assemblies approved for use in Dresden 3 by Technical Specification Change 16 to Facility Operating License No. DPR-25. Because Dresden 2 and 3 operate at identical conditions and the fuels used are nearly identical, the evaluation of mechanical design discussed in our safety evaluation for Technical Specification Change No. 16 to Facility Operating License No. DPR-25 is applicable to the 8 x 8 reload fuel assemblies for Dresden 2. The 8 x 8 fuel assemblies for Dresden 2 are of similar design and material to the 7 x 7 fuel assemblies which

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rod, and misorientation of a fuel assembly. The calculated MCHFR's during a pump seizure accident are 1.08 and 1.17 for the 8 x 8 and 7 x 7 fuel assemblies respectively. Rod block monitors are used to maintain MCHFR above 1.05 in the event of a rod withdrawal error. These results are acceptable.

The rod withdrawal error is discussed in the application for Dresden 2 Reload 1 in terms of the worst case condition. The report shows that the local power range monitor subsystem (LPRMs) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will present rod movement at indicated set points and therefore will prevent fuel damage by maintaining MCHFR $\geq$ 1.0.

The application considers loading errors in which an 8 x 8 fuel assembly is placed in a 7 x 7 fuel assembly position and the 7 x 7 fuel assembly is placed in a 8 x 8 fuel assembly position. The report states that no fuel damage would be incurred during subsequent reactor operation with the misplaced fuel bundles at the maximum permitted power. In all cases, the results of these analyses show that the fuel damage limits, i.e., a MCHFR of unity and a cladding strain of one percent, are not reached.

On the basis of the above, we conclude that operation with the reload core will not result in exceeding fuel damage limits during anticipated transients for Dresden 2.

Transient analyses have also been evaluated to determine the effect of Reload 1 on calculated primary system pressure transients. The limiting transient for these analyses is the turbine trip without bypass. The application states that the transient analyses previously performed for the Dresden 3 reload core containing 7 x 7 and 8 x 8 fuel(4) are applicable to the Dresden 2 Reload 1 Cycle 4 core. We reviewed the parameters used for Dresden 3 in our safety evaluation for Technical Specification Change No. 16 to Facility Operating License No. DPR-25 and found no significant differences from the parameters applicable for Dresden 2. We conclude also that our previous safety evaluation for Dresden 3 reload is applicable to Dresden 2 Reload 1 ~~and, therefore,~~ , and the Dresden 2 reload core is acceptable.

Accident Analysis

The generic re-evaluation of accidents to account for the effects of 8 x 8 fuel was discussed in the staff evaluation(1) and is applicable for Dresden 2. Plant specific aspects of the accident review were discussed in our evaluation for the Dresden 3 reload for Technical Specification Change No. 16 to Facility Operating License No. DPR-25. Because of the similarity of the reactors and reload fuel to Dresden 2, the Dresden 3 evaluation is applicable to Dresden 2. The ECCS

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evaluation discussed in our safety evaluation for Technical Specification Change No. 16 to Dresden 3 was based on the Interim Acceptance Criteria and is also applicable to Unit 2. Our evaluation of the ECCS with regard to 10 CFR 50.46 ~~will be issued prior to the startup of Dresden 2 operation for Cycle 4.~~ for the Dresden units will be addressed in a subsequent report.

Proposed Changes to Technical Specifications

Although the performance characteristics of the Reload 1 fuel are similar to previously authorized loadings, certain changes to the technical specifications are necessary to accommodate this fuel. In addition, changes have been made to the limitations related to APRM flux scram and rod block and to the core thermal safety limit.

The changes consist of:

1. Changing LHGR limits related to effects of fuel densification in the 7 x 7 and 8 x 8 Reload 1 fuel assemblies.

2. Adding a maximum average planar LHGR curve related to the IAC for the 7 x 7 and 8 x 8 Reload 1 fuel assemblies.

3. Adding definitions for total peaking factor and limiting total peaking factor.

4. Stating values for limiting total peak factors for 7 x 7 and for 8 x 8 fuel assemblies.

5. Modifying the core thermal safety limit, Figure 1.1.1 to add a correction for high peaking factors for 8 x 8 fuel assemblies and to slightly increase the safety limit for 7 x 7 fuel assemblies with high peaking factors.

6. Modifying the form of the APRM flux scram and rod block for clarity, for conformity to the form presently used in other boiling water reactor technical specifications and to set limitations associated with 8 x 8 fuel assemblies.

The acceptability of these changes is discussed below:

The local LHGR limits have been changed to incorporate the effects of fuel densification on the operation of the reload 8 x 8 and 7 x 7 fuel assemblies. The methods used to calculate appropriate limits to account for fuel densification have been previously approved by the staff for both 8 x 8 and 7 x 7 fuel(1). The proposed specification assures that the maximum linear heat generation rate in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated.

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in accordance with the Commission's regulations,

The maximum average planar LHGR curves related to the Interim Acceptance Criteria for the 7 x 7 and 8 x 8 reload fuel assemblies have been added in accordance with the Commission's regulations, ~~in accordance with Paragraph 50.46 of 10 CFR Part 50 as an interim measure until we have reviewed and approved the method of analysis in accordance with the criteria of 10 CFR Part 50 Appendix K.~~ The methods used to calculate appropriate MAPLHGR limits related to the Interim Acceptance Criteria are the methods previously approved by the staff(1). As indicated above we will subsequently address consideration of 10 CFR 50.46.

The proposed change in total peaking factor (TPF) recognizes that different TPF are used for the 7 x 7 and 8 x 8 fuel assembly designs. The limiting total peak factors (LTPF) of 3.05 and 3.01 have been proposed for 7 x 7 and 8 x 8 fuel assemblies respectively. The LTPFs are reference numbers used in calculating the core thermal safety limit, APRM flux scram settings and APRM rod block settings. The values of 3.05 and 3.01 are total peaking factors that would result in peak linear heat generation rates under 17.5 and 13.4 kw/ft in 7 x 7 and 8 x 8 fuel assemblies, respectively. If the peaking factors are above the LTPF, the limits must be reduced, as proposed. to assure that the LHGR's remain acceptable. The values of 3.05 for 7 x 7 fuel is higher than the 3.0 reference peaking factor used previously and results in slightly higher core thermal safety limit for high peaking factors. However, the former reference peaking factor was based on a nominal value of peaking while the proposed limit is based on the design LHGR of 17.5 kw/ft. Therefore, the change from 3.0 to 3.05 is consistent with the analyses previously reviewed and accepted.

The change in format of the APRM rod block and APRM flux trip limits is consistent with the limitations generally used in other boiling water reactors. The format updates the settings and surveillance to reflect use of both 7 x 7 and 8 x 8 fuel assemblies.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

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

Fredric D. Anderson, Acting Chief
Operating Reactors Branch #2
Directorate of Licensing

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