JUN 14 1971

Docket No. 50-237

DISTRIBUTION:
See attached sheet.

Commonwealth Edison Company
ATTN: Byron Lee, Jr.
Assistant to the President
P.O. Box 767
Chicago, Illinois 60670

Change No. 11
License No. DPR-19

Gentlemen:

We have reviewed your Proposed Change No. 11, dated May 11, 1971, and supplemented by letter, dated May 21, 1971, requesting changes to Section 2.1 and Tables 3.1.1 and 4.1.1 of the Technical Specifications of Provisional Operating License No. DPR-19 for Dresden Nuclear Power Station Unit 2. These proposed changes would update the specification requirements as a result of plant modifications and operating experience as described in your letter of April 6, 1971. These changes would specifically add a scram function during the start-up/hot standby mode of plant operation and add a scram function in the event of a failure of the control oil piping of the electro-hydraulic control system for the turbine-generator.

We have concluded that the proposed changes do not present significant hazards considerations not described or implicit in the safety analysis report and that there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, the enclosed pages 5, 6, 6A, 16, 16A, 17, 18, 18A, 23, 24, 25, and 27 are replaced or added to the Technical Specifications.

Sincerely,

ORIGINAL SIGNED BY
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Chg.11 to the Tech. Spec.

cc: Arthur C. Gehr, Esquire
Isham, Lincoln & Beale
Counselors at Law

OFFICE ▶	DRL:BWR-2	DRL:BWR-2	DRL:BWR	DRL:A/Dir	DRL:DIR	
SURNAME ▶	G. Lainas	RLTedesco	RSBoyd	FSchroeder	PAMorris	
DATE ▶	6/2/71	6/1/71	6/2/71	6/2/71	6/12/71	

Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

Commonwealth Edison Company
(Change No. 11 to License No. DPR-19)

JUN 14 1971

DISTRIBUTION:
AEC PDR (50-237)
Docket File (50-237)
DR Files
DRL Reading
BWR-2 Files
G. C. Lainas
E. G. Case
OGC
R. S. Boyd
CO (2)
R. DeYoung
S. H. Hanauer, DR
D. Skovholt
P. Collins
BWR Branch Chiefs
B. Grimes
S. Kari (2)
M. Wetterhahn
N. Dube (5)

bcc: H. J. McAlduff, ORO
Harvey Mueller, GMR/H
J. A. Harris, PI
R. Leith, OC
J. Buchanan, ORNL
A. A. Wells, ASLBP
T. W. Laughlin, DTIE
S. Robinson, SECY

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

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 peak heat flux is above the curve in Figure 2.1.2. When the combination of power and peak heat flux is above the curve in Figure 2.1.2 a scram setting(s) as given by:

$$S \leq \frac{502,000\ P}{X}$$

where:

P = percent of rated power
X = peak heat flux – ($Btu/hr/ft^2$)

* 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 heat flux is above the curve in Figure 2.1.2. When the combination of power and peak heat flux is above the curve in Figure 2.1.2 a rod block trip setting (S_{RB}) as given by:

$$S_{RB} = \frac{451,440\ P}{X}$$

where:

P = percent of rated power
X = peak heat flux ($Btu/hr/ft^2$)

shall be used.

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" ($^{+4''}_{-0''}$) 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.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING
	H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be $\leq$850 psig. * I. Turbine Control Valve Fast Closure Scram on loss of control oil pressure shall be set at greater than or equal to 1100 psig.

the equation will prevent MCHFR from becoming less than 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.

* 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.

B. 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 greater than 3.0 exist. This assures a 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 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 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.

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.

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.

* I. The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number of Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action*
			Refuel (7)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	≤120/125 of Full Scale	X	X	X(5)	A
3	Inoperative		X	X	X(5)	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(9)	X	A or B
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥5/125 of Full Scale	X(12)	X(12)	X(13)	A or B
* 2	High Flux (15% scram)	Specification 2.1.A.2	X	X	X(14)	A
2	High Reactor Pressure	≤1060 psig	X(11)	X	X	A
2	High Drywell Pressure	≤2 psig	X(8), (10)	X(8), (10)	X(10)	A
2	Reactor Low Water Level	≥1 inch***	X	X	X	A
2	High Water Level in Scram Discharge Tank	≤50 gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥23 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Streamline High Radiation	≤7 X Normal Full Power Background	X(3)	X(3)	X	A or C
4 (6)	Main Streamline Isolation Valve Closure	≤10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection	****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	≤10% Valve Closure	X(4)	X(4)	X(4)	A or C
* 2	Turbine Control- Loss of control oil pressure	Greater than or equal to 1100 psig	X	X	X	A or C

TABLE 3.1.1 (cont)

Notes:

1. There shall be two operable or tripped trip systems for each function.
2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.
3. Permissible to bypass when reactor pressure is < 600 psig.
4. Permissible to bypass when first stage turbine pressure is less than that which corresponds to 45% rated steam flow.
5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.
6. The design permits closure of any one valve without a scram being initiated.
7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:
   a. Mode Switch in Shutdown
   b. Manual Scram
   c. High Flux IRM
   d. Scram Discharge Volume High Level
8. Not required to be operable when primary containment integrity is not required.
9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).
10. May be bypassed when necessary during purging for containment inerting or deinerting.
11. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
12. The APRM downscale trip function is automatically bypassed when the reactor mode switch is in the refuel and startup/hot standby positions.
13. The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

* 14. The APRM 15% scram is bypassed in the run mode.

* If the first column cannot be met for one of the trip systems, that trip system shall be tripped.
If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:
A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.
C. Reduce turbine load and close main steamline isolation valves within 8 hours.

** An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

*** 1 inch on the water level instrumentation is 143" above the top of the active fuel.

**** Trips upon actuation of the fast closure solenoid which trips the turbine control valves.

TABLE 4.1.1

SCRAM INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Instrument Channel	Group (3)	Functional Test	Minimum Frequency (4)
	Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
	Manual Scram	A	Trip Channel and Alarm	Every 3 Months
	IRM			
	High Flux	C	Trip Channel and Alarm (5)	Before Each Startup
	Inoperative	C	Trip Channel and Alarm	Before Each Startup
	APRM			
	High Flux	B	Trip Output Relays (5)	Once Each Week
	Inoperative	B	Trip Output Relays	Once Each Week
	Downscale	B	Trip Output Relays (5)	Once Each Week
*	High Flux (15% scram)	B	Trip output relays	Before each start-up
	High Reactor Pressure	A	Trip Channel and Alarm	(1)
	High Drywell Pressure	A	Trip Channel and Alarm	(1)
	Reactor Low Water Level (2)	A	Trip Channel and Alarm	(1)
	High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 3 Months
	Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	(1)
	Main Steamline High Radiation (2)	B	Trip Channel and Alarm (5)	Once Each Week
	Main Steamline Isolation Valve Closure	A	Trip Channel and Alarm	(1)
	Generator Load Rejection	A	Trip Channel and Alarm	(1)
	Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
*	Turbine Control-Loss of Control Oil Pressure	A	Trip channel and alarm	(1)

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test	Minimum Frequency (2)
High Flux IRM	C	Comparison to APRM after Heat Balance	Every Shutdown
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Water Level	Every 3 Months
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steamline High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine Control-Loss of Control Oil Pressure	A	Pressure Source	Every 3 months

Notes:

1. A description of the three groups is included in the bases of this Specification.
2. Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.
3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made during each refueling outage.