UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 43
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 15, 1979, as supplemented March 2, 1979, April 6, 1979, April 12, 1979 and April 20, 1979, complies 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. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.B and 3.F of Provisional Operating License No. DPR-19 are hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 43, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   F. Restrictions

      Reactor power level shall be limited to maintain pressure margin to the safety valve setpoints during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 4 licensing submittal for Dresden Unit 2 (NEDO-24160). Subsequent operation in the coastdown mode to 70% rated power is permitted based on the Generic Reload Fuel Application (NEDE-24011). Plant operation in the coastdown mode from 70% to 40% rated power shall be limited to the operating plan described in NEDO-24034 (Dresden 2 Reload No. 3) using full recirculation flow.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Albert Schwencer /for

Richard H. Vollmer, Assistant Director
for Systems & Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 24, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 43

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise Appendix A Technical Specifications by removing the following pages and by inserting the enclosed pages. The revised pages contain the captioned amendment number and marginal lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
5	5
6	6
7	7
8	8
9	9
10	10
11	11
12	12
16	16
18	18
19	19
24	24
38	38
40	40
42	42
46	46
48	48
49	49
57	57
57A	57A
62	62
62A	62A
62B	62B
63	63
81C	81C
81C-1	81C-1
81C-2	81C-2
81D	81D
82	82
85A	85A
90	90
125	125

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

A. Reactor Pressure >800 psig and Core Flow >10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

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

with a maxium set point of 120% for core flow equal to 98 x 10^6 lb/hr and greater.

where:

S = setting in percent of rated power

W - percent of drive flow required to produce a rated core flow of 98 Mlb/hr.

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.

LTPF = 3.05 (7x7 fuel assemblies)
3.01 (8x8 fuel assemblies)
2.98 (8x8 R fuel assemblies)

2. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

When the reactor mode switch is in the refuel or startup/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

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psig)

When the reactor pressure is ≤ 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

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

D. Reactor Water Level (Shutdown Condition)

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.*

*Top of active fuel is defined to be 360" above vessel zero. (see Bases 3.2)

3. IRM Flux Scram Trip Setting

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

B. APRM Rod Block Setting

The APRM rod block setting shall be:

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

The definitions used above for the APRM scram trip apply.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING
	C. Reactor low water level scram setting shall be $\geq$144" above the top of the active fuel at normal operating conditions.*
	D. Reactor low water level ECCS initiation shall be 84" ($^{+4"}_{-0"}$) above the top of the active fuel at normal operating conditions.*
	E. Turbine stop valve scram shall be $\leq$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 $\leq$10% valve closure from full open.
	H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be $\geq$ 850 psig.
	I. Turbine Control Valve Closure Scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.
	*Top of active fuel is defined to be 360" above vessel zero. (See Bases 3.2)

Amendment No. ~~21~~, 43

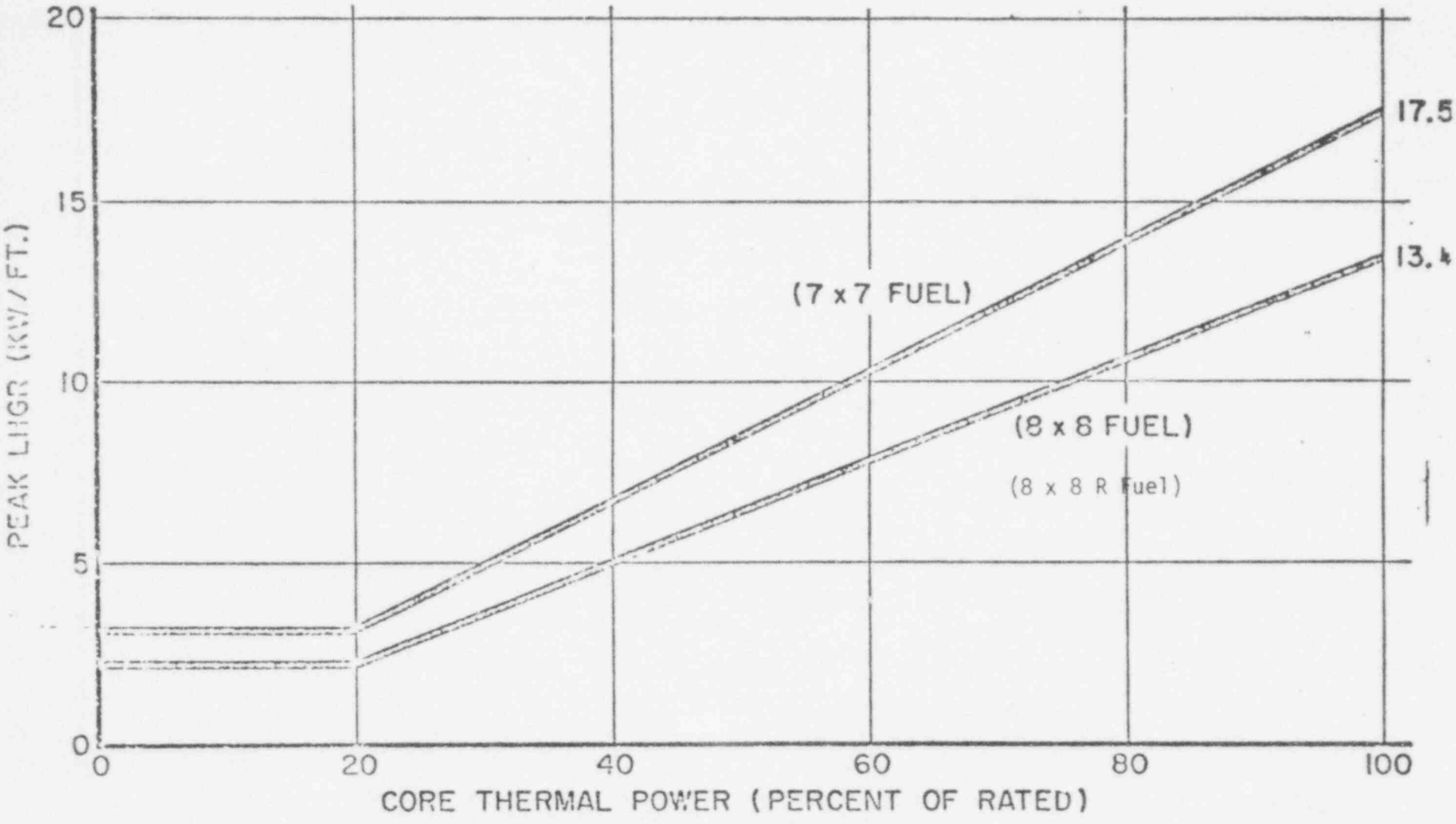

Figure 2.1-2
PEAK LHGR VERSUS CORE THERMAL POWER
FOR A LIMITING TOTAL PEAKING FACTOR

FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such
that no calculated fuel damage would occur as
a result of an abnormal operational transient.
Because fuel damage is not directly observable,
a step-back approach is used to establish a
Safety Limit such that the minimum critical power
ratio (MCPR) is no less than 1.07. MCPR > 1.07
represents a conservative margin relative to the
conditions required to maintain fuel cladding
integrity.

The fuel cladding is one of the physical barriers
which separate radioactive materials from the
environs. The integrity of this cladding barrier
is related to its relative freedom from perfora-
tions or cracking. Although some corrosion or use
related cracking may occur during the life of the
cladding, fission product migration from this
source is incrementally cumulative and continuously
measurable. Fuel cladding perforations, however,
can result from thermal stresses which occur from
reactor operation significantly above design
conditions and the protection system safety
settings. While fission product migration from
cladding perforation is just as measurable as
that from use related cracking, the thermally
caused cladding perforations signal a threshold,
beyond which still greater thermal stresses may
cause gross rather than incremental cladding
deterioration. Therefore, the fuel cladding
Safety Limit is defined with margin to the
conditions which would produce onset of transition
boiling, (MCPR of 1.0). These conditions
represent a significant departure from the
condition intended by design for planned
operation.

A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

Onset of transition boiling results in
a decrease in heat transfer from the
clad and, therefore, elevated clad
temperature and the possibility of clad
failure. However, the existence of critical
power, or boiling transition, is not a
directly observable parameter in an
operating reactor. Therefore, the margin
to boiling transition is calculated from
plant operating parameters such as core
power, core flow, feedwater temperature,
and core power distribution. The margin
for each fuel assembly is characterized by
the critical power ration (CPR) which is
the ratio of the bundle power which would
produce onset of transition boiling divided
by the actual bundle power. The minimum
value of this ratio for any bundle in the
core is the minimum critical power ratio
(MCPR). It is assumed that the plant
operation is controlled to the nominal
protective setpoints via the instrumented
variables. (Figure 2.1-3).

1.1.A Reactor Pressure >800 psig and Core Flow >10% of Rated (Cont'd)

The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, 1.07, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation. See e.g. Reference (1)

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.07 would not produce boiling transition.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psig during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR), operation is constrained to a maximum LHGR - 17.5 kw/ft for 7 x 7 fuel and 13.4 kw/ft for 8 x 8 and 8 x 8R fuel. This constraint is established by specifications 2.1.A.1 and 3.5.J. Specification 2.1.A.1 established limiting total peaking factors (LTPF) which constrain LHGR's to the maximum values at 100% power and established procedures for adjusting APRM scram settings which maintain equivalent safety margins when the total peak factor (TPF) exceeds the LTPF. Specification 3.5.J established the LHGR max. which cannot be exceeded under steady power operation.

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(1) NEDO-20694, "General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Dresden Nuclear Power Station Unit 3."

Safety Limit Bases (cont'd)

B. Core Thermal Power Limit (Reactor Pressure < 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.84 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

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 in detail. 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 and at least every 32 weeks 50% are checked to assume adequate insertion times. 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 MCPR = 1.07 is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

2.1.A. Neutron Flux Trip Setting

3. IRM Flux Scram Trip Setting (cont'd)

The IRM scram trip 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 a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. 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.

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 one percent of rated power, thus maintaining MCPR above **1.07.** 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 backup protection for the APRM.

2.1.B APRM Rod Block Trip Setting

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 to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. 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 trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the limiting total peaking factor, thus preserving the APRM rod block safety margin.

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E. Turbine Stop Valve Scram - The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 even during the worst case transient that assumes the turbine bypass is closed.

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 MCPR from becoming less than 1.07 for this transient. For the load rejection from 100% power, the LHGR increases to only 106.5% of its rated value which results in only a small decrease in MCPR.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure, there is no increase in neutron flux.

1.2 SAFETY LIMIT	2.2 LIMITING SAFETY SYSTEM SETTING

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be ≤ 1060 psig.

B. Primary System Safety Valve Nominal Settings shall be as follows:

1 valve at 1115 psig*
2 valves at 1240 psig
2 valves at 1250 psig
2 valves at 1260 psig
2 valves at 1260 psig

The allowable setpoint error for each valve shall be ±1%.

*Target Rock combination safety/relief valve

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 ≥ 504" above vessel 0 (See Bases 3.2).

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

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Change No. ~~11~~
Amendment No. 43

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum No. of Operable Inst. Channels per Trip System (1)	Instruments	Trip Level Setting	Action (3)
2	Reactor Low Water	>144" above top of active fuel *	A
2	Reactor Low Low Water	≥ 84" above top of active fuel *	A
2	High drywell pressure	≤2 psig rated (4), (5)	A
2 (2)	High Flow Main Steam line	≤120% of rated steam flow	B
2 of 4 in each of 4 sets	High Temperature Main Steam Line Tunnel	≤200°F	B
2	High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated power background	B
2	Low Pressure Main Steamline	≥850 psig	B
	High Flow Isolation Condenser Line		
1	Steamline Side	≤20 psi diff. on steamline side	C
1	Condensate Return Side	≤ 32" water diff. on condensate return side	C
2	High Flow HPCI Steam Line	≤150" water	D
4	High Temperature HPCI Steam Line Area	≤200°F	D

Notes:

1. Whenever primary containment integrity is required, there shall be two operable or tripped trip systems for each function, except for low pressure main steamline which only need be available in the RUN position.
2. Per each steamline.
3. Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA Analysis (See Bases 3.2).

Change No. ~~28~~
Amendment No. 43

TABLE 3.2.2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Min. No. of Operable Inst. Channels per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low Low Water Level	84" ($^{+4"}_{-0"}$) above top of active fuel *	1. In conjunction with low reactor pressure initiates core spray and LPCI. 2. In conjunction with high dry-well pressure 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and SBGTS. 4. Initiates starting of diesel generators.
2	High Drywell Pressure (2), (3)	$\leq$ 2 psig	1. Initiates core spray, LPCI, HPCI, and SBGTS. 2. In conjunction with low low water level, 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators.
1	Reactor Low Pressure	300 psig$\leq$p$\leq$350 psig	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
	Containment Spray Interlock		Prevents inadvertent operation of containment spray during accident conditions.
1(4)	2/3 Core Height	$\geq$2/3 core height	
2(4)	Containment High Pressure	0.5 psig$\leq$p$\leq$1.5 psig	
1	Timer Auto Blowdown	$\leq$120 seconds	In conjunction with low-low reactor water level, high dry-well pressure, and low pressure core cooling interlock initiates auto blowdown.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analysis (See Bases 3.2).

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias)(7)	$\leq$ [.65W + 43] [illegible] (2)
* 1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$12/125 full scale
2	APRM downscale (7)	$\geq$3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	$\leq$ [.65W + 42] (2)
1	Rod block monitor downscale (7)	$\geq$5/125 full scale
3	IRM downscale (3)	$\geq$5/125 full scale
3	IRM upscale	$\leq$108/125 full scale
* 3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec

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Bases:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 504" (1" on the instrument) above vessel zero. 504" above vessel zero corresponds to 144" above the top of active fuel for types 7 x7, 7 x 7R and 8 x8D fuel. However, since fuel type 8 x 8 DRB (8 x 8 retrofit) has an active fuel length which is 1.24" longer than 7 x 7, 7 x 7R and 8 x 8D, and the actual water levels for the scram and ECCS initiation in the LOCA analysis remained unchanged, the top of active fuel for all technical Specification and licensing analysis purposes is defined as 360" above vessel zero. (It should be noted that the level referenced to the top of active fuel is not meaningful in the analysis, since the level referenced to vessel zero, which is used in the (NEDO-24146) LOCA analyis, has remained unchanged). This trip initiates closure of Group 2, and 3 primary containment isolation valves but does not trip the recirculation pumps. Ref. Section 7.7.2.2 SAR. For a trip setting of 504" above vessel zero, and a 60 second valve closure time the valves will be closed before perforation of the clad occurs even for the maximum break and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 444" above vessel zero (-59" on the instrument). This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section 7.7.2.2 SAR, and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Ref.

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 insturmentation 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 MCPR does not approach 1.07. 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 are 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 MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The APRM rod block function 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 MCPR from decreasing below 1.07 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reaches 1.07 thus allowing adequate margin.

Below 30 percent power, the worst case withdrawal of a single control rod results in a MCPR greater than 1.07 without rod block action. Thus, below this power level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches 1.07.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 5/125 of full scale.

* The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a back-up to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. Ref. Section 6.2.6.3 SAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. The trip logic is a 1 out of 2 for each set and each set can initiate a trip independent of the other set. Any upscale trip will cause the desired action. Trip settings of 11 mr/hr for the monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation to limit the dose

POOR ORIGINAL

3.3 LIMITING CONDITION FOR OPERATION

3. (a) Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not result in a peak fuel enthalpy in excess of 280 cal/gm.

   (b) Whenever the reactor is in the startup or run mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable Rod Worth Minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The Rod Worth Minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of specifications 3.3.A.1 if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4.3 SURVEILLANCE REQUIREMENTS

3. (a) To consider the rod worth minimizer operable, the following steps must be performed:

   (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

   (ii) The rod worth minimizer computer on-line diagnositc test shall be successfully completed.

   (iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

   (iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

   (b) If the rod worth minizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

3.3 LIMITING CONDITIONS FOR OPERATION

4. Control rod shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operating with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so tho the MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 SURVEILLANCE REQUIREMENTS

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have been observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

E. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the SAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident more severe than analyzed.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to result in a peak fuel enthalpy of 280 cal/gm if they were to drop out of the core in the manner defined for the Rod Drop Accident.(5) These sequences are developed prior to operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. These sequences are developed to limit reactivity worths of control rods and together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

These techniques are described in a topical report [1] and two supplements. [2] [3] In addition, a banked position withdrawal sequence described in Reference (4) has been developed to further reduce incremental rod worths.

By using the analytical models described in those reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 20% of rated power, the specified limit (typically 1.3% ΔK) on insequence control rod or control rod segment worths will limit the peak fuel enthalpy to less than 280 cal/gm. Above 20% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy of 280 cal/gm should a postulated control rod drop accident occur.

The following parameters and worst-case bounding assumptions have been utilized in the reload analysis to determine compliance with the 280 cal/gm peak fuel enthalpy. Details of this analysis are contained in Reference 6. Each core reload will be analyzed to show conformance to the limiting parameters.

a. An inter-assembly local power peaking factor [5]

b. The delayed neutron fraction chosen for the bounding reactivity curve.

c. A beginning-of-life Doppler reactivity feedback.

d. Scram times slower than the technical Specification rod scram insertion rate (Section 3.3.2.1)

e. The maximum possible rod drop velocity (3.11 ft./sec.)

f. The design accident and scram reactivity shape function.

g. The minimum moderator temperature to reach criticality.

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(1) Paone, C.J., Stirn, R.C. and Wooley, J.A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.

(2) Stirn, R.C., Paone, C.J., and Young, R.M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, July 1972

(3) Stirn, R.C., Paone, C.J., and Haun, J.M., "Rod Drop Accident Analysis for Large BWR's Addendum No. 2, Exposed Cores", Supplement 2-NEDO 10527, January 1973.

(4) C.J. Paone, "Banked Position Withdrawal Sequence" Licensing Topical Report NEDO-2123, January 1977.

(5) To include the power spike effect caused by gaps between fuel pellets.

(6) Generic Reload Fuel Application NEDE-24011-P-A, August 1978

Bases (con'd)

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of insequence rods or rod segments

in conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit.

Should a control drop accident result in a peak fuel energy content of 280 cal/gm less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The Rod Worth Minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.9 SAR. It serves as a backup to procedural control of control rod worth. In the event that the Rod Worth Minimizer is out of service, when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance functions of the Rod Worth Minimizer. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is requried to be operating during a startup for the withdrawal of a significant number control rods for any startup after June 1, 1974.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the

operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPRs less than 1.07. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than 1.07. Reference (1) shows the control rod scram reactivity used in analyzing the transients. Reference (1) should not be confused with the total control rod worth, 18%Δk, as listed in some amendments to the SAR. The 18%Δk value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Reference (1) represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the

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(1) "Dresden Station Special Report No. 29, Supplement B", Figure 1.

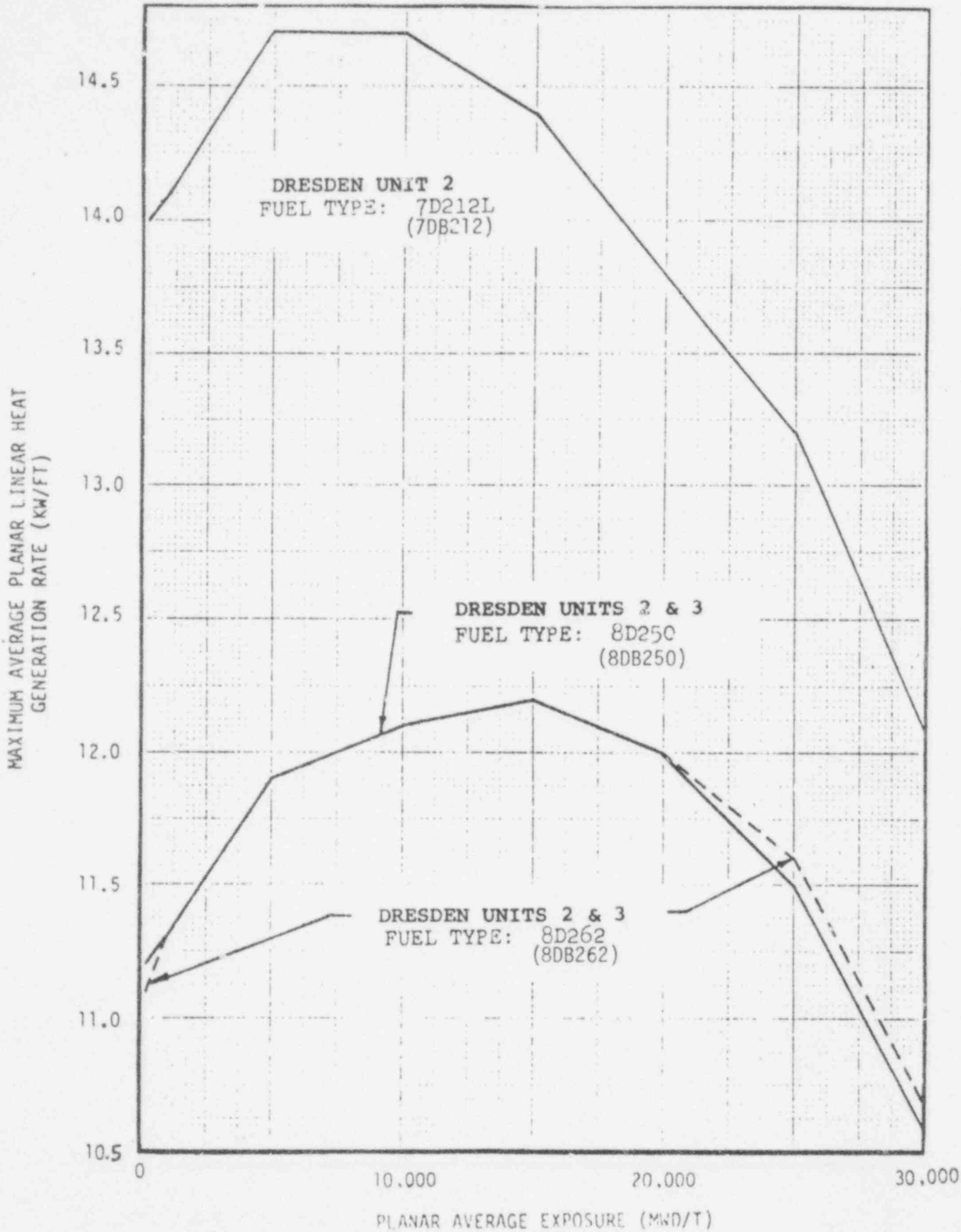

FIGURE 3.5.1-A MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

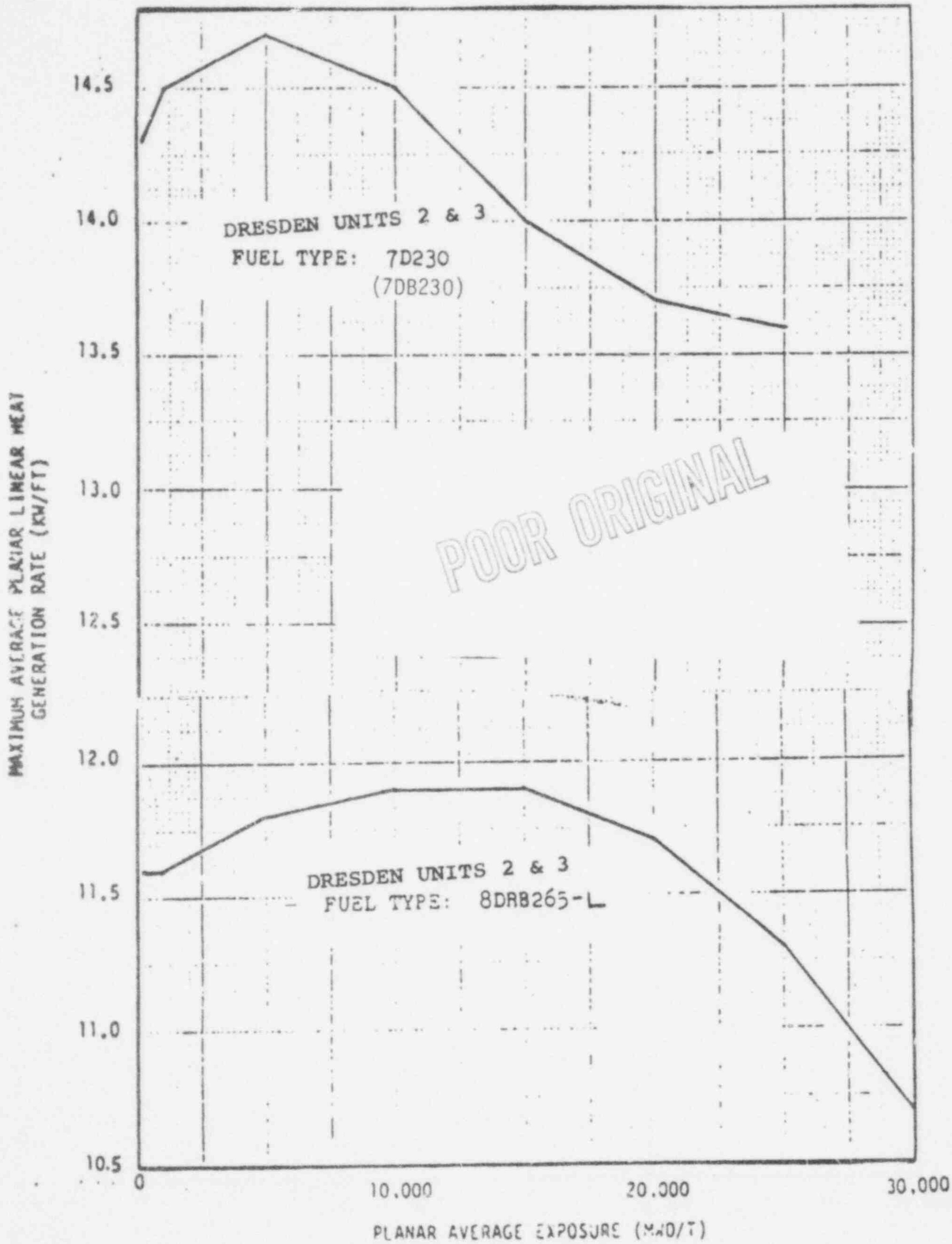

FIGURE 3.5.1-B MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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. $LHGR_{max} < LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max}\left(\frac{L}{L_T}\right)\right]$ $LHGR_d$ - Design LHGR = 17.5 kw/ft, 7x7 fuel assemblies - 13.4 kw/ft, 8x8 fuel assemblies 8x8 R fuel assemblies $\left(\frac{\Delta P}{P}\right)_{max}$ - Maximum power spiking penalty - .037 initial core fuel - .026 reload 1, 7x7 fuel - .022 8x8 fuel .000 8x8 R fuel L_T - Total Core Length - 12 ft. L - Axial distance from bottom of core If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the	J. Local LHGR The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

POOR ORIGINAL

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENTS
K. Minimum Critical Power Ratio (MCPR) During steady state operation, MCPR shall be greater than or equal to - Unit 2 1.24 (7 x 7 fuel) 1.31 (8 x 8 fuel/8x8R) at rated power and flow. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of K_f, where K_f is as shown in Figure 3.5-2. If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	K. Minimum Critical Power Ratio (MCPR) MCPR shall be determined daily during a reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

3.5 Limiting Conditions for Operation Bases

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the

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(1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 NEDO-24146, September 1978"

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" April 1969, I.M. Jacobs and P.W. Marriott.

3.5 Limiting Condition for Operation Bases (Cont'd)

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1). Power operation with LHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit. These values represent limits for operation to ensure conformance with 10CFR50 and Appendix K only if they are more limiting than other design parameters.

(1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations, NEDO-24146, September 1978."

The maximum average planar LHGRs plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any rod is less than the design linear

POOR ORIGINAL

3.6 LIMITING CONDITION FOR OPERATION

an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all eight of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be $\geq$90 psig and $\geq$320°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (Psig)
1	1115*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is ±1%

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (Psig)
1	1115*
2	$\leq$ 1130
2	$\leq$ 1135

*Target Rock combination safety/relief valve

Bases:

3.7

A. Primary Containment — The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Ref. Section 5.2.3 SAR.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet and the minimum volume of 112,000 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests (9) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifica-

(9) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205 December 28, 1962.

36