ATTACHMENT NO. 2

PROPOSED
TECHNICAL SPECIFICATIONS
CHANGES

1.26 Fraction of Limiting Power Density (FLPD) - The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

1.27 Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

1.28 Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power.

1.29 Top of Active Fuel (TAF) - 353.3 inches above vessel zero.

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications:

A. When the reactor pressure is greater than or equal to 800 psia and the core flow is greater than or equal to 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.

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

C. In the event that reactor parameters exceed the limiting safety system settings in specification 2.3 and a reactor scram is not initiated by the associated protective instrumentation, the reactor shall be brought to, and remain in, the cold shutdown condition until an analysis is performed to determine whether the safety limit established in specification 2.1.A and 2.1.B was exceeded.

D. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 10 inches above the top of active fuel.

E. During all modes of operation except when the reactor head is off and the reactor is flooded to a level above the main steam nozzles, at least two [2] recirculation loop suction valves and their associated discharge valves will be in the full open position.

Bases:

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core thermal power.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all 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 power and all 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 irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. 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. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psi or core flow less than 10% is conservative.

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 2.1.A or 2.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. Specification 2.1.C requires that appropriate analysis be performed to verify that backup protective instrumentation has prevented exceeding the fuel cladding integrity safety limit prior to resumption of power operation. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

If reactor water level should drop below the top of the active fuel, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. With a water level above the top of the active fuel, adequate cooling is maintained and the decay heat can easily be accommodated. It should be noted that during power generation there is no clearly defined water level inside the shroud and what actually exists is a mixture level. This mixture begins within the active fuel region and extends up through the moisture separators. For the purpose of this specification water level is defined to include mixture level during power operations.

The lowest point at which the water level can presently be monitored is 20 inches below the top of active fuel. Although the lowest reactor water level limit which ensures adequate core cooling is the top of the active fuel, the safety limit has been conservatively established at 10 inches above the top of active fuel.

Specification 2.1.E assures that an adequate flow path exists from the annular space, between the pressure vessel wall and the core shroud, to the core region. This provides for good communication between these areas, thus assuring that reactor water level instrument readings are indicative of the water level in the core region.

REFERENCES

(1) NEDO-24195, General Electric Reload Fuel Application for Oyster Creek.

2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

Applicability: Applies to the limit on reactor coolant system pressure.

Objective: Preserve the integrity of the reactor coolant system.

Specification: The reactor coolant system pressure shall not exceed 1375 psig whenever irradiated fuel is in the reactor vessel.

Bases:

The reactor coolant system[1] represents an important barrier in the prevention of the uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1375 psig was derived from the design pressures of the reactor pressure vessel, coolant piping, and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1200 psig at 570°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 I for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III for the isolation condenser and the ASA Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over the design pressure (110% x 1250 = 1375 psig) and the ASA Code permits pressure transients up to 15% over the design pressure (115% x 1200 = 1380 psig).

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 20,000 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 2 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 22,000 psi, still almost a factor of 2 below the yield strength. The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

The normal operating pressure of the reactor coolant system is 1020 psig. An overpressurization analysis (2) is performed each cycle to assure the pressure safety limit is not exceeded. The reactor fuel cladding can withstand pressures up to the safety limit, 1375 psig, without collapsing.(3) Finally, reactor system pressure is continuously monitored in the control room during reactor operation.

REFERENCES

(1) FDSAR, Volume I, Section IV.
(2) NEDO-24195, General Electric Reload Fuel Application for Oyster Creek.
(3) FDSAR, Volume I, Section III-2.3.3

2.3 LIMITING SAFETY SYSTEM SETTINGS

Applicability: Applies to trip settings on automatic protective devices related to variables on which safety limits have been placed.

Objective: To provide automatic corrective action to prevent the safety limits from being exceeded.

Specification: Limiting safety system settings shall be as follows:

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
A. Neutron Flux, Scram	
A.1 APRM	When the reactor mode switch is in the Run position, the APRM flux scram setting shall be

$$S \leq [(1.34 \times 10^{-6})\ W + 34.0]\left[\frac{FRP}{MFLPD}\right]$$

with a maximum setpoint of 115.7% for core flow equal to 61 x 10^6 lb/hr and greater,

where:

S = setting in percent of rated power
W = recirculation flow (lb/hr)

FRP = fraction of rated thermal power is the ratio of core thermal power to rated thermal power

MFLPD = maximum fraction of limiting power density where the limiting power density for each bundle is the design linear heat generation rate for that bundle.

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
	The ratio of FRP/MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0 in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow reference APRM High Flux Scram Curve by the reciprocal of the APRM gain change.
A.2 IRM	$\leq$15 percent of rated neutron flux
B) Neutron Flux, Control Rod Block	
	The Rod Block setting shall be $S \leq [(1.34 \times 10^{-6}) W + 24.3] [\frac{FRP}{MFLPD}]$ with a maximum setpoint of 106% for core flow equal to 61 x 10^6 lb/hr and greater. The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.

FUNCTION		LIMITING SAFETY SYSTEM SETTINGS
C.	Reactor High, Pressure, Scram	≤1060 psig
D.	Reactor High Pressure, Relief Valves Initiation	2 @ ≤1070 psig 3 @ ≤1090 psig
E.	Reactor High Pressure, Isolation Condenser Initiation	≤1060 psig with time delay ≤3 seconds
F.	Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig 4 @ 1221 psig 4 @ 1230 psig 4 @ 1239 psig ± 12 psi
G.	Low Pressure Main Steam Line, MSIV Closure	≥825 psig
H.	Main Steam Line Isolation Valve Closure, Scram	≤10% Valve Closure from full open
I.	Reactor Low Water Level, Scram	≤11'5" above the top of the active fuel as indicated under normal operating conditions
J.	Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	≤7'2" above the top of the active fuel as indicated under normal operating conditions
K.	Reactor Low-Low Water Level, Core Spray Initiation	≤7'2" above the top of the active fuel
L.	Reactor Low-Low Water Level, Isolation Condenser Initiation	≤7'2" above the top of the active fuel with time delay ≤3 seconds
M.	Turbine Trip, Scram	10 percent turbine stop valve(s) closure from full open
N.	Generator Load Rejection, Scram	Initiate upon loss of oil pressure from turbine acceleration relay

Bases:

Safety limits have been established in Specifications 2.1 and 2.2 to protect the integrity of the fuel cladding and reactor coolant system barriers, respectively. Automatic protective devices have been provided in the plant design for corrective actions to prevent the safety limits from being exceeded in normal operation or operational transients caused by reasonably expected single operator error or equipment malfunction. This Specification establishes the trip settings for these automatic protection devices.

The Average Power Range Monitor, APRM(1), trip setting has been established to assure never reaching the fuel cladding integrity safety limit. The APRM system responds to changes in neutron flux. However, near the rated thermal power, the APRM is calibrated using a plant heat balance, so that the neutron flux that is sensed is read out as percent of the rated thermal power. For slow maneuvers, such as those where core thermal power, surface heat flux, and the power transferred to the water follow the neutron flux, the APRM will read reactor thermal power. For fast transients, the neutron flux will lead the power transferred from the cladding to the water due to the effect of the fuel time constant. Therefore, when the neutron flux increases to the scram setting, the percent increase in heat flux and power transferred to the water will be less than the percent increase in neutron flux.

The APRM trip setting will be varied automatically with recirculation flow, with the trip setting at the rated flow of 61.0×10^6 lb/hr or greater being 115.7% of rated neutron flux. Based on a complete evaluation of the reactor dynamic performance during normal operation as well as expected maneuvers and the various mechanical failures, it was concluded that sufficient protection is provided by the simple fixed scram setting (2,3). However, in response to expressed beliefs (4) that variation of APRM flux scram with recirculation flow is a prudent measure to ensure safe plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit and yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.3.A, when the MFLPD is greater than the fraction of the rated power (FRP). The adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

For operation in the STARTUP mode while the reactor is at low pressure, the IRM scram setting of ≤15% of the rated power provides adequate thermal margin between the maximum power and the safety limit of 25% rated to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power low flow condition. Effects of increasing pressure at zero or low void content are minor, because cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a constrained rod pattern. In a sequenced rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of the rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram. The Reactor Protection System is designed such that reactor pressure must be above 825 psig to successfully transfer into the RUN mode, thus assuring protection for the fuel cladding safety limit.

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 gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. 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 106% of the 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 incore LPRM system. As with APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of the rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gains.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (5). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of turbine stop valve closure while operating at 1930 MWt, followed by [1] a delay of all scrams, [2] failure of the turbine bypass valves to open, and [3] failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 16 safety valves are required to turn the pressure transient. The ASME B&PV Code allows a $\pm 1\%$ of working pressure (1250 psig) variation in the lift point of the valves. This variation is recognized in Specification 4.3.

The low pressure isolation of the main steam lines at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down 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 825 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.

The low water level trip setting of 11'5" above the top of the active fuel has been established to assure that the reactor is not operated at a water level below that for which the fuel cladding integrity safety limit is applicable. With the scram set at this point, the generation of steam, and thus the loss of inventory, is stopped. For example, for a loss of feedwater flow a reactor scram at the value indicated and isolation valve closure at the low-low water level set point results in more than 4 feet of water remaining above the core after isolation (6).

During periods when the reactor is shut down, decay heat is present and adequate water level must be maintained to provide core cooling. Thus, the low-low level trip point of 7'2" above the core is provided to actuate the core spray system to provide cooling water should the level drop to this point. In addition, the normal reactor feedwater system and control rod drive hydraulic system provide protection for the water level safety limit both when the reactor is operating at power and in the shutdown condition.

The turbine stop valve(s) scram is provided to anticipate the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system.

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 to a load rejection and failure of the turbine bypass system. This scram is initiated by the loss of turbine acceleration relay oil pressure. The timing for this scram is almost identical to the turbine trip.

References

(1) FDSAR, Volume I, Section VII-4.2.4.2
(2) FDSAR, Amendment 28, Item III.A-12
(3) FDSAR, Amendment 32, Question 13
(4) Letters, Peter A. Morris, Director, Division of Reactor Licensing, USAEC to John E. Logan, Vice President, Jersey Central Power and Light Company, dated November 22, 1967 and January 9, 1968
(5) FDSAR, Amendment 65, Section B.XI.
(6) FDSAR, Amendment 65, Section B.IX.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in Which Function must be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Sys.	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
K. Rod Block								No con-trol rod withdrawals permitted
1. SRM Upscale	$\leq 5 \times 10^5$ cps		X	X(1)		1	2	
2. SRM Downscale	$\leq$ 100 cps (f)		X	X(1)		1	2	
3. IRM Downscale	$\leq$ 5/125 fullscale(g)		X	X		2	3	
4. APRM Upscale	**		X(s)	X	X	2	3(c)	
5. APRM Downscale	$\leq$ 2/150 fullscale				X	2	3(c)	
6. IRM Upscale	$\leq$ 108/125 fullscale		X	X		2	3	
L. Condenser Vacuum Pump Isolation								Insert con-trol rods
1. High radia-tion in Main Steam Tunnel	$\leq$ 10 x Normal Background		During Startup and run when vacuum pump is operating			2	2	
N. Diesel Generator Load Sequence	Time delay after energiz. of relay							
1. Containment Spray Pump	40 sec $\pm$ 15%	X	X	X	X	2(m)	1(n)	Consider containment spray loop inoperable and comply with Spec. 3.4.C (See Note q).

TABLE 3.1.1 (CONTD)

* Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. When necessary to conduct tests and calibrations, one channel may be made inoperable for up to one hour per month without tripping its trip system.

** See Specification 2.3 for Limiting Safety System Settings.

Notes:

a. Permissible to bypass, with control rod block, for reactor protection system reset in refuel mode.

b. Permissible to bypass below 800 psia in refuel and startup modes.

c. One (1) APRM in each operable trip system may be bypassed or inoperable provided the requirements of specification 3.1.C and 3.10.C are satisfied. Two APRM's in the same quadrant shall not be concurrently bypassed except as noted below or permitted by note.

   Any one APRM may be removed from service for up to one hour for test or calibration without inserting trips in its trip system only if the remaining operable APRM's meet the requirements of specification 3.1.B.1 and no control rods are moved outward during the calibration or test. During this short period, the requirements of specifications 3.1.B.2, 3.1.C and 3.10.C need not be met.

d. The IRM shall be inserted and operable until the APRM's are operable and reading at least 2/150 full scale.

e. Air ejector isolation valve closure time delay shall not exceed 15 minutes.

f. Unless SRM chambers are fully inserted.

g. Not applicable when IRM on lowest range.

h. One instrument channel in each trip system may be inoperable provided the circuit which it operates in the trip system is placed in a simulated tripped condition. If repairs cannot be completed within 72 hours the reactor shall be placed in the cold shutdown condition. If more than one instrument channel in any trip system becomes inoperable in the reactor shall be placed in the cold shutdown condition. Relief valve controllers shall not be bypassed for more than 3 hours (total time for all controllers) in any 30-day period and only one relief valve controller may be bypassed at a time.

i. The interlock is not required during the startup test program and demonstration of plant electrical output but shall be provided following these actions.

TABLE 3.1.1 (CONT'D)

j. Not required below 40% of turbine rated steam flow.

k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that primary containment integrity is not required and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.

l. Bypassed in IRM Ranges 8, 9, & 10.

m. There is one time delay relay associated with each of two pumps.

n. One time delay relay per pump must be operable.

o. There are two time delay relays associated with each of two pumps.

p. Two time delay relays per pump must be operable.

q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.

r. Time delay starts after closing of containment spray pump circuit breaker.

s. These functions not required to be operable with the reactor temperature less than 212°F and the vessel head removed or vented.

t. These functions may be inoperable or bypassed with corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.

u. These functions not required to be operable when primary containment integrity is not required to be maintained.

v. These functions not required to be operable when the ADS is not required to be operable.

w. These functions must be operable only when irradiated fuel is in the fuel pool or reactor vessel and secondary containment integrity is required per specification 3.5.B.

3.2 REACTIVITY CONTROL

Applicability: Applies to core reactivity and the operating status of the reactivity control systems for the reactor.

Objective: To assure reactivity control capability of the reactor.

Specification:

A. Core Reactivity

The core reactivity shall be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

B. Control Rod System

1. The control rod drive housing support shall be in place during power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.2.A is met.

2. The Rod Worth Minimizer (RWM) shall be operable during each reactor startup until reactor power reaches 10% of rated power except as follows:

   (a) Should the RWM become inoperable after the first twelve rods have been withdrawn, the startup may continue provided that a second licensed operator verifies that the licensed operator at the reactor console is following the rod program.

   (b) Should the RWM be inoperable before a startup is commenced or before the first twelve rods are withdrawn, one startup during each calendar year may be performed without the RWM provided that the second licensed operator verifies that the licensed operator at the reactor console is following the rod program and provided that a reactor engineer from the Core Engineering Group also verifies that the rod program is being followed. A startup without the RWM as described in this subsection shall be reported in a special report to the Nuclear Regulatory Commission (NRC) within 30 days of the startup stating the reason for the failure of the RWM, the action taken to repair it and the schedule for completion of the repairs.

Control rod withdrawal sequences shall be established so that the maximum reactivity that could be added by dropout of any increment of any one control blade would be such that the rod drop accident design limit of 280 cal/gm is not exceeded.

3. The average of the scram insertion times of all operable control rods shall be no greater than:

Rod Length Inserted (Percent)	Insertion Time (Seconds)
5	0.375
20	0.900
50	2.00
90	5.00

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Rod Length Inserted (Percent)	Insertion Time (Seconds)
5	0.398
20	0.954
50	2.120
90	5.300

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.2.A are met. Time zero shall be taken as the de-energization of the pilot scram valve solenoids.

4. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a

shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. Inoperable control rods shall be valved out of service, in such positions that Specification 3.2.A is met. In no case shall the number of rods valved out of service be greater than six during the power operation. If this specification is not met, the reactor shall be placed in the shutdown condition.

5. Control Rods shall not be withdrawn for approach to criticality unless at least two source range channels have an observed count rate equal to or greater than 3 counts per second.

C. Standby Liquid Control System

1. The standby liquid control system shall be operable at all times when the reactor is not shutdown by the control rods such that Specification 3.2.A is met and except as provided in Specification 3.2.C.3.

2. The standby liquid control solution shall be maintained within the volume-concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3. If one standby liquid control system pumping circuit becomes inoperable during the RUN mode and Specification 3.2.A is met, the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is demonstrated daily to be operable.

D. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded and the discrepancy cannot be explained, the reactor shall be brought to the cold shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The NRC shall be notified within 24 hours of this situation in accordance with Specification 6.6.

Bases:

Limiting conditions of operation on core reactivity and the reactivity control systems are required to assure that the excess reactivity of the reactor core is controlled at all times. The conditions specified herein assure the capability to provide reactor shutdown from steady state and transient conditions and assure the capability of limiting reactivity insertion rates under accident conditions to values which do not jeopardize the reactor coolant system integrity or operability of required safety features.

The core reactivity limitation is required to assure the reactor can be shut down at any time when fuel is in the core. It is a restriction that must be incorporated into the design of the core fuel; it must be applied to the conditions resulting from core alterations; and it must be applied in determining the required operability of the core reactivity control devices. The basic criterion is that the core at any point in its operation be capable of being made subcritical in the cold, xenon-free condition with the operable control rod of highest worth fully withdrawn and all other operable rods fully inserted. At most times in core life more than one control rod drive could fail mechanically and this criterion would still be met.

In order to assure that the basic criterion will be satisfied an additional design margin was adopted; that the k_{eff} be less than 0.99 in the cold xenon-free condition with the rod of highest worth fully withdrawn and all others fully inserted. Thus the design requirement is $k_{eff} \angle 0.99$, whereas the minimum condition for operation is $k_{eff} \angle 1.0$ with the operable rod of highest worth fully withdrawn (1). This limit allows control rod testing at any time in core life and assures that the plant can be shut down by control rods alone.

Fuel bundles containing gadolinia as a burnable neutron absorber results in a core reactivity characteristic which increases with exposure, goes through a maximum and then decreases. Thus it is possible that a core could be more reactive later in the cycle than at the beginning. Satisfaction of the above criterion can be demonstrated conveniently only at the time of refueling since it requires the core to be cold and xenon-free. The demonstration is designed to be done at these times and is such that if it is successful, the criterion is satisfied for the entire subsequent fuel cycle. The criterion will be satisfied by demonstrating Specification 4.2.A at the beginning of each fuel cycle with the core in the cold, xenon-free condition. This demonstration will include consideration for the calculated reactivity characteristic during the following operating cycle and the uncertainty in this calculation.

The control rod drive housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure(2). 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 reactor coolant system. The support is not required when no fuel is in the core since no nuclear consequences could occur in the absence of fuel. The 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. The support is not required if all control rods are fully inserted since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod(3).

The Rod Worth Minimizer(4) provides automatic supervision of conformance to the specified control rod patterns. It serves as a back-up to procedural control of control rod worth. In the event that the RWM is out of service when required, a licensed operator can manually fulfill the control rod pattern conformance functions of the RWM in which case the normal procedural controls are backed up by independent procedural controls to assure conformance during control rod withdrawal. This allowance to perform a startup without the RWM is limited to once each calendar year to assure a high operability of the RWM which is preferred over procedural controls.

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 cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second licensed operator. 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 limits 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. Method and basis for the rod drop accident analyses are documented in Reference 5.

The control rod system is designed to bring the reactor subcritical from a scram signal at a rate fast enough to prevent fuel damage. Scram reactivity curve for the transient analyses is calculated and evaluated with each reload core. In the analytical treatment of the transients, 290 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 typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays when the pilot scram solenoid de-energizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.2.B.3. The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.2.A.

Control rods (6) which cannot be moved with control rod drive pressure are clearly indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperabe rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.2.A, which assures the core can be shutdown at all times with control rods. Before rod is valved out of service in a non-fully inserted position an analysis is performed to insure specification 3.2.A is met.

The number of rods permitted to be valved out of service could be many more than the six allowed by the specification, particularly late in the operating cycle; however, the occurrence of more than six could be indicative of a generic 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. Placing the reactor in the shutdown condition inserts the control rods and accomplishes the objective of the specifications on control rod operability. This operation is normally expected to be accomplished within eight hours.

The source range monitor (SRM) system (7) performs no automatic safety function. It does provide the operator with a visual indication of neutron level which is needed for knowledgeable and efficient reactor startup at low neutron levels. The results of the reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 cps assures that any transient 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 critical using homogeneous patterns of scattered control rods.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady state operating condition at any time in core life independent of the control rod system capabilities (8). If the reactor is shutdown by the control rod system and would be subcritical in its most reactive condition as required in Specification 3.2.A, there is no requirement for operability of this system. To bring the reactor from full power to cold shutdown, sufficient liquid control must be inserted to give a negative reactivity worth equal to the combined effects of rated coolant voids, fuel Doppler, xenon, samarium, and temperature change plus shutdown margin. This requires a boron concentration of 600 ppm in the reactor. An additional 25% boron, which results in an average boron concentration in the reactor of 750 ppm, is inserted to provide margin for mixing uncertainties in the reactor. The system is required to insert the solution in a time interval between 60-120 minutes to provide for good mixing in the reactor and to override the rate of reactivity insertion due to cooldown of the reactor following the xenon peak.

The liquid control tank volume-concentration requirements of Figure 3.2-1 assure that the above requirements for liquid control insertion are met with one 30 gpm liquid control pump. The point (1937 gal, 19.6% solution) [9] results in the required amount of solution being inserted into the reactor is not less than 60 minutes, and therefore, defines the maximum concentration-minimum volume requirement. The point (3737 gal, 10.3% solution) [9] results in the required amount of solution being injected into the reactor is not more than 120 minutes, and therefore, defines the minimum concentration requirement. The boundary joining these points results in the required amount of solution being inserted into the reactor in the interval 60-120 minutes. The maximum volume of 4213 gal is established by the tank capacity. The tank volume requirements include consideration for 137 gal of solution which is contained below the point where the pump takes suction from the tank and, therefore, cannot be inserted into the reactor. The range of solution volume during normal operation is expected to be 2387-2937 gal.

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution will be maintained at least 5°F above the saturation temperature to guard against precipitation. The 5°F margin is included in Figure 3.2-1. Temperature and liquid level alarms for the system are annunciated in the control room.

The acceptable time out of service for a standby liquid control system pumping circuit as well as other safety features is determined to be 10 days. However, the allowed time out of service for a standby liquid control system pumping circuit is conservatively set at 7 days in the specification. Systems are designed with redundancy to increase their availability and to provide backup if one of the components is temporarily out of service.

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory with expected inventory based on appropriately corrected past data. Experience at Oyster Creek and other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References:

(1) FDSAR, Volume I, Section III - 5.3.1
(2) FDSAR, Volume I, Section VI-3
(3) FDSAR, Volume I, Section III - 5.2.1
(4) FDSAR, Volume I, Section VII-9
(5) NEDO-24195, General Electric Reload Fuel Application for Oyster Creek.
(6) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
(7) FDSAR, Volume I, Sections VII - 4.2.2 and VII - 4.3.1
(8) FDSAR, Volume I, Section VI-4
(9) FDSAR, Amendment No. 55, Section 2

3.4 EMERGENCY COOLING

Applicability: Applies to the operating status of the emergency cooling systems.

Objective: To assure operability of the emergency cooling systems.

Specifications:

A. Core Spray System

1. The core spray system shall be operable at all times with irradiated fuel in the reactor vessel, except as otherwise specified in this section.

2. The absorption chamber water volume shall be at least 82,000 $ft.^3$ in order for the core spray system to be considered operable.

3. If one core spray system loop or its core spray header P instrumentation becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided the remaining loop has no inoperable components and is demonstrated daily to be operable.

4. If one of the redundant active loop components in the core spray system becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 15 days provided the other similar component in the loop is demonstrated daily to be operable. If two of the redundant active loop components become inoperable, the limits of Specification 3.4.A shall apply.

5. During the period when one diesel is inoperable, the core spray equipment connected to the operable diesel shall be operable.

6. If Specifications 3.4.A.3, 3.4.A.4, and 3.4.A.5 are not met, the reactor shall be placed in the cold shutdown condition. If the core spray

system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 10" above the top of the active fuel.

7. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is: (a) maintained in the cold shutdown condition or (b) in the refuel mode with the reactor coolant system maintained at less than 212°F and vented, and (c) no work is performed on the reactor vessel and connected systems that could result in lowering the reactor water level to less than 10" above the top of the active fuel. Reduced Core Spray System Availability is minimally defined as follows:

    a. At least one core spray pump, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations.

    b. The fire protection system is operable, and

    c. These systems are demonstrated to be operable on a weekly basis.

8. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is in the refuel mode with the reactor coolant system maintained at less than 212°F or in the startup mode for the purposes of low power physics testing. Reduced core spray system availability is defined as follows:

    a. At least one core spray pump in each loop, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves in each loop can be started or operated from the control room or from local control stations.

b. The fire protection system is operable and,

c. Each core spray pump and all components in 3.4.A.8a are demonstrated to be operable every 72 hours.

9. If Specifications 3.4.A.7 and 3.4.A.8 cannot be met, the requirements of Specification 3.4.A.6 will be met and work will be initiated to meet minimum operability requirements of 3.4.A.7 and 3.4.A.8.

10. The core spray system is not required to be operable when the following conditions are met:

    a. The reactor mode switch is locked in the "refuel" or "shutdown" position.

    b. (1) There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

       (2) The fire protection system is operable.

    c. The reactor coolant system is maintained at less than 212°F and vented.

    d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

e. (1) No work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 10" above the top of the active fuel and the condensate storage tank level is greater than thirty (30) feet (360,000 gallons). At least two redundant systems including core spray pumps and system components must remain operable as defined in d. above.

OR

(2) The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

NOTE: When filling the reactor cavity from the condensate storage tank and draining the reactor cavity to the condensate storage tank, the 30 foot limit does not apply provided there is sufficient amount of water to complete the flooding operation.

B. Automatic Depressurization System

1. Five electromatic relief valves of the automatic depressurization system shall be operable when the reactor water temperature is greater than 212°F and pressurized above 110 psig, except as specified in 3.4.B.2. The automatic pressure relief function of these valves (but not the automatic depressurization function) may be inoperable or bypassed during the system hydrostatic pressure test required by ASME Code Section XI, IS-500 at or near the end of each ten year inspection interval.

2. If at any time there are only four operable electromatic relief valves, the reactor may remain in operation for a period not to exceed 3 days provided the motor operated isolation and condensate

makeup valves in both isolation condensers are demonstrated daily to be operable.

3. If Specifications 3.4.B.1 and 3.4.B.2 are not met; reactor pressure shall be reduced to 110 psig or less, within 24 hours.

4. The time delay set point for initiation after coincidence of low-low-low reactor water level and high drywell pressure shall be set to exceed two minutes.

C. Containment Spray System and Emergency Service Water System

1. The containment spray system and the emergency service water system shall be operable at all times with irradiated fuel in the reactor vessel, except as specified in Specifications 3.4.C.3, 3.4.C.4, 3.4.C.6 and 3.4.C.8.

2. The absorption chamber water volume shall not be less than 82,000 ft^3 in order for the containment spray and emergency service water system to be considered operable.

3. If one emergency service water system loop becomes inoperable, its associated containment spray system loop shall be considered inoperable. If one containment spray system loop and/or its associated emergency service water system loop becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided the remaining containment spray system loop and its associated emergency service water system loop each have no inoperable components and are demonstrated daily to be operable.

4. If a pump in the containment spray system or emergency service water system becomes inoperable, the reactor may remain in operation for a period not to exceed 15 days provided the other similar pump is demonstrated daily to be operable. A maximum of two pumps may be inoperable

provided the two pumps are not in the same loop. If more than two pumps become inoperable, the limits of Specification 3.4.C.3 shall apply.

5. During the period when one diesel is inoperable, the containment spray loop and emergency service water system loop connected to the operable diesel shall have no inoperable components.

6. If primary containment integrity is not required (see Specification 3.5.A), the containment spray system may be made inoperable.

7. If Specifications 3.4.C.3, 3.4.C.4, 3.4.C.5 or 3.4.C.6 are not met, the reactor shall be placed in the cold shutdown condition. If the containment spray system or the emergency service water system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 10" above the top of the active fuel.

8. The containment spray system may be made inoperable during the integrated primary containment leakage rate test required by Specification 4.5, provided that the reactor is maintained in the cold shutdown condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor level to less than 10" above the top of the active fuel.

D. Control Rod Drive Hydraulic System

1. The control rod drive (CRD) hydraulic system shall be operable when the reactor water temperature is above 212°F except as specified in 3.4.D.2 below.

2. If one CRD hydraulic pump becomes inoperable when the reactor water temperature is above 212°F, the reactor may remain in operation for a period not to exceed 7 days provided the second CRD hydraulic pump is operating and is checked at least once every 8 hours. If this condition cannot be met, the reactor water temperature shall be reduced to 212°F.

E. Core Spray and Containment Spray Pump Compartments Doors

The core spray and containment spray pump compartments doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

F. Fire Protection System

1. The fire protection system shall be operable at all times with fuel in the reactor vessel except as specified in Specification 3.4.F.2.

2. If the fire protection system becomes inoperable during the run mode, the reactor may remain in operation provided both core spray system loops are operable with no inoperable components.

Bases:

This specification assures that adequate emergency core cooling capability is available when the core spray system is required. Based on the loss-of-coolant analysis for the worst line break, a core spray of at least 3400 gpm is required within 35 seconds to assure effective core cooling.*(1) Thus, if one loop becomes inoperable, the operable loop is capable of providing cooling to the core and the reactor may remain in operation for a period of 7 days provided repairs can be completed within that time. The 7 days is based upon the consideration discussed in the bases of Specification 3.2 and the pump operability tests of Specification 4.4. If repairs cannot be made, the reactor is depressurized and vented to prevent pressure buildup and no work is allowed to be performed on the reactor which could result in lowering the water level below 10" above the top of active fuel.

Each core spray loop contains redundant active components. Therefore, with the loss of one of these components the system is still capable of supplying

* Core Spray System 2 is required to deliver 3640 gpm.

rated flow and the system as a whole (both loops) can tolerate an additional single failure of one of its active components and still perform the intended function and prevent clad melt. Therefore, if a redundant active component fails, a longer repair period is justified based on the consideration given in the bases of Specification 3.2. The consideration indicates that for a one out of 4 requirement the time out of service would be

$$\frac{T}{1.71} = \frac{30 \text{ days}}{1.71} = 17.5 \text{ days}$$

Specification 3.4.A.5 ensures that if one diesel is out of service for repair, the core spray system loop on the other diesel must be operable with no components out of service. This ensures that the loop can perform its intended function, even assuming one of its active components fails. If this condition is not met, the reactor is placed in a condition where core spray is no longer required.

When the reactor is in the shutdown or refueling mode and the reactor coolant system is less than 212°F and vented and no work is being performed that could result in lowering the water level to less than 10" above the core, the likelihood of a leak or rupture leading to uncovering of the core is very low. The only source of energy that must be removed is decay heat and one day after shutdown this heat generation rate is conservatively calculated to be not more than 0.6% of rated power. Sufficient core spray flow to cool the core can be supplied by one core spray pump or one of the two fire protection system pumps under these conditions. When it is necessary to perform repairs on the core spray system components, power supplies or water sources, Specification 3.4.A.7 permits reduced cooling system capability to that which could provide sufficient core spray flow from two independent sources. Manual initiation of these systems is adequate since it can be easily accomplished within 15 minutes during which time the temperature rise in the reactor will not reach 2200°F.

In order to allow for certain primary system maintenance, which will include control rod drive repair, LPRM removal/installation, reactor leak test, etc., (all performed according to approved procedure), Specification 3.4.A.8 requires the availability of an additional core spray pump in an independent loop, while this maintenance is being performed the likelihood of the core being uncovered is still considered to be very low, however. the requirement of a second core spray pump capable of full rated flow and the 72 hour operability demonstration of both core spray pumps is specified.

Specification 3.4.A.10 allows the core spray system to be inoperable in the cold shutdown or refuel modes if the reactor cavity is flooded and the spent fuel pool gates are removed and a source of water supply to the reactor vessel is available. Water would then be available to keep the core flooded.

The relief valves of the automatic depressurization system enable the core spray system to provide protection against the small break in the event the feedwater system is not active.

The containment spray system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. The flow from one pump in either loop is more than ample to provide the required heat removal capability (2). The emergency service water system provides cooling to the containment spray heat exchangers and, therefore, is required to provide the ultimate heat sink for the energy release in the event of a loss-of-coolant accident. The emergency service water pumping requirements are those which correspond to containment cooling heat exchanger performance implicit in the containment cooling description. Since the loss-of-coolant accident while in the cold shutdown condition would not require containment spray, the system may be deactivated to permit integrated leak rate testing of the primary containment while the reactor is in the cold shutdown condition.

The control rod drive hydraulic system can provide high pressure coolant injection capability. For break sizes up to 0.002 ft^2, a single control rod drive pump with flow of 110 gpm is adequate for maintaining the water level nearly five feet above the core, thus alleviating the necessity for auto-relief actuation(3).

The core spray main pump compartments and containment spray pump compartments were provided with water-tight doors(4). Specification 3.4.E ensures that the doors are in place to perform their intended function.

Similarly, since a loss-of-coolant accident when primary containment integrity is not being maintained would not result in pressure build-up in the drywell or torus, the system may be made inoperable under these conditions. This prevents possible personnel injury associated with contact with chromated torus water.

References

(1) Licensing Application, Amendment 34, Question 1

(2) Licensing Application, Amendment 32, Question 3

(3) Licensing Application, Amendment 18, Question 1

(4) Licensing Application, Amendment 18, Question 4

and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. 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, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The absorption chamber water volume provides the heat sink for the reactor coolant system energy released following the loss-of-coolant accident. The core spray pumps and containment spray pumps are located in the corner rooms and due to their proximity to the torus, the ambient temperature in those rooms could rise during the design basis accident. Calculations (7) made, assuming an initial torus water temperature of 100°F and a minimum water volume of 82,000 ft.^{3}, indicate that the corner room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits, and, therefore, would not adversely affect the long term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability (8) with a maximum water volume of 92,000 ft^3 is reduced by not more than 5.5% metal-water reaction below the capability with 82,000 ft^3.

Experimental data includes that excessive steam condensing leads 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. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

The technical specifications allow for torus repair work or inspections that might require draining of the suppression pool when all irradiated fuel is removed or when the potential for draining the reactor vessel has been minimized. This specification also provides assurance that the irradiated fuel has an adequate cooling water supply for normal and emergency conditions with the reactor mode switch in shutdown or refuel whenever the suppression pool is drained for inspection or repair.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the containment external design pressure limits are not exceeded.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100% vacuum relief breaker subsystems (2 parallel sets of 2 valves in series). Operation of either subsystem will maintain the containment external pressure less than the external design pressure of the drywell by 2 psi; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section 11).

The capacity of the fourteen suppression chambers to drywell vacuum relief valves is sized to limit the external pressure of the drywell during post-accident drywell cooling operations to the design limit of 2 psi. They are sized on the basis of the Bodega Bay pressure suppression tests.[9][10] In Amendment 15 of the Oyster Creek FDSAR, Section II, the area of 2920 sq. in. is stated as the minimum area for flow of non-condensible gases from the suppression chamber to the drywell. To achieve this requirement, at least 12 of the 14 vacuum breaker valves (18" diameter) must be operable.

Each suppression chamber drywell vacuum breaker is fitted with a redundant pair of limit switches to provide fail safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.1" at any point along the seal surface of the disk. These switches are capable of transmitting the disk closed-to-open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundancy of components and circuits and, most importantly, the accessibility of the valve lever arm and position reference external to the valve. The fail-safe feature of the alarm circuits assures operator attention if a line fault occurs.

Conservative estimates of the hydrogen produced, consistent with the core cooling system provided, show that the hydrogen air mixture resulting from a loss-of-coolant accident is considerably below the flammability limit and hence it cannot burn, and inerting would not be needed. However, inerting of the primary containment was included in the proposed design and operation. The 5% oxygen limit is the oxygen concentration limit stated by the American Gas Association for hydrogen-oxygen mixtures below which combustion will not occur.

To preclude the possibility of starting up the reactor and operating a long period of time with a significant leak in the primary system, leak checks must be made when the system is at or near rated temperature and pressure. It has been shown[9][10] that an acceptable margin with respect to flammability exists without containment inerting. Inerting the primary containment provides additional margin to that already considered acceptable. Therefore, permitting access to the drywell for the purpose of leak checking would not reduce the margin of safety below that considered adequate and is judged prudent in terms of the added plant safety offered by the opportunity for leak inspection. The 24-hour time to provide inerting is judged to be a reasonable time to perform the operation and establish the required O_2 limit.

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 11) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to define precisely an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.5.A.7.d prohibits startup with inoperable snubbers.

Secondary containment[5] is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the overall containment system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations and whenever work is being performed on the reactor or its connected systems in the reactor building since their operation could result in inadvertent release of radioactive material.

The standby gas treatment system[6] filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs.

Two separate filter trains are provided each having 100% capacity.[6] If one filter train becomes inoperable, there is no immediate threat to secondary containment and reactor operation may continue while repairs are being made. Since the test interval for this system is one month (Specification 4.5), the time out-of-service allowance of 7 days is based on considerations presented in the Bases in Specification 3.2 for a one-out-of-two system.

References:
(1) FDSAR, Volume I, Section V-1
(2) FDSAR, Volume I, Section V-1.4.1
(3) FDSAR, Volume I, Section V-1.7
(4) Licensing Application, Amendment 11, Question III-25
(5) FDSAR, Volume I, Section V-2
(6) FDSAR, Volume I, Section V-2.4
(7) Licensing Application, Amendment 42
(8) Licensing Application, Amendment 32, Question 3
(9) Robbins, C. H., "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.
(10) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.
(11) Report H. R. Erickson, Bergen-Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed on August 2, 1976, which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system. The maintenance of a drywell-suppression chamber differential pressure within the range shown on Figure 3.5-1 with a suppression chamber water level corresponding to a downcomer submergence range of 3.0 to 5.3 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3.10 CORE LIMITS

Applicability: Applies to core conditions required to meet the Final Acceptance Criteria for Emergency Core Cooling Performance.

Objective: To assure conformance to the peak clad temperature limitations during a postulated loss-of-coolant accident as specified in 10 CFR 50.46 (January 4, 1974) and to assure conformance to the 14.5 KW/ft (for V and VB fuel) and 13.4 KW/ft (for P8x8R fuel) operating limits for local linear heat generation rate.

Specification:

A. Average Planar LHGR

During 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:

A.1 Fuel Types V and VB

The product of the maximum average planar LHGR (MAPLHGR) limit shown in Figures 3.10-1 (for 5-loop operation) and 3.10-2 (for 4-loop operation) and the axial MAPLHGR multiplier in Figure 3.10-3.

A.2 Fuel Type P8x8R

The maximum average planar LHGR (MAPLHGR) limit shown in Figure 3.10-4 (for 5-loop operation) and 3.10-5 (for 4-loop operation).

A.3 If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

B. Local LHGR

During 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:

B.1 Fuel Types V and VB

As calculated by the following equation;

$$LHGR \quad LHGR_d \left[1 - \frac{\Delta P}{P} \max \left(\frac{L}{LT} \right) \right]$$

Where: $LHGR_d$ = Limiting LHGR (=14.5)

$\frac{\Delta P}{P}$ = Maximum Power Spiking Penalty (=0.033 and 0.039 for Fuel Types V and VB respectively)

LT = Total Core Length - 144 inches

L = Axial position above bottom of core

B.2 Fuel Type P8x8R

LHGR 13.4 KW/ft.

B.3 If at any time during operation it is determined by normal surveillance that the limiting value of LHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be greater than or equal to the following:

APRM Status	MCPR Limit
1. If any two (2) LPRM assemblies which are input to the APRM system and are separated in distance by less than three (3) times the control rod pitch contain a combination of (3) out of four (4) detectors located in either the A and B or C and D levels which are failed or bypassed i.e., APRM channel or LPRM input bypassed or inoperable.	1.30
2. If any LPRM input to the APRM system at the B, C, or D level is failed or bypassed or any APRM channel is inoperable (or bypassed).	1.30
3. All B, C, and D LPRM inputs to the APRM system are operating and no APRM channels are inoperable or bypassed.	1.30

When APRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, provided that the control rod block is placed in operation during this interval.

For core flows other than rated, the nominal value for MCPR shall be increased by a factor of k_f, where k_f is as shown in Figure 3.10-6.

If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated 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, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

Bases:

The Specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46 (January 4, 1974) considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected location 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 linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46 (January 4, 1974).

The maximum average planar LHGR limits of fuel types V and VB are shown in Figure 3.10-1 for five loop operation and in Figure 3.10-2 for four loop operation, and are the result of LOCA analyses performed by Exxon Nuclear Company utilizing an evaluation model developed by Exxon Nuclear Company in compliance with Appendix K to 10 CFR 50 (1). Operation is permitted with the four-loop limits of Figure 3.10-2 provided the fifth loop has its discharge valve closed and its bypass and suction valves open. In addition, the maximum average planar LHGR limits shown in Figures 3.10-1 and 3.10-2 for Type V and VB fuel were analyzed with 100% of the spray cooling coefficients specified in Appendix K to 10 CFR Part 50 for 7 x 7 fuel. These spray heat transfer coefficients were justified in the ENC Spray Cooling Heat Transfer Test Program (2).

The maximum average planar LHGR limits of fuel type P8x8R are shown in Figure 3.10-4 for five loop operation and in Figure 3.10-5 for four loop operation, and are based on calculations employing the models described in Reference 3.

Power operation with LHGR's at or below those shown in Figures 3.10-4 and 3.10-5 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

The effect of axial power profile peak location for fuel types V and VB is evaluated for the worst break size by performing a series of fuel heat-up calculations. A set of multipliers is devised to reduce the allowable bottom skewed axial power peaks relative to center or above center peaked profiles. The major factors which lead to the lower MAPLHGR limits with bottom skewed axial power profiles are the change in canister quench time at the axial peak location and a deterioration in heat transfer during the extended downward flow period during blowdown. The MAPLHGR multiplier in Figure 3.10-3 shall only be applied to MAPLHGR determined by the evaluation model described in reference 1.

The possible effects of fuel pellet densification are:

1) creep collapse of the cladding due to axial gap formation;
2) increase in the LHGR because of pellet column shortening;
3) power spikes due to axial gap formation; and
4) changes in stored energy due to increased radial gap size.

Calculations show that clad collapse is conservatively predicted not to occur during the exposure lifetime of the fuel. Therefore, clad collapse is not considered in the analyses.

Since axial thermal expansion of the fuel pellets is greater than axial shrinkage due to densification, the analyses of peak clad temperatures do not consider any change in LHGR due to pellet column shortening. Although the formation of axial gaps might produce a local power spike at one location on any one rod in a fuel assembly, the increase in local power density would be on the order of only 2% at the axial midplane. Since small local variations in power distribution have a small effect on peak clad temperature, power spikes were not considered in the analysis of loss-of-coolant accidents[1].

Changes in gap size affect the peak clad temperatures by their effect on pellet clad thermal conductance and fuel pellet stored energy. Treatment of this effect combined with the effects of pellet cracking, relocation and subsequent gap closure are discussed in XN-174. Pellet-clad thermal conductance for Type V and VB fuel was calculated using the GAPEX model (XN-174).

The specification for local LHGR assures that the linear heat generation rate in any rod is less than the limiting linear heat generation rate even if fuel pellet densification is postulated. The power spike penalty for Type V and VB fuel is based on analyses presented in Facility Change Request No. 6 and FDSAR Amendment No. 76, respectively. The analysis assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

The power spike penalty for fuel type P8x8R is described in Reference 3.

The loss of coolant accident (LOCA) analyses are performed using an initial core flow that is 70% of the rated value. The rationale for use of this value of flow is based on the possibility of achieving full power (100% rated power) at a reduced flow condition. The magnitude of the reduced flow is limited by the flow relationship for overpower scram. The low flow condition for the LOCA analysis ensures a conservative analysis because this initial condition is associated with a higher initial quality in the core relative to higher flow-lower quality conditions at full power. The high quality-low flow condition for the steady-state core operation results in rapid voiding of the core during the blowdown period of the LOCA. The rapid degradation of the coolant conditions due to voiding results in a decrease in the time to boiling transition and thus degradation of heat transfer with consequent higher peak cladding temperatures. Thus, analysis of the LOCA using 70% flow and 102% power provides a conservative basis for evaluation of the peak cladding temperature and the maximum average planar linear heat generation rate (MAPLHGR) for the reactor.

The APRM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM status as a consequence of bypassed APRM channel and/or failed bypassed LPRM inputs. The results for the reference cycle (3) indicate that the steady state MCPR required to protect the minimum transient MCPR of 1.07 is 1.23 or higher for the worst case APRM status condition (APRM STATUS 1). This steady state limit conservatively applies to APRM status 2 and 3. The steady state MCPR values for APRM status conditions 1, 2, and 3 will be evaluated each cycle. In order to provide for a limit which is considered to be bounding to future operating cycles, the limits for each APRM status condition have been conservatively adjusted upward to 1.30. This is also the assumed value for LOCA analysis.

The time interval of eight (8) hours to adjust the steady state MCPR to account for a degradation in the APRM status is justified on the basis of instituting a control rod block which precludes the possibility of experiencing a rod withdrawal error transient since rod withdrawal is physically prevented. This time interval is adequate to allow the operator to either increase the MCPR to the appropriate value or to upgrade the status of the APRM system while in a condition which prevents the possibility of this transient occurring.

The steady-state MCPR limit was selected to provide margin to accommodate transients and uncertanties in monitoring the core operating state, manufacturing, and in the critical power correlation itself[3]. This limit was derived by addition of the ΔCPR for the most limiting abnormal operational transient caused by a single operator error of equipment malfunction to the fuel cladding integrity MCPR limit designated in Specification 2.1. Transients analyzed each fuel cycle will be evaluated with respect to the steady-state MCPR limit specified in this specification.

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient.

The K_f factor curves shown in Figure 3.10-6 were developed generically using the flow control line corresponding to rated thermal power at rated core flow and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the value of K_f.

REFERENCES

(1) XN-75-55-(A), XN-75-55, Supplement 1-(A), XN-75-55. Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek plant," April 1977.

(2) XN-75-36 (NP)-(A), XN-75-36(NP) Supplement 1-(A), "Spray Cooling Heat Transfer phase Test Results, ENC - 8 x 8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975.

(3) NEDO-24195, General Electric Reload Fuel Application for Oyster Creek.

FIGURE 3.10-1

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE
(FIVE LOOP OPERATION)

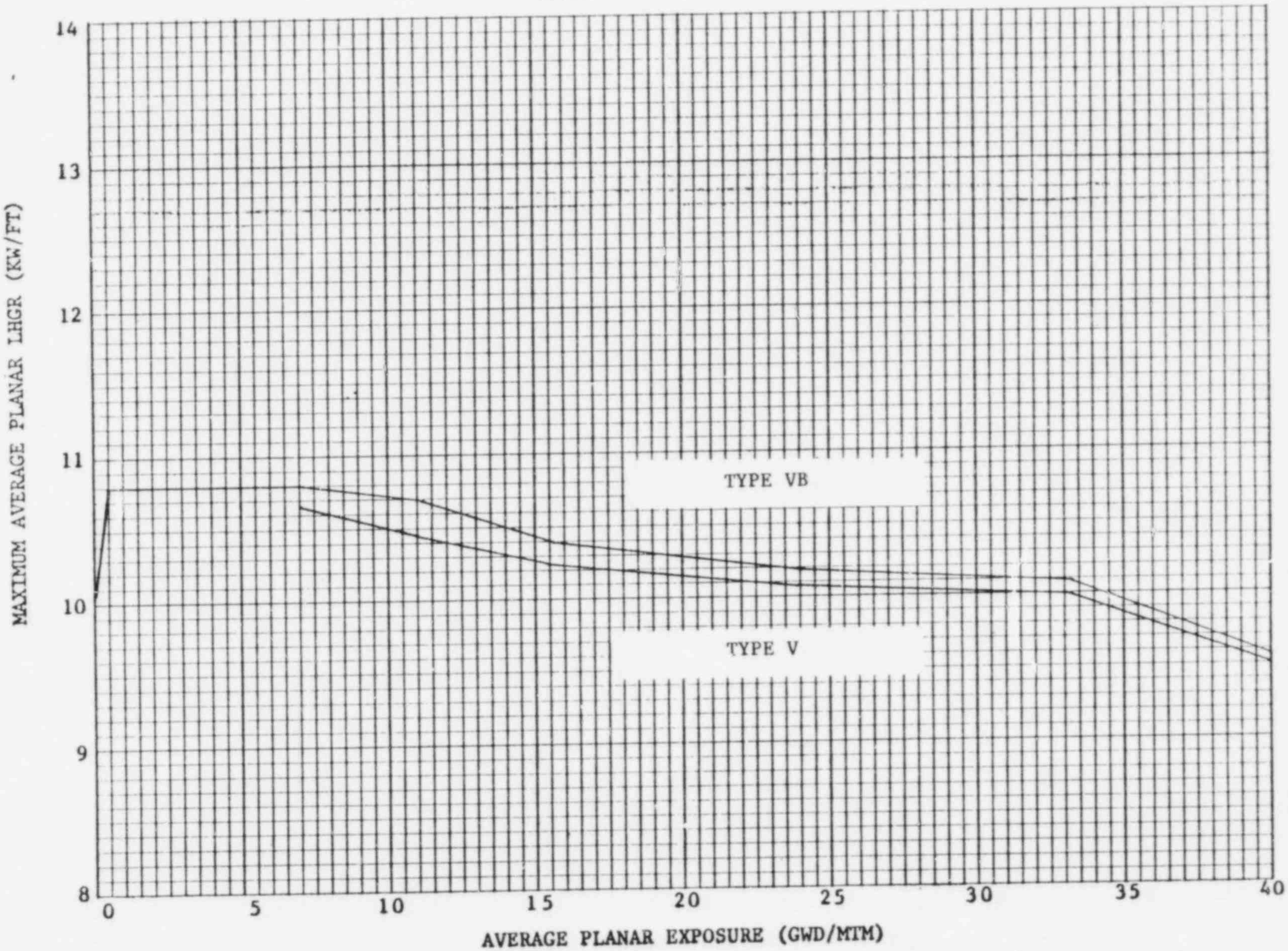


FIGURE 3.10-2

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE
(FOUR LOOP OPERATION)

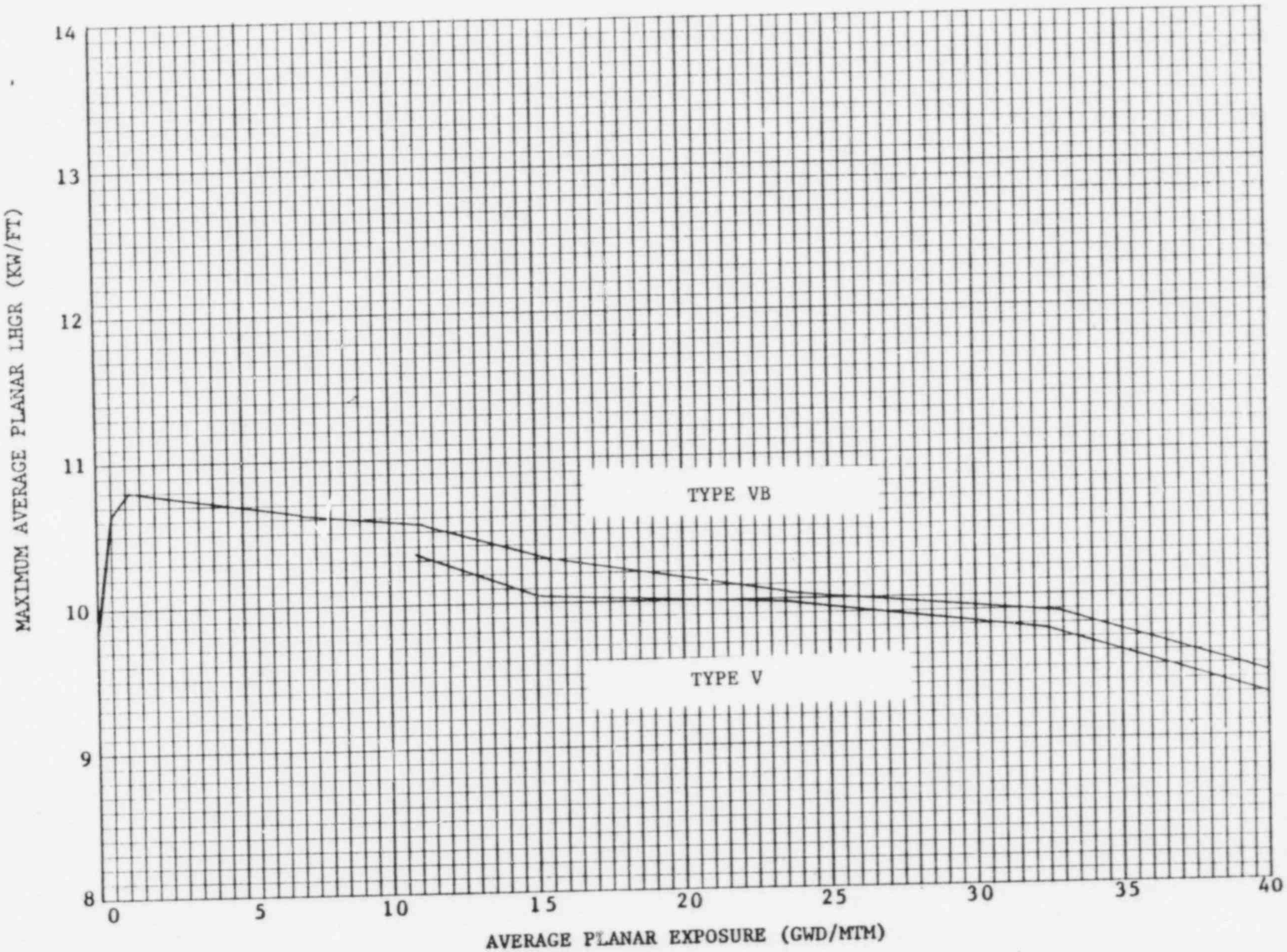


FIGURE 3.10-3

AXIAL MAPLHGR MULTIPLIER
(FOR FUEL TYPES V AND VB ONLY)

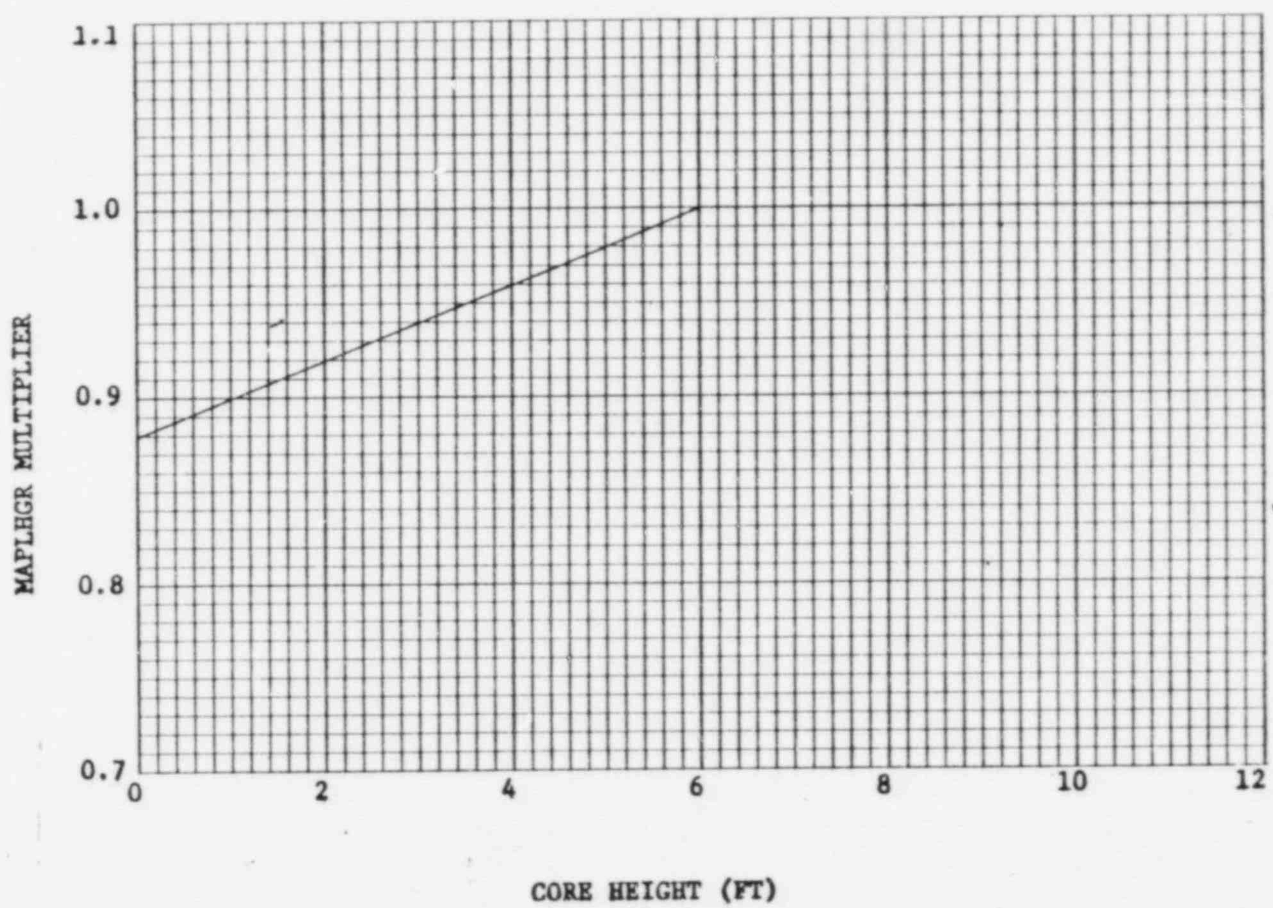


FIGURE 3.10-4

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAP HEAT GENERATION RATE
(FIVE LOOP OPERATION)

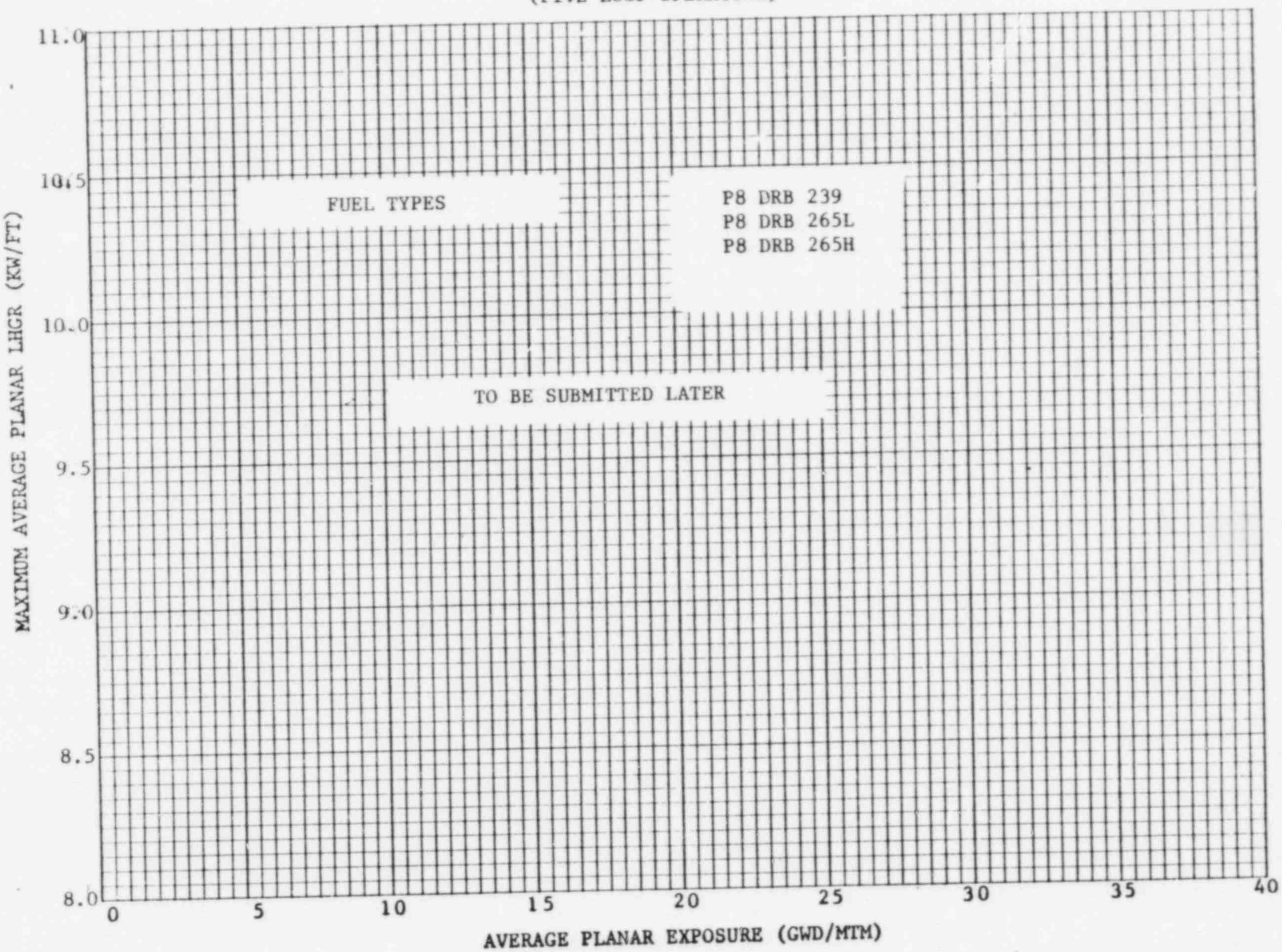


FIGURE 3.10-5

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE
(FOUR LOOP OPERATION)

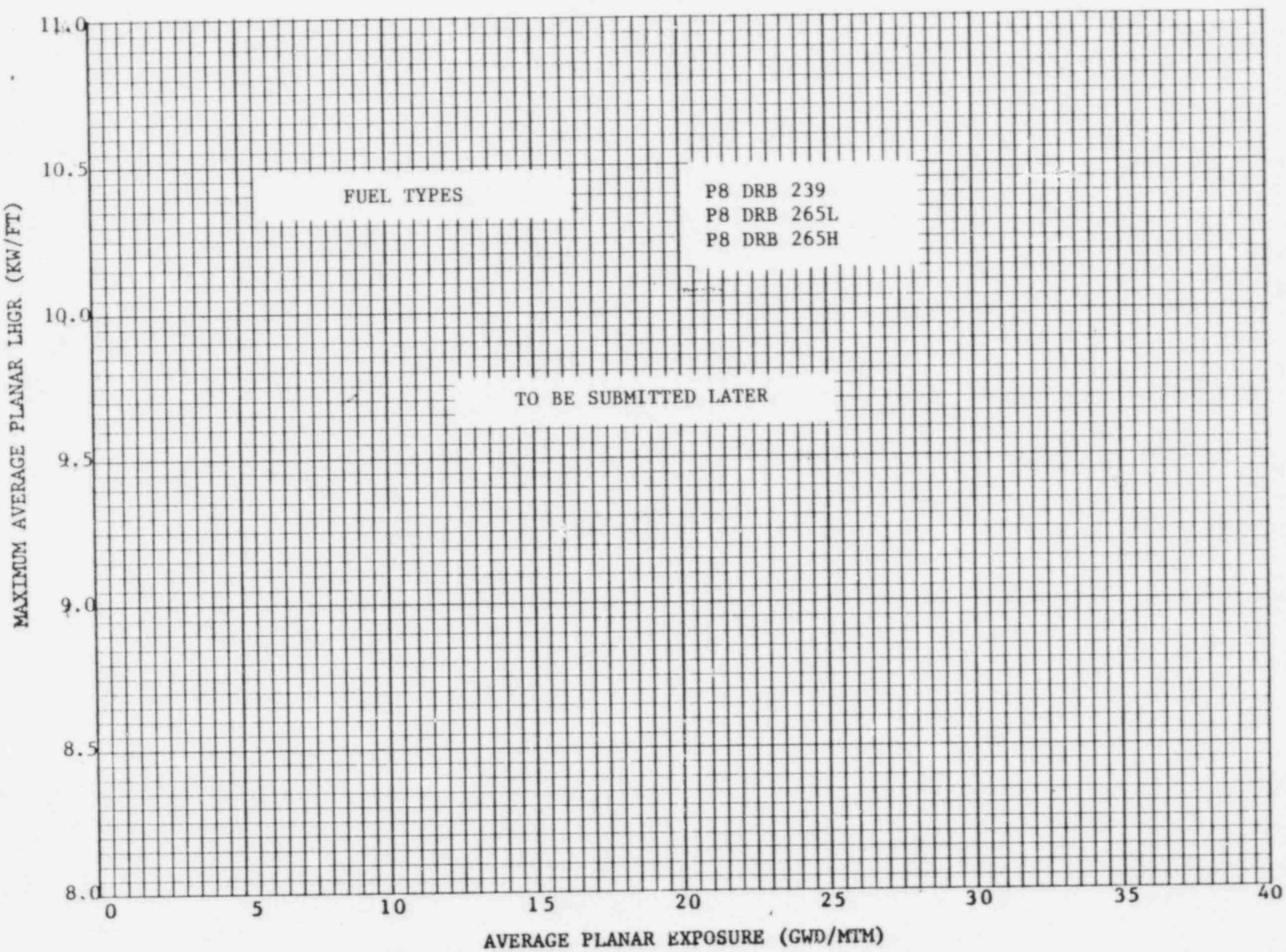
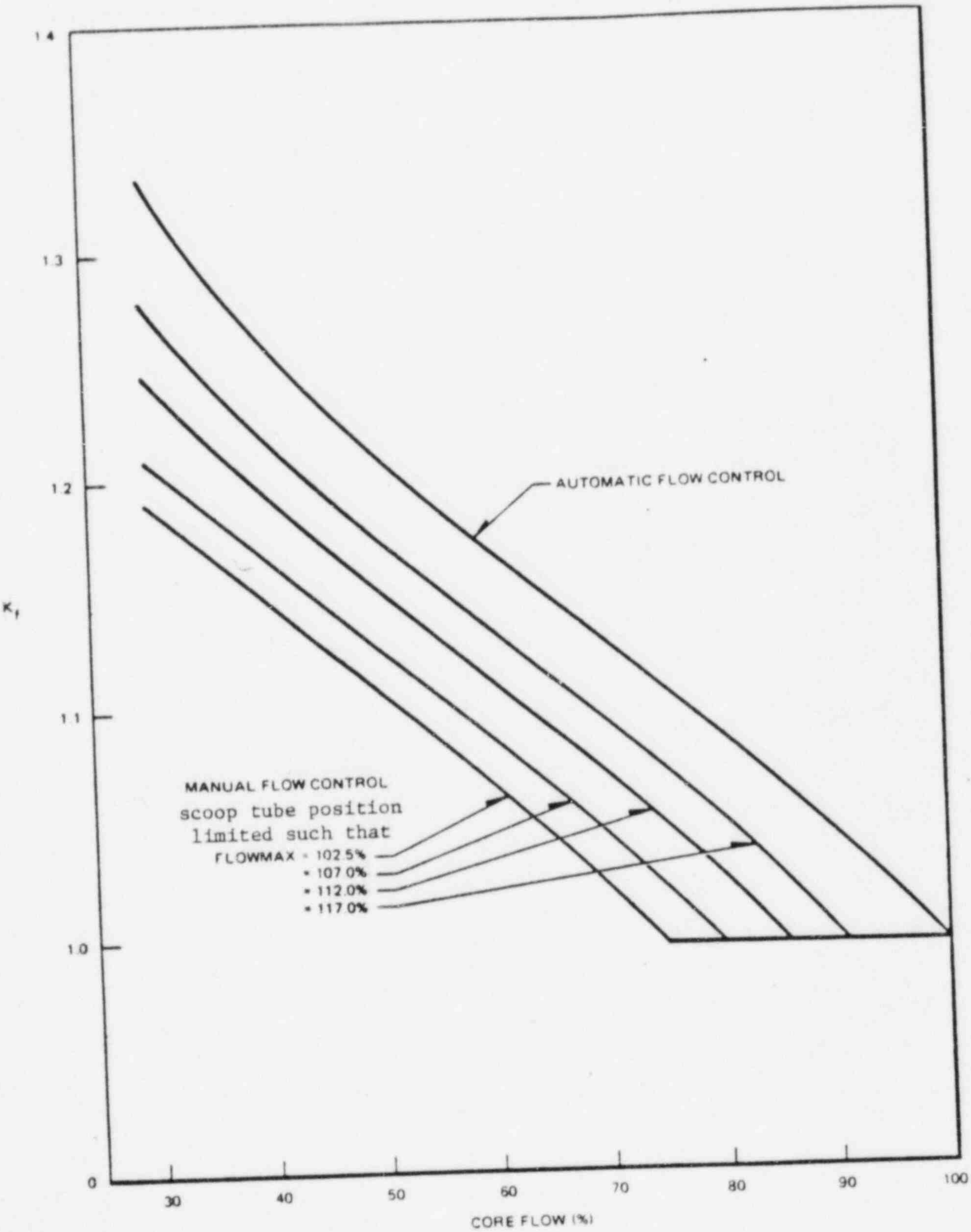


FIGURE 3.10-6 FLOW FACTOR, K_f



4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25% Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

B. The control rod drive housing support system shall be inspected after reassembly.

C. 1. After each major refueling outage and prior to resuming power operation, all operable control rods shall be scram time tested from the fully withdrawn position with reactor pressure above 800 psig.

2. Following each reactor scram from rated pressure, the mean 90% insertion time shall be determined for eight selected rods. If the mean 90% insertion time of the selected control rod drives does not fall within the range of 2.4 to 3.1 seconds or the measured scram time of any one drive for 90% insertion does not fall within the range of 1.9 to 3.6 seconds, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is maintained.

3. Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig provided these have not been measured in six months. The same criteria of 4.2.C(1) shall apply.

D. Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

E. Surveillance of the standby liquid control system shall be as follows:

1. Pump operability	Once/month
2. Boron concentration determination	Once/month
3. Functional test	Each refueling outage
4. Solution volume and temperature check	Once/month

F. At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed when equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

Basis:

The core reactivity limitation (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration is performed with the reactor core in the cold, xenon-free condition and will show that the reactor is sub-critical at that time by at least R + 0.25% Δk with the highest worth operable control rod fully withdrawn.

The value of R is the difference between two calculated values of reactivity of the cold, xenon-free core with the strongest operable control rod fully withdrawn. The reactivity value at the beginning of life is subtracted from the maximum reactivity value anytime later in life to determine R, which must be a positive quantity or its value is conservatively taken as zero. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all possibly inverted tubes present in the core. The value 0.25%Δk in the expression R + 0.25% Δk serves at the beginning of life as a finite, demonstrable shutdown margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of a diagonally adjacent rod to a position calculated to insert an R + 0.25% Δk reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R + 0.25% Δk margin beyond this.

The control rod drive housing support system(2) is not subject to deterioration during operation. However, reassembly must be assured following a partial or complete removal.

The scram insertion times for all control rods(3) will be determined at the time of each refueling outage. The scram times generated at each refueling outage when compared to scram times previously recorded gives a measurement of the functional effects of deterioration for each control rod drive. The more frequent scram insertion time measurements of eight selected rods are performed on a representative sample basis to monitor performance and give an early indication of possible deterioration and required maintenance. The times given for the eight-rod tests are based on the testing experience of control rod drives which were known to be in good condition.

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system. Experience with this control rod system has indicated that weekly tests are adequate, and that rods which move by drive pressure will scram when required as the pressure applied is much higher. The frequency of exercising the control rods has been increased under the conditions of two or more control rods which are valved out of service in order to provide even further assurance of the reliability of the remaining control rods.

Pump operability, boron concentration, solution temperature and volume of the standby liquid control system(4) are checked on a frequency consistent with instrumentation checks described in Specification 4.1. Experience with similar systems has indicated that the test frequencies are adequate. The only practical time to functionally test the liquid control system is during a refueling outage. The functional test includes the firing of explosive charges to open the shear plug valves and the pumping of demineralized water into the reactor to assure operability of the system downstream of the pumps. The test also includes recirculation of liquid control solution to and from the solution tanks.

Pump operability is demonstrated on a more frequent basis. This test consists of recirculation of demineralized water to a test tank. A continuity check of the firing circuit on the shear plug valves is provided by pilot lights in the control room. Tank level and temperature alarms are provided to alert the operator to off-normal conditions.

The functional test and other surveillance on components, along with the monitoring instrumentation, gives a high reliability for standby liquid control system operability.

<u>References</u>

(1) FDSAR, Volume II, Figure III-5-11
(2) FDSAR, Volume I, Section VI-3
(3) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
(4) FDSAR, Volume I, Section VI-4.

4.10 ECCS RELATED CORE LIMITS

Applicability: Applies to the periodic measurement during power operation of core parameters related to ECCS performance.

Objective: To assure that the limits of Section 3.10 are not being violated.

Specification:

A. Average Planar LHGR.

The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

B. Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR).

MCPR shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

Bases:

The LHGR shall be checked daily to determine whether fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

The minimum critical power ratio (MCPR) is unlikely to change significantly during steady state power operation so that 24 hours is an acceptable frequency for surveillance. In the event of a single pump trip, 24 hours surveillance interval remains acceptable because the accompanying power reduction is much larger than the change in MAPLHGR limits for four loop operation at the corresponding lower steady state power level as compared to five loop operation. The 24 hours frequency is also acceptable for the APRM status check since neutron monitoring system failures are infrequent and a downscale failure of either an APRM or LPRM initiates a control rod withdrawal block, thus precluding the possibility of a control rod withdrawal error.

At core power levels less than or equal to 25% rated thermal power the reactor will be operating at or above the minimum recirculation pump speed. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicate that the resulting APLHGR, LHGR and MCPR values all have considerable margin to the limits of Specification 3.10. Consequently, monitoring of these quantities below 25% of the rated thermal power is not required.