ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

TECHNICAL SPECIFICATION CHANGES

List of Attached Pages

Units 2 and 3

1, 2, 3, 4, 9, 9a, 10, 11, 11a, 15, 16, 19, 20, 35, 37, 40, 54, 73, 74, 74a, 133a, 133b, 140, 140b, 140c, 141a, 141b, 256

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud with the vessel head removed and fuel in the vessel.

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

Average Planar Linear Heat Generation Rate (APLHGR) - The APLHGR shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod, for all the fuel rods in the specific bundle at the specific height, divided by the number of fuel rods in the fuel bundle at that height.

Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Condition - Reactor coolant temperature equal to or less than 212 F.

Cold Shutdown - The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212 F, and the reactor vessel is vented to atmosphere.

Core Operating Limits Report (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current Operating Cycle. These cycle-specific core operating limits shall be determined for each Operating Cycle in accordance with specification 6.9.1.e. Plant operation within these limits is addressed in individual Specifications.

Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958).

Dose Equivalent I-131 - That concentration of I-131 (Ci/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

Downscale Trip Set Point (DTSP) - The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

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Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

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

Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

Gaseous Radwaste Treatment System - Any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

High (power) Trip Set Point (HPTS) - The high power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable above 85% reactor thermal power.

Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212 F.

Hot Standby Condition - Hot Standby Condition means operation with coolant temperature greater than 212 F, system pressure less than 1055 psig, and the mode switch in the Startup/Hot Standby position. The main steam isolation valves may be opened to provide steam to the reactor feed pumps.

Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

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Instrument or Channel Calibration - An instrument or channel calibration means the adjustment of an instrument or channel signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument or channel monitors. The known value of the parameter shall be injected into the channel or instrument as close to the primary sensor as practicable.

Instrument or Channel Check - An instrument or channel check is a qualitative determination of acceptable operability by observation of instrument or channel behavior during operation. This determination shall include, where possible, comparison of the instrument or channel with other independent instruments measuring the same variable.

Instrument or Channel Functional Test - An instrument or channel functional test means the injection of a simulated signal into the channel or instrument as close to the primary sensor as practicable to verify the proper instrument channel response, alarm and/or initiating action.

Intermediate (power) Trip Set Point (ITSP) - The intermediate power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 65% and 85% reactor thermal power.

Limiting Conditions for Operations (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

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

Logic - A logic is an arrangement of relays, contacts and other components that produces a decision output.

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(a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

Low (power) Trip Set Point (LTSP) - The low power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 30% and 65% reactor thermal power.

MAPFAC(F) (MAPLHGR Flow Factor) - A core flow dependent multiplication factor used to flow bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MAPFAC(P) (Power Dependent MAPLHGR Multiplier) - A core power dependent multiplication factor used to power bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MEMBERS OF THE PUBLIC - Members of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreation, occupational, or other purposes not associated with the plant.

Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core. Associated with the minimum critical power ratio is a core flow dependent (MCPR(F)) and core power dependent (MCPR(P)) minimum critical power ratio.

Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided: Refuel Mode, Run Mode, Shutdown Mode, Startup/Hot Standby Mode.

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SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objectives:

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

Specification:

A. Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation, or 1.07 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objectives:

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

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66W + 71\% - 0.66\,\Delta W$
(Clamp @ 120%)

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design.

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SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY	2.1 FUEL CLADDING INTEGRITY
	ΔW = Difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 ΔW) is accomplished by correcting the flow input of the flow biased scram to preserve the original (two loop) relationship between APRM scram setpoint and recirculation drive flow or by adjusting the APRM flux trip setting. ΔW = 0 for two loop operation. The APRM flux scram trip setting shall not exceed 120% of rated thermal power.

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SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING

2.1.A (Cont'd)

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

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

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SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia) When the reactor pressure is ≤ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.	B. APRM Rod Block Trip Setting $S_{RB} \leq (0.66\ W + 59\% - 0.66\ \Delta W)$ (Clamp @ 108%) where: S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of design. ΔW = Difference between two loop and single loop effective recirculation drive flow at the same core flow. During single loop operation, the reduction in trip setting ($-0.66\ \Delta W$) is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between APRM Rod block setpoint and recirculation drive flow or by adjusting the APRM Rod block trip setting. $\Delta W = 0$ for two loop operation. The APRM rod block trip setting shall not exceed 108% of rated thermal power.

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SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia)	B. APRM Rod Block Trip Setting
C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than minus 160 inches indicated level (378 inches above vessel zero).	C. Scram and isolation--reactor low water level ≥ 538 in. above vessel zero (0" on level instruments)

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1.1.C BASES (Cont'd.)

However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit, provided scram signals are operable, is supported by the extensive plant safety analysis.

The computer provided with Peach Bottom Units 2 and 3 has a sequence annunciation program which will indicate the sequence in which events such as a scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied upon to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at minus 160 inches indicated level (378 inches above vessel zero) provides adequate margin to assure sufficient cooling during shutdown conditions. This level will be continuously monitored.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, January 1977 (NEDO-10958-A).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340).

3. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

4. "Peach Bottom Atomic Power Station Units 2 and 3 Single-Loop Operation", NEDO-24229-1, May 1980.

5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993.

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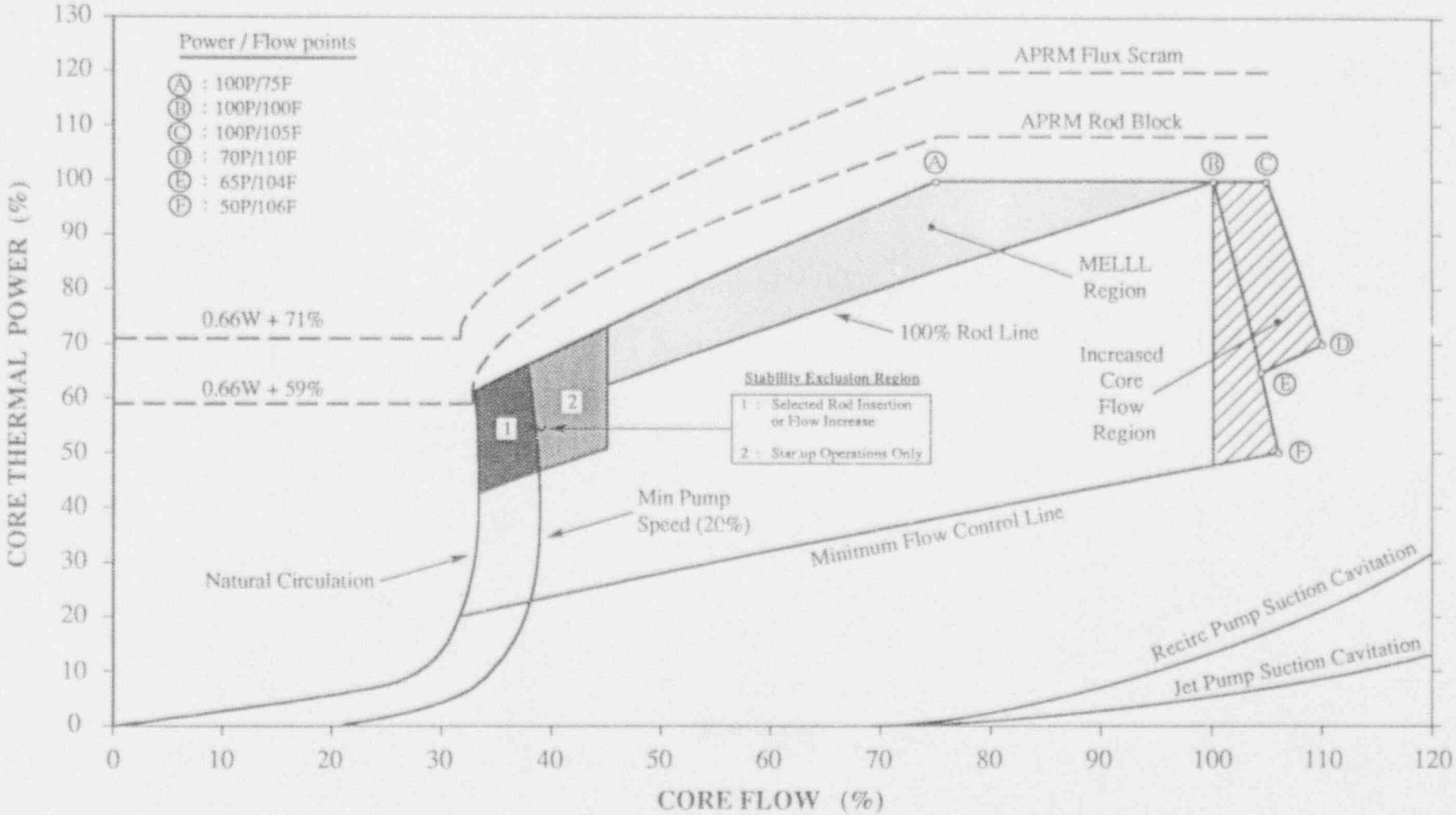
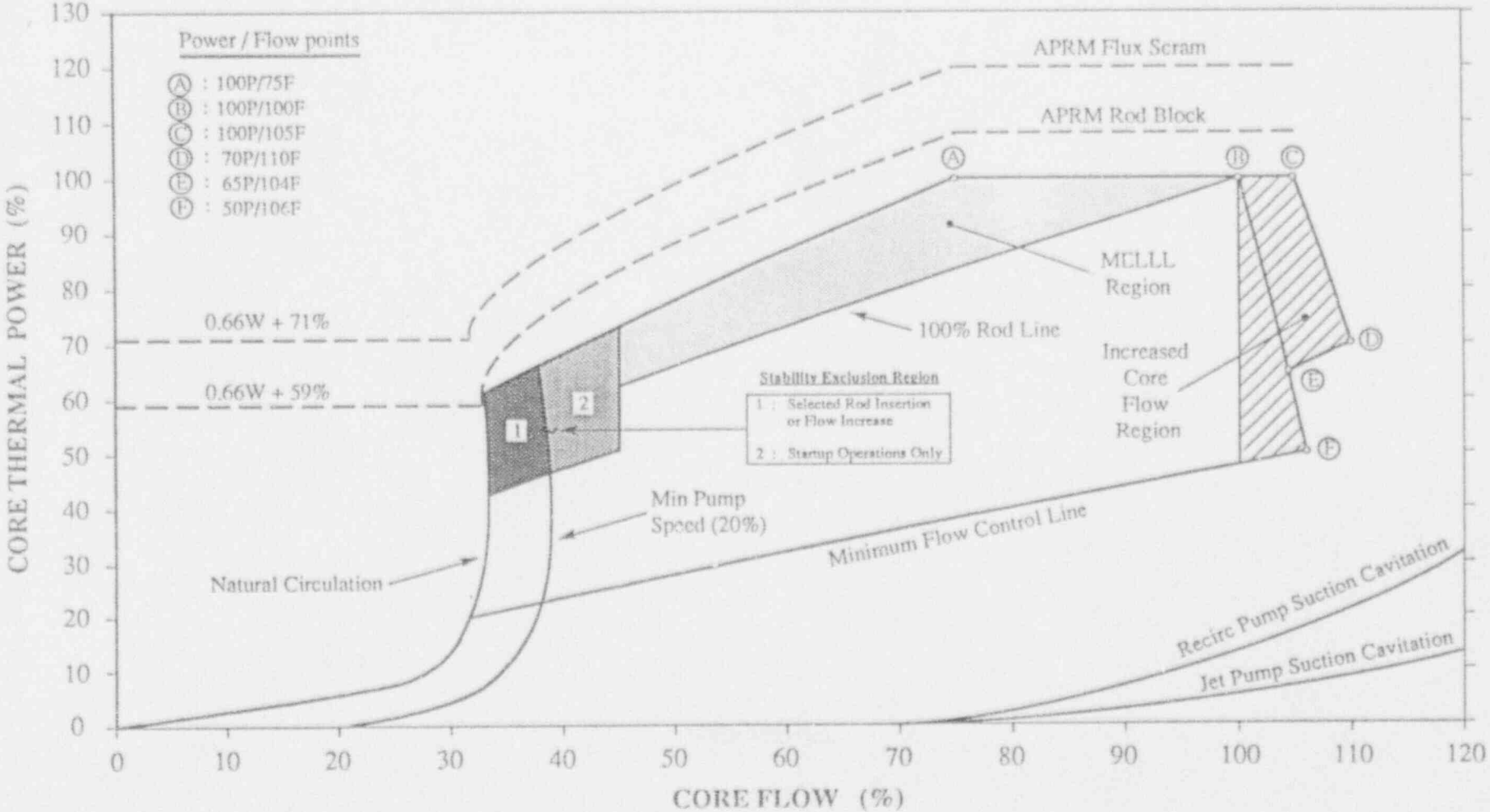


APRM Flow Bias Scram
Relationship to Normal Operating Conditions

Figure 1.1-1



APRM Flow Bias Scram
Relationship to Normal Operating Conditions

Figure 1.1-1

2.1.A BASES (Cont'd)

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 yet allows operating margin that reduces the possibility of unnecessary scrams.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than the fuel cladding integrity safety limit when the transient is initiated from MCPR greater than the operating limit given in Specification 3.5.K, adjusted for power and flow as specified in the COLR.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the Rod Worth Minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals dos not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of change of power is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when the reactor pressure is greater than 850 psig.

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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 yet allows operating margin that reduces the possibility of unnecessary scrams.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than the fuel cladding integrity safety limit when the transient is initiated from MCPR greater than the operating limit given in Specification 3.5.K, adjusted for power and flow as specified in the COLR.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the Rod Worth Minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals dos not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of change of power is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when the reactor pressure is greater than 850 psig.

2.1.A BASES (Cont'd.)

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5-decades are covered by the IRM by means of a range switch and the 5-decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to assure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scramed and peak power limited to one percent of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in-sequence and provides backup protection for the APRM.

B. APRM Rod Block Trip Setting

The APRM system provides a control rod block to avoid conditions which would result in an APRM scram trip if allowed to proceed. The APRM rod block trip setting, like the APRM scram trip setting, is automatically varied with recirculation loop flow rate. The flow variable APRM rod block trip setting provides margin to the APRM scram trip setting over the entire recirculation flow range.

2.1.A BASES (Cont'd.)

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5-decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

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B. APRM Rod Block Trip Setting

The APRM system provides a control rod block to avoid conditions which would result in an APRM scram trip if allowed to proceed. The APRM rod block trip setting, like the APRM scram trip setting, is automatically varied with recirculation loop flow rate. The flow variable APRM rod block trip setting provides margin to the APRM scram trip setting over the entire recirculation flow range.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective

To assure the operability of the reactor protection system.

Specification:

A. When there is fuel in the vessel the setpoint, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1.

B. The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds. Otherwise, the affected trip system shall be placed in the tripped condition, or the action listed in Table 3.1.1 for the specific trip function shall be taken.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective

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

Specification:

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

B. DELETED

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective

To assure the operability of the reactor protection system.

Specification:

A. When there is fuel in the vessel the setpoint, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1.

B. The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds. Otherwise, the affected trip system shall be placed in the tripped condition, or the action listed in Table 3.1.1 for the specific trip function shall be taken.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective

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

Specification:

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

B. DELETED

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Items	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes In which Function Must Be Operable			Number of Instrument Channels Provided by Design	Action (1)
				Refuel (7)	Startup	Run		
1	1	Mode Switch In Shutdown		x	x	x	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		x	x	x	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	x	x	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		x	x	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(0.66W+71%-0.66ΔW) (Clamp @ 120%) (12) (13)			x	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	x	x	x	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	x	x		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	x(9)	x	x	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	x(8)	x(8)	x	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	x	x	x	4 Instrument Channels	A

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				Refuel (7)	Startup	Run		
1	1	Mode Switch In Shutdown		x	x	x	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		x	x	x	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	x	x	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		x	x	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(0.66W+71%-0.66ΔW) (Clamp @ 120%) (12) (13)			x	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	x	x	x	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	x	x		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	x(9)	x	x	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	x(8)	x(8)	x	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	x	x	x	4 Instrument Channels	A

NOTES FOR TABLE 3.1.1 (Cont'd)

10. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

11. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 14 LPRM inputs of the normal complement.

12. W = Loop Recirculation flow in percent of design.

    Delta W = The difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 delta W) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. Delta W equals zero for two loop operation.

    Trip level setting is in percent of rated power (3293 MWt).

13. See Section 2.1.A.1.

NOTES FOR TABLE 3.1.1 (Cont'd)

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    Trip level setting is in percent of rated power (3293 MWt).

13. See Section 2.1.A.1.

4.1 BASES (Cont'd)

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

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

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches, and, hence, calibration during operation is not applicable.

B. The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating twice a week using heat balance data and by calibrating individual LPRM's every 6 weeks, using TIP traverse data.

4.1 BASES (Cont'd)

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
4 (2)	APRM Upscale (Flow Biased)	(0.66W+59%-0.66ΔW) (Clamp at 108% max)	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	≤12%	6 Inst. Channels	(10)
4	APRM Downscale	≥2.5 indicated on scale	6 Inst. Channels	(10)
1 (7)(11)	Rod Block Monitor (Power Biased)	(RTP ≥85%), S_{RB} ≤HTSP (65% ≤RTP <85%), S_{RB} ≤ITSP (30% ≤RTP <65%), S_{RB} ≤LTSP	2 Inst. Channels	(1)
1 (7)(11)	Rod Block Monitor Downscale	≥DTSP	2 Inst. Channels	(1)
6	IRM Downscale (3)	≥2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	≤108 indicated on scale	8 Inst. Channels	(10)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	≤10^5 counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Level	≤25 gallons	1 Inst. Channel	(9)

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4	APRM Downscale	≥2.5 indicated on scale	6 Inst. Channels	(10)
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1 (7)(11)	Rod Block Monitor Downscale	≥DTSP	2 Inst. Channels	(1)
6	IRM Downscale (3)	≥2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	≤108 indicated on scale	8 Inst. Channels	(10)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	≤10^5 counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Level	≤25 gallons	1 Inst. Channel	(9)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W = Loop Recirculation flow in percent of design.

   Trip level setting is in percent of rated power (3293 MWt).

   ΔW is the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow. $\Delta W = 0$ for two loop operation.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is $\leq$ 30%.

8. This function is bypassed when the mode switch is placed in Run.

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6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is $\leq$ 30%.

8. This function is bypassed when the mode switch is placed in Run.

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:

    a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

    b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

11. The values for HTSP, ITSP, LTSP, and DTSP are specified in the CORE OPERATING LIMITS REPORT.

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:

    a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

    b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

11. The values for HTSP, ITSP, LTSP, and DTSP are specified in the CORE OPERATING LIMITS REPORT.

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure and reactor power/flow multipliers (provided in the CORE OPERATING LIMITS REPORT) shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limit for the most limiting lattice (excluding natural uranium) specified in the CORE OPERATING LIMITS REPORT during two recirculation loop operations. If at any time during operation, it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure and reactor power/flow multipliers (provided in the CORE OPERATING LIMITS REPORT) shall be checked daily during reactor operation at ≥25% rated thermal power.

3.5.J 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 design LHGR.

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.

LHGR ≤ LHGRd

LHGRd = Design LHGR
The values for Design LHGR for each fuel type are specified in the CORE OPERATING LIMITS REPORT.

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure and reactor power/flow multipliers (provided in the CORE OPERATING LIMITS REPORT) shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limit for the most limiting lattice (excluding natural uranium) specified in the CORE OPERATING LIMITS REPORT during two recirculation loop operations. If at any time during operation, it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure and reactor power/flow multipliers (provided in the CORE OPERATING LIMITS REPORT) shall be checked daily during reactor operation at ≥25% rated thermal power.

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During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed design LHGR.

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.

LHGR ≤ LHGRd

LHGRd = Design LHGR
The values for Design LHGR for each fuel type are specified in the CORE OPERATING LIMITS REPORT.

LIMITING CONDITIONS FOR OPERATION

3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

1. During power operation the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Specification 3.5.K.2 or 3.5.K.3, or MCPR(F), or the MCPR operating limit as determined by application of MCPR(P), whichever is greater. MCPR(F) and MCPR(P) are provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥25% rated thermal power.

2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotation basis and performing the following:

a. The average scram time to the 20% insertion position shall be:

$$\tau \text{ ave} \leq \tau B$$

b. The average scram time to the 20% insertion position is determined as follows:

$$\tau \text{ ave} = \frac{\sum_{i=1}^{n} Ni\ \tau i}{\sum_{i=1}^{n} Ni}$$

where: n = number of surveillance tests performed to date in the cycle.

LIMITING CONDITIONS FOR OPERATION

3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

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SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥25% rated thermal power.

2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotation basis and performing the following:

a. The average scram time to the 20% insertion position shall be:

$$\tau \text{ ave} \leq \tau B$$

b. The average scram time to the 20% insertion position is determined as follows:

$$\tau \text{ ave} = \frac{\sum_{i=1}^{n} Ni \, \tau i}{\sum_{i=1}^{n} Ni}$$

where: n = number of surveillance tests performed to date in the cycle.

3.5 BASES (Continued)

H. Engineered Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicated that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR Part 50, Appendix K.

The peak cladding temperature (PCT) 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. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is the LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the applicable figure for each fuel type in the CORE OPERATING LIMITS REPORT, and must be adjusted for power and flow by application of MAPFAC(P) and MAPFAC(F). MAPFAC(P) and MAPFAC(F) are provided in the CORE OPERATING LIMITS REPORT.

Only the most limiting APLHGR operating limits are shown in the figures for the multiple lattice fuel types. Compliance with the lattice-specific APLHGR limits is ensured by using the process computer. When an alternate method to the process computer is required (i.e. hand calculations and/or alternate computer simulation), the most limiting lattice APLHGR limit for each fuel type shall be applied to every lattice of that fuel type.

The calculational procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (G.E.) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. The plant specific results using the Reference 4 methodology are presented in Reference 8.

3.5 BASES (Continued)

H. Engineered Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicated that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

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3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in References 7, 10 and 11. Input data and operating conditions used in this analysis are shown in References 7, 10 and 11 in the Supplemental Reload Licensing Analysis.

3.5.L. Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective action to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe - TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution, for up to 43 in-core locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indication value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.K. BASES (Cont'd)

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3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate including MELLL operation with implementation of ARTS restrictions (Ref. 11).

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. Letter, C. O. Thomas (NRC) to J. F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), 'The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident'," June 1, 1984.

5. DELETED.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analyses," NEDC-32163P, January, 1993.

9. DELETED.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

11. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993.

3.5.L. BASES (Cont'd)

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11. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993.

4.5.K Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

4.5.L MCPR Limits for Core Flows Other Than Rated

A flow dependent MCPR limit, MCPR(F), is necessary to assure that the safety limit MCPR is not violated during recirculation flow increase events. The design basis flow increase event is a slow-power increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Flow runout events are analyzed along a constant xenon flow control line assuming a quasi steady state heat balance.

The flow dependent MCPR limit, MCPR(F), is provided in the CORE OPERATING LIMITS REPORT. The MCPR(F) is independent of the rated flow limit provided in Specification 3.5.K.2 and 3.5.K.3. To verify applicability of this curve to PBAPS, recirculation flow runout events were analyzed with a PBAPS specific model at a typical mid cycle exposure condition. These flow runout events were simulated along the Maximum Extended Load Line Limit rod line to the maximum core flow runout value of 105%. The results of the analyses indicated that application of the MCPR(F) curve will preclude a violation of the MCPR safety limit in the event of a recirculation flow runout. The MCPR(F) curve is cycle independent.

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6.9.1 Routine Reports (cont'd)

c. Annual Safety/Relief Valve Report

Describe all challenges to the primary coolant system safety and relief valves. Challenges are defined as the automatic opening of the primary coolant safety or relief valves in response to high reactor pressure.

d. Monthly Operating Report

Routine reports of operating statistics and shutdown experience and a narrative summary of the operating experience shall be submitted on a monthly basis. Each report shall be submitted no later than the 15th of the month following the calendar month covered by the report.

e. Core Operating Limits Report

(1) Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each Operating Cycle, or prior to any remaining portion of an Operating Cycle, for the following:

a. The APLHGR for Specification 3.5.I,

b. The MCPR for Specification 3.5.K,

c. The core flow and power adjustment factors for Specification 3.5.K and 3.5.I,

d. The LHGR for Specification 3.5.J,

e. The upscale power biased Rod Block Monitor setpoints and corresponding power levels.

(2) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents as amended and approved:

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)

b. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993

c. Philadelphia Electric Company Methodologies as described in:

(1) PECo-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

6.9.1 Routine Reports (cont'd)

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b. The MCPR for Specification 3.5.K,

c. The core flow and power adjustment factors for Specification 3.5.K and 3.5.I,

d. The LHGR for Specification 3.5.J,

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ATTACHMENT 3

General Electric Affidavit

and

General Electric Analysis

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, ROBERT C. MITCHELL, being duly sworn, depose and state as follows:

(1) I am Project Manager, Safety and Communications, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32162P, Rev. 1, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Unit 2 And 3", Class III, dated February 1993.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his

competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in (6) and (7) following. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and

potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations supporting the expanded operating domain of the BWR under normal, transient and accident conditions.

The development and approval of the system, component and thermal hydraulic models and computer codes used to analyze transient and accident events was achieved at a significant cost, on the order of several million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such

information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) SS:
COUNTY OF SANTA CLARA)

Robert C. Mitchell, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 30TH day of MARCH, 1993

Robert C. Mitchell
Robert C. Mitchell
General Electric Company

Subscribed and sworn before me this 30th day of March, 1993

Paula F. Hussey
Notary Public, State of California

OFFICIAL SEAL
PAULA F HUSSEY
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires APR 5, 1994

3/29/93