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Docket No.: 50-333

MAR. 1 2 1976

Power Authority of the State
of New York
ATTN: Mr. George T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

Gentlemen:

The Commission has issued the enclosed Amendment No. 14 to Facility License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment includes changes to the Technical Specifications and is in response to your requests dated July 9, 1975, and supplements thereto dated July 24, 1975, August 1, 1975, September 12 and 22, 1975, October 28, 1975, December 23, 1975, January 6, 8, 14, 16, 23, 26, 27, and 29, 1976, and February 5 and 11, 1976.

The amendment authorizes operation of the FitzPatrick Plant (1) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), (2) with modified operating limits based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's regulations, (3) with a modification to the Low Pressure Coolant Injection System (LPCIS) authorized by Amendment No. 8 to the license, and (4) with plugged bypass flow holes authorized by Amendment No. 9 to the license. Also included in Amendment No. are additional surveillance requirements on the swing buses and associated electrical systems which are to be implemented following the completion of the proposed LPCIS modifications.

Copies of the related Safety Evaluation, Environmental Impact Appraisal, and Federal Register Notice are also enclosed.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures: See next page

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Power Authority of the State
of New York

Enclosures:
1. Amendment No. 14
2. Environmental Impact Appraisal
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures: See next page

RLB
R. Baer
3/10/76

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Power Authority of the State
of New York

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

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York and Niagara Mohawk Power Corporation (the licensees) dated July 9, 1975, as supplemented July 24, 1975, August 1, 1975, September 12 and 22, 1975, October 28, 1975, December 23, 1975, January 6, 8, 14, 16, 23, 26, 27, and 29, 1976, and February 5 and 11, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, Facility License No. DPR-59 is amended by deleting Paragraphs 2.C.(3) and (4) and by changing the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: MAR. 1 2 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 14
FACILITY OPERATING LICENSE NO. DPR-59
DOCKET NO. 50-333

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
vii	vii
5-22	5-20
23	23
30 - 32	30 - 32
35 & 36	35 & 36
39 & 40	39 & 40
47	47
-------	47a
48	48
57 & 58	57 & 58
69 & 70	69 & 70
79	79
93 & 94	93 & 94
101 - 104	101 - 104
113 & 114	113 & 114
115	115 & 115a
123 & 124	123 & 124
129 - 136	129 - 136
145 & 146	145 & 146
215 & 126	215 & 216

Changes on revised pages are indicated by marginal lines.

LIST OF FIGURES

opened to perform necessary operational activities.

2. At least one door in each air-lock is closed and sealed.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

N. Rated Power - Rated power refers to operation at a reactor power of 2,436 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power.

O. Reactor Power Operation - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the Plant subsequent to that refueling.

R. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

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

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

2. The Standby Gas Treatment System is operable.

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

T. Surveillance Frequency - Periodic

1.0 (cont'd)

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ± 25 percent. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)-Ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience *boiling* transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).

2. Peaking Factor - The ratio of the maximum fuel rod surface heat flux in any assembly to the average surface heat flux of the core.

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

V. Electrically Disarmed Control Rod-

To disarm a rod drive electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

1.1 SAFETY LIMITS

1.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specifications

A. Reactor Pressure >*785 psig* and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

2.1 LIMITING SAFETY SYSTEM SETTINGS

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specifications

A. Trip Settings

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

1. Neutron Flux Trip Settings

   a. IRM - The IRM flux scram setting shall be set at ≤ 120/125 of full scale.

1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure ≤ 785 psig)

When the reactor pressure is ≤ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be ≤ 15 percent of rated neutron flux, with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

1.1 (cont'd)

1.1.D. Reactor Water Level (Hot or Cold Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 in. (-146.5 in. indicated level) above the top of the active fuel when it is seated in the core.

2.1 (cont'd)

2.1.A.1.c. APRM Flux Scram Trip Setting (Run Mode)

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

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.60, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \frac{2.60}{\text{MTPF}}$$

where:

MTPF = The value of the existing maximum total peaking factor

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

1.1 (cont'd)

2.1 (cont'd)

2.1.A.1.d. <u>APRM Rod Block Trip Setting</u>

The APRM Rod block trip setting shall be:

$$S_{RB} \leq 0.66\ W + 42\%$$

where :

S_{RB} = *Rod block setting in percent of rated thermal power (2436 MWt)*

W = *Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals (34.2 x 10^6 lb/hr))*

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.60, the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\%)\ \frac{2.60}{MTPF}$$

where:

MTPF = *The value of the existing maximum total peaking factor*

2.1 (cont'd)

2. <u>Reactor Water Low Level Scram Trip Setting (LL1)</u>

Reactor low water level scram setting shall be $\leq$ 177 in. (+12.5 in. indicated level) above the top of the active fuel (TAF) at normal operating conditions.

3. <u>Turbine Stop Valve Closure Scram Trip Setting</u>

Turbine stop valve scram shall be $\leq$10 percent valve closure from full open when above 217 psig turbine first stage pressure.

4. <u>Turbine Control Valve Fast Closure Scram Trip Setting</u>

Turbine control valve fast closure scram on control oil pressure shall be set at 500 <P<850 psig.

5. <u>Main Steam Line Isolation Valve Closure Scram Trip Setting</u>

Main steam line isolation valve closure scram shall be $\leq$10 percent valve closure from full open.

6. <u>Main Steam Line Isolation Valve Closure on Low Pressure</u>

When in the run mode main steam line low pressure initiation of main steam line isolation valve closure shall be $\geq$ 850 psig.

1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.06. MCPR > 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding, perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.06 has sufficient conservation to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > 1.37 more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit 1.06 is derived from a detailed statistical analysis considering all of the

1.1 BASES (Cont'd.)

uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference I. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle. Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.06 would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrance of loss of cladding integrity.

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

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

In addition to the boiling transition limit (MCPR = 1.06) operation is constrained to a maximum LHGR=18.5 Kw/ft. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.60. For the case of the MTPF exceeding 2.60, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by specification 2.1.A.1.C.

B. Core Thermal Power Limit (Reactor Pressure < 785 psig)

At pressures below 785 psig the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28x10^{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 28x10^{3} lbs/hr. Full scale ATLAS test data taken at pressures from 0 psig to 785 psig 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 785 psig is conservative.

1.1 BASES (Cont'd.)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.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. 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.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958.

2.1 BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Fitzpatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2535 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 is the licensed maximum power level of Fitzpatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity, coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. Active coolant flow is equal to 88% of total core flow, The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

2.1 BASES (Cont'd.)

For analyses of the thermal consequences of the transients a MCPR of 1.37 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The abnormal operational transients were analyzed to a power level of 2535 MWt.

ii. The licensed maximum power level is 2436 MWt.

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

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. IRM Flux Scram Trip Setting

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 a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence 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.

2.1 BASES (Cont'd.)

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.06. 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 Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

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

c. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2436 MWt). Because fission chambers provide the basic input signals, the APRM

2.1 BASES (Cont'd.)

system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnromal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin. 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.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c, when the maximum total peaking factor is greater than 2.60.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 when the transient is initiated from MCPR > 1.37.

d. <u>APRM Rod Block Trip Setting</u>

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.06. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established

2.1 BASES (Cont'd)

by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds 2.60, thus preserving the APRM rod block safety margin.

2. Reactor Water Low Level Scram Trip Setting (LLI)

The reactor low water level scram is set at a point which will assure that the water level used in the Bases for the Safety Limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator dise dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar and no more severe than for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section 14.5 of the Final Safety Analysis Report. This scram is by passed when turbine steam flow is below 30 percent of rated, as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

2.1 BASES (Cont'd.)

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

6. Main Steam Line Isolation Valve Closure on Low Pressure

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

C. References

1. Linford, R.B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor, "NEDO-10802, Feb., 1973.

RATED THERMAL POWER = 2436

RATED CORE FLOW = 77.0×10^6

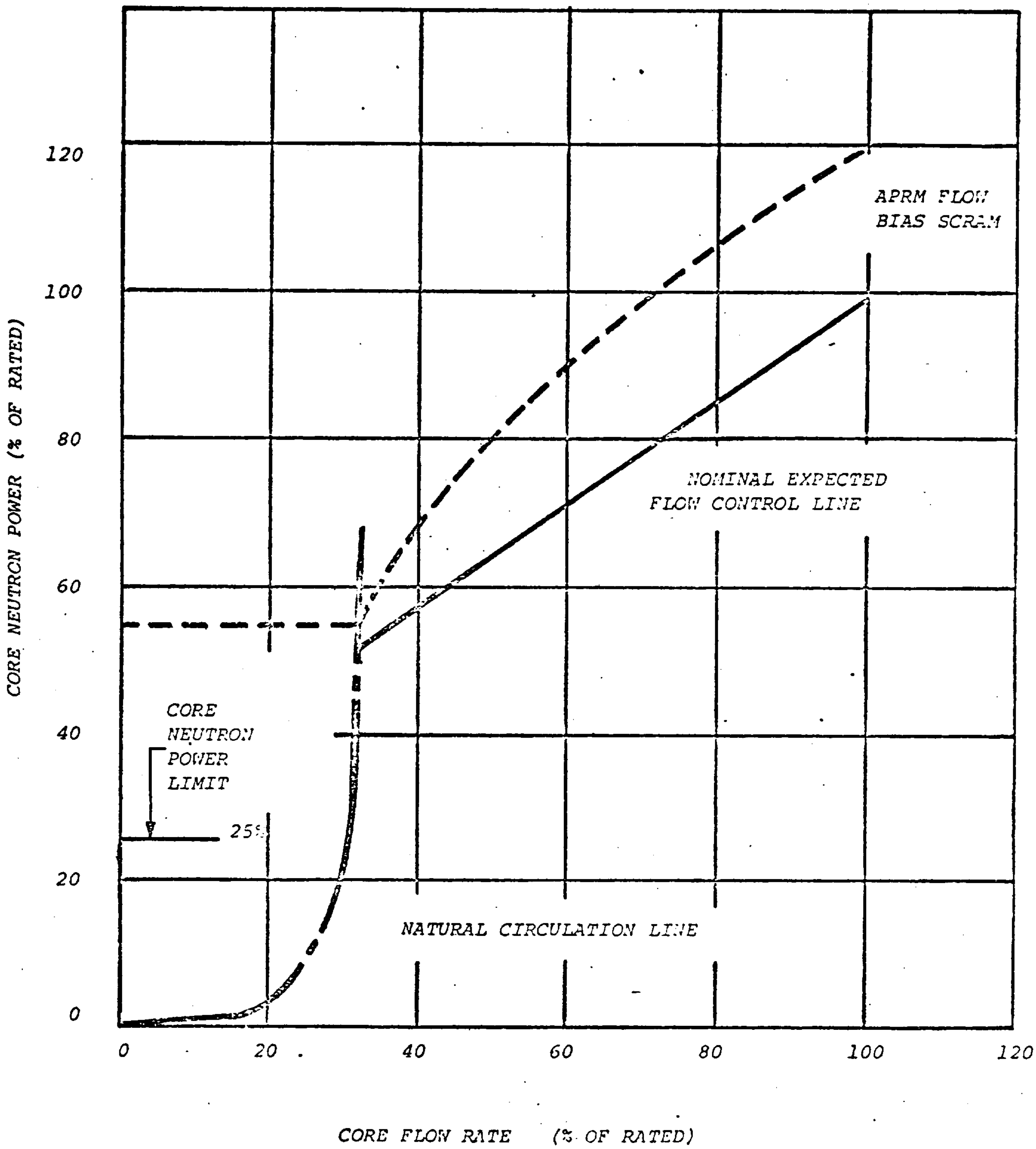


FIGURE 1.1-1 APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. *The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 100 msec.*

B. *Minimum Critical Power Ratio (MCPR)*

MCPR shall be $\geq$ 1.37 at rated power and flow. If at any time during steady state operation it is determined that *the limiting value for MCPR is being exceeded action shall then be initiated within 15 [illegible] to restore operation [illegible] steady state*

4.1 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. *Daily, during reactor power operation, while in the RUN MODE, the peak heat flux and peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specifications 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds 2.6.*

3.1 (cont'd)

reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, the MCPR shall be $\geq$ 1.37 times K_f where K_f is as shown in Figure 3.1.1.

B. The limiting transient which determines the required steady state MCPR limit is the turbine trip without bypass. This transient yields the largest ΔMCPR (0.31) which when added to the Safety Limit MCPR of 1.06 yields the minimum operating limit MCPR of 1.37. The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, the Technical Specifications limit operation of the reactor to the more conservative MCPR of 1.37 for 7 x 7 fuel based on consideration of the limiting transient.

C. *MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.*

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

Amendment No. 14

3.1 BASES

A. The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the Reactor Coolant System.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The Reactor Protection System is of the dual channel type (Reference subsection 7.2 FSAR). The System is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 (1971) for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the average power range monitor (APRM) channels, the intermediate range monitor (IRM) channels, the main steam isolation valve closure and the turbine stop valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal EHC oil pressure of 1,600 psig so that, based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the startup and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. *See 4.1.C Bases*

4.1 BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in Reference (6). This concept was specifically adapted to the 1 out of 2X2 logic of the Reactor Protection System. The analysis shows that the sensors are primarily responsible for the reliability of the Reactor Protection System. This analysis makes use of unsafe failure rate experience at conventional and nuclear power plants in a reliability model for the system. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, and faulted cables, which result in upscale or downscale readings on the reactor instrumentation are safe and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1-1 and 4.1-2 are divided into three groups for functional testing. These are:

A. On-off sensors that provide a scram trip function.

B. Analog devices coupled with bi-stable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up Group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval is planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration.

is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

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

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

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

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 percent/month; i.e., in the period of a month a maximum drift of 0.4 percent could occur, thus providing for adequate margin.

For the APRM System, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every 7 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 peak heat flux is checked once/day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the peak heat flux is adequate.

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)

It is highly improbable that in actual operation with TPF at 2.6 that MCPR will be as low as 1.2. Usually with peaking factors of this magnitude the peak occurs low in the core in a low quality region where the initial heat flux is very high. The MCPR design power shape (TPF = 2.43) assumes that the peak occurs higher in the core and represents the worst combination of individual peaking factor magnitude and shape, from a MCPR consideration that can be expected to occur in the core. Therefore, with TPF < 2.43 there are no technical specification requirements for calculating MCPR. With TPF > 2.43 the daily requirement for calculating MCPR 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 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.

TABLE 4.1-2 (CONT'D)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUN CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

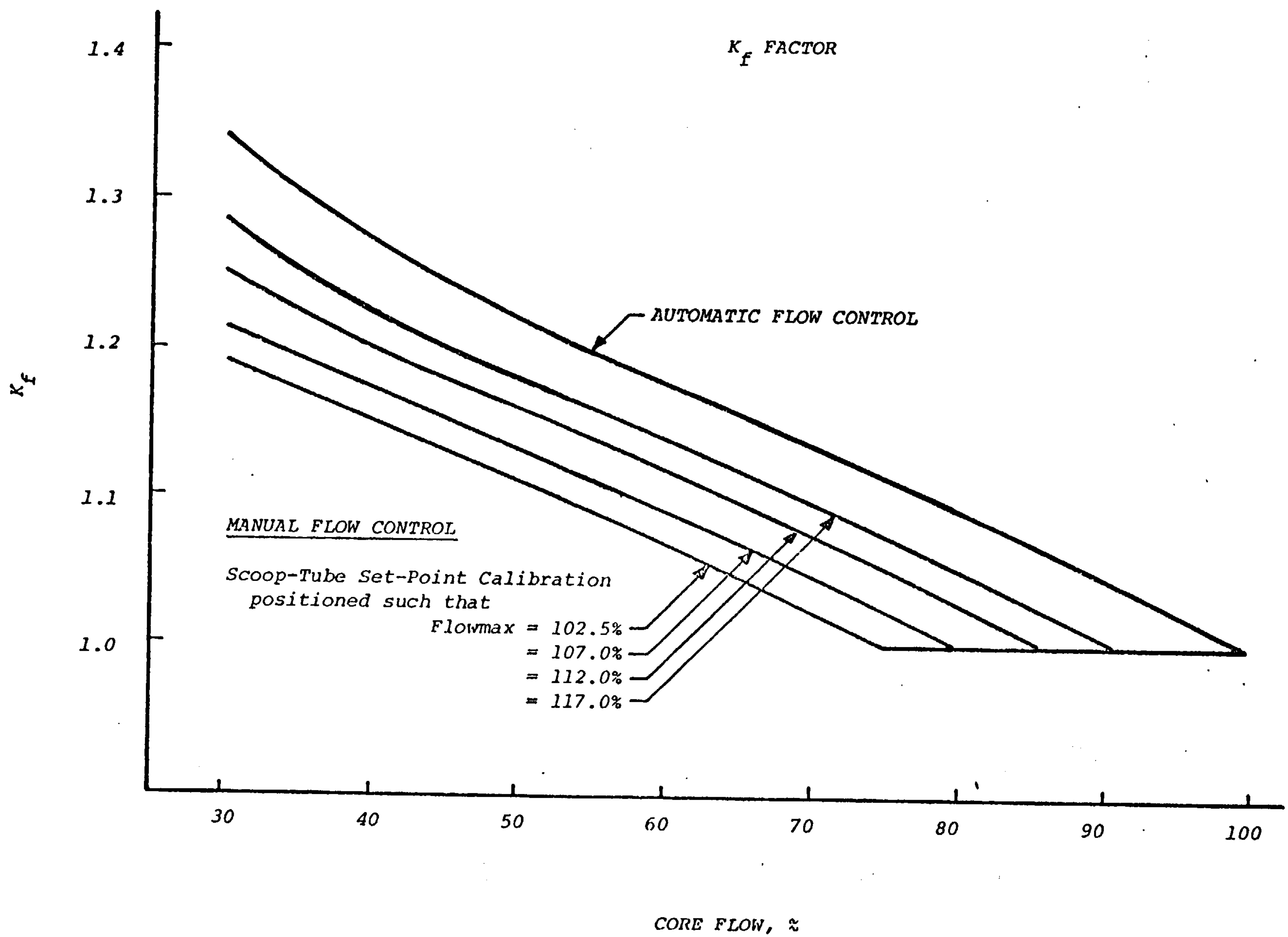
2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Response time is not a part of the routine instrument channel test but will be checked once per operating cycle.

5. Actuation of these switches by normal means will be performed during the refueling outages.

FIGURE 3.1.1



K_f FACTOR
1.4
1.3
1.2
1.1
1.0
K_f
AUTOMATIC FLOW CONTROL
MANUAL FLOW CONTROL
Scoop-Tube Set-Point Calibration
positioned such that
Flowmax = 102.5%
= 107.0%
= 112.0%
= 117.0%
30
40
50
60
70
80
90
100
CORE FLOW, %

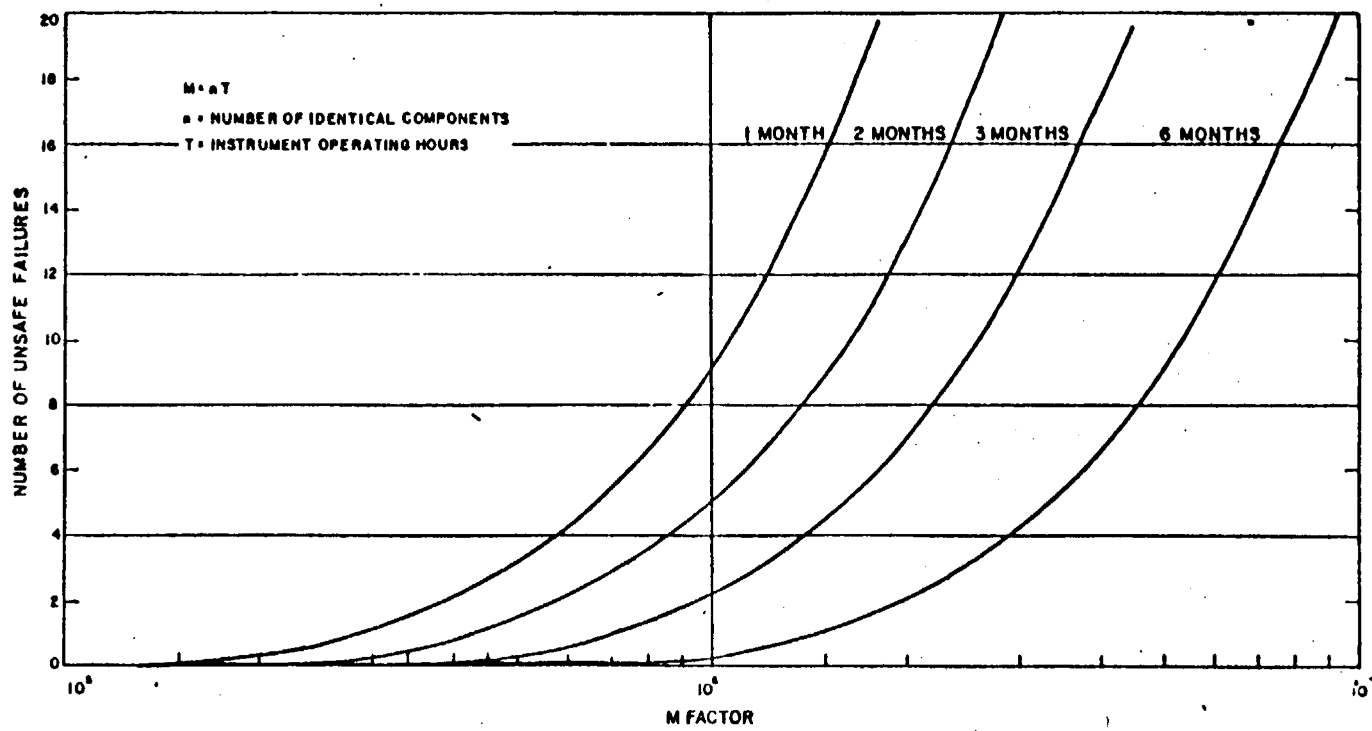


FIG. 4.1-1
GRAPHICAL AID IN THE SELECTION OF AN ADEQUATE INTERVAL BETWEEN TESTS

3.2 BASES (cont'd)

backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 6 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10CFR100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drops below 850 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of ≤ 300 percent of design flow for high flow and 40° F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of ≤ 300 percent for high flow and 40° F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high flow temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not de-

3.2 BASES (cont'd)

crease to 1.06. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only three percent of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in *MCPR* especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that *MCPR* is maintained >1.06.

The RBM rod block function provides local protection of the core: i.e., the prevention of *boiling transition* in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection.

The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the Mode Switch is in the Refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Remarks
2	Recirculation Pump A d/p	≤2.0 psid	4 Inst. Channels	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop.
2	Recirculation Pump B d/p	≤2.0 psid	4 Inst. Channels	
2	Recirculation Riser d/p A>B	0.5 < P < 1.5 psid	4 Inst. Channels	
1	Core Spray Sparger to Reactor Pressure Vessel d/p	≤5 psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break.
2	Condensate Storage Tank Low Level	≥ 59.5 in. above tank bottom (=15,600 gal avail)	2 Inst. Channels	Provides interlock to HPCI suction valves.
2	Supression Chamber High Level	≤6 in. above normal level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
1	RCIC Turbine Steam Line High Flow	≤282 in. H_2O psid	2 Inst. Channels	Close Isolation Valves in RCIC Subsystem
6	RCIC Steam Line Area Temperature	≤ 40° F Above max. ambient	12 Inst. Channels	Close isolation valves in RCIC Subsystem
2	RCIC Steam Line Low Pressure	100 > P > 50 psig	4 Inst. Channels	Close isolation valves in RCIC Subsystem
1	HPCI Turbine Steam Line High Flow	≤230 in. H_2O psid (3)	2 Inst. Channels	Close Isolation Valves in HPCI Subsystem
1	RCIC Turbine High Exhaust Pressure	≤25 psig	2 Inst. Channels	Trips RCIC Turbine
1	HPCI Turbine High Exhaust Pressure	≤ 150 psig	2 Inst. Channels	Trips HPCI Turbine
1	LPCI Cross-Connect Position	NA	1 Inst.	Initiates annunciation when valve is not closed

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Remarks
4 (3)	HPCI Steam Line Low Pressure	100 > P > 50 psig (3)	2 Inst. Channels	Close Isolation Valves in HPCI Subsystem
4	HPCI Steam Line Area Temperature	≤ 40°F. (3) above max. ambient	5 Inst. Channels	Close Isolation Valves in HPCI Subsystem
1	HPCI Low Pump Suction Pressure	≤ 15 in. Hg vac	1 Inst. Channel	Trips HPCI Turbine
1	RCIC Low Pump Suction Pressure	≤ 15 in. Hg vac	1 Inst. Channel	Trips RCIC Turbine
1 per 4KV bus	4 KV Emergency Bus Undervoltage Relay	85 secondary volts ± 5%.2.50 sec ± 2% time delay		1. Trips all loaded breakers 2. Dead bus start of diesel
1 per 4KV bus	4KV Emergency Bus Under Voltage Timer	2.0 sec x 0.1 sec	2 Inst.	1. Initiates sequential starting of vital loads 2. Initiates diesel breaker close permissive 3. Initiates bus tie breaker trip
2	Reactor Low Pressure	285 to 335 psig.	4 Inst. Channels	Permissive for closing recirculation pump discharge valve.

TABLE 4.2-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level	(1)	Once/3 months	Once/day
2) Drywell Pressure	(1)	Once/3 months	None
3) Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	Once/operating cycle	None
7) Recirculation System d/p	(1)	Once/3 months	Once/day
8) Core Spray Sparger d/p	(1)	Once/6 months	Once/day
9) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
10) Steam Line High Temp. (HPCI & RCIC)	(1)	Once/operating cycle	Once/day
11) Safeguards Area High Temp.	(1)	Once/operating cycle	None
12) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
13) HPCI Suction Source Levels	(1)	Once/3 months	None
14) 4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/5 years	None
15) HPCI and RCIC Exhaust Pressure High	(1)	Once/3 months	None
16) HPCI and RCIC Low Pump Suction Pressure	(1)	Once/3 months	None

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

17) LPCI/Cross Connect Valve Position	Once/refueling outage	N/A	N/A

shall verify that the operator at the reactor console is following the control program.

c. During the shutdown procedure no rod movement is permitted following the testing performed between 35% and 20% power level and the automatic reinstatement of the RSCS restraints at the preset power level. Alignment of rod groups shall be accomplished prior to performing the tests.

d. Control rod withdrawal sequence shall be established so that the maximum reactivity that could be added by drop of any increment of any one control blade, would not make the core more than 0.0125Δk supercritical.

e. If Specifications 3.3.B.3a through c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20 percent rated power, it shall be brought to a shutdown condition immediately.

(1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.

(2) The RWM computer on line diagnostic test shall be successfully performed.

(3) Prior to startup, proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

(4) The rod block function of the RWM shall be verified by withdrawing the first rod during startup only as an out-of-sequence control rod no more than to the block point.

c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.

3.3 (cont'd)

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

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RBM channels shall be operable, or

   b. Control rod withdrawal shall be blocked, or

   c. The operating power level shall be limited so that MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 (cont'd)

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

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

3.3 and 4.3 BASES (cont'd)

NEDO-10527. The peak fuel energy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage are assumed to occur.

The RSCS will prevent the operator from inadvertently selecting and moving a high worth rod in the startup and low power ranges. Above 20 percent power, the results of the rod drop accident with the worst single operator error are less than 170 cal/gm. Therefore, this system, in addition to normal operating procedures and the RWM, prevents the postulated rod drop accident from exceeding 280 cal/gm over the entire range of plant operating conditions.

The effectiveness of RSCS in limiting peak fuel enthalpy has been positively evaluated only up through the first refueling outage. Thus a complete RSCS re-evaluation will be required subsequent to the first refueling outage.

In the event that the RWM is out of service, when required, a second licensed operator can manually fulfill the control rod pattern conformance functions of the RWM.

The RSCS can be functionally tested prior to control rod withdrawal before the reactor is at 20 percent power and prior to reactor startup.

By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full out all A_{12} rods, selecting A_{34}, and attempting to withdraw, by one notch, a rod or all rods in group B or C, and A_{34} group is determined exclusive. The same procedure can be repeated for the B and C groups.

4. The Source Range Monitor (SRM) System performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per sec assures that

any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transient cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR 1.37 or LHGR = 18.5 kW/ft). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Analyst to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the Plant Superintendent.

C. Scram Insertion Times

The Control Rod System is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.06. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure 3.6-14) with the average response of all the drives as given in the above Specification, provide the required protection, and *MCPR* remains greater than 1.06.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on JAFNPP.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 390 msec are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 msec. Approximately 70 msec after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 msec later, control rod motion begins. The 200 msec are included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times generated at each refueling outage and during operation when compared to scram times generated during pre-operational tests demonstrate that the control rod drive scram function has not deteriorated. In addition, each instant when control rods are scram timed during operation or reactor trips, individual evaluations shall be performed to insure that control rod scram times have not deteriorated.

D. Reactivity Anomalies

During each fuel cycle, excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of

3.3 and 4.3 BASES (cont'd)

the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δ k. Deviations in core reactivity greater than 1% Δ k are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3.5 (cont'd)

2. From and after the date that one of the Core Spray Systems is made or found inoperable for

4.5 (cont'd)

b. Flow Rate Test - Core spray pumps shall deliver at least 4,625 gpm against a system head corresponding to a total pump developed head of ≥113 psig	Once/3 months
c. Pump Operability	Once/month
d. Motor Operated Valve	Once/month
e. Core Spray Header Δ p Instrumentation	
Check	Once/day
Calibrate	Once/3 months
Test	Once/3 months
f. Logic System Functional Test	Once/each operating cycle

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, the LPCI System, and the emergency diesel generators shall be

3.5 (cont'd)

any reason, continued reactor operation is permissible during the succeeding 7 days unless the system is made operable earlier, provided that during the 7 days all active components of the other Core Spray System and the LPCI System and the emergency diesel generators shall be operable.

3. The LPCI mode of the RHR System shall be operable whenever irradiated fuel is in the reactor and prior to reactor startup from a cold condition, except as specified below.

   a. From the time that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the pump is made operable earlier provided that during such 7 days the remaining active components of the LPCI, containment spray mode, all active components of both Core Spray Systems, and the emergency diesel generators are operable.

4.5 (cont'd)

demonstrated to be operable immediately. The remaining Core Spray System shall be demonstrated to be operable daily thereafter.

3. LPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, and f except that three RHR pumps shall deliver at least 23,100 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

   a. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI, containment spray subsystem, both Core Spray Systems, and the emergency diesel generators required for operation shall be demonstrated to be operable immediately, and the remaining RHR pumps shall be demonstrated to be operable daily thereafter.

3.5 (cont'd)

b. From the time that the LPCI mode is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the LPCI mode is made operable earlier provided that during these 7 days all active components of both Core Spray Systems, the containment spray subsystem (including two RHR pumps) and the emergency diesel generators shall be operable.

c. The motor operator for the RHR cross-tie valve shall be maintained disconnected from its power source. It shall be maintained chain-locked in the closed position. The manually operated gate valve in the cross-tie line, in series with the motor operated valve, shall be maintained locked in the closed position.

4. The reactor shall not be started up with the RHR System supplying cooling to the fuel pool.

4.5 (cont'd)

b. When it is determined that the LPCI mode is inoperable, both Core Spray Systems, the containment spray subsystem, and the emergency diesel generators shall be demonstrated to be operable immediately and daily thereafter.

c. The power source disconnect and chain lock to motor operated RHR cross-tie valve, and lock on manually operated gate valve shall be inspected once each operating cycle to verify that both valves are closed and locked.

5. If the requirements of 3.5.A cannot be met, the reactor shall be placed in the cold condition within 24 hr.

B. <u>Containment Cooling Subsystem Mode (of the RHR System)</u>

1. Both subsystems of the containment cooling mode, each including two RHR, one ESW pump and two RHRSW pumps shall be operable whenever there is irradiated fuel in the reactor

B. <u>Containment Cooling Subsystem Mode (of the RHR System)</u>

1. Subsystems of the containment cooling mode are tested in conjunction with the tests performed on the LPCI System and given in 4.5.A.1.a, b, c, and d. Residual heat removal

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.5.1 and 3.5.2. If at any time it is determined that the limiting value for APLHGR is being exceeded action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high poing prior to the return of the system to service.

3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The pressure switches which monitor the Core Spray and LPCI discharge lines to ensure that they are full shall be functionally tested every month and calibrated every three months.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

3.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \ (1 - \{(\Delta P/P)_{max} \ (L/LT)\})$$

$LHGR_d$ = Design LHGR = 18.5 KW/ft.

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = 0.026

LT = Total core length = 12 feet

L = Axial position above bottom of core

If at any time it is determined that the limiting value for LHGR is being exceeded action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

I. Linear Heat Generation Rate (LHGR)

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

vessel head off the LPCI and Core Spray Systems will perform their designed safety function without the help of the ADS.

E. Reactor Core Isolation Cooling (RCIC) System

The RCIC is designed to provide makeup to the Reactor Coolant System as a planned operation for periods when the normal heat sink is unavailable. The RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 7 days is specified. Immediate and daily demonstrations of HPCI operability during RCIC outage is considered adequate based on judgement and practicality.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the RCIC System is not required, (reactor coolant temperature ≤212°F and coolant pressure ≤150 psig). If the plant parameters are below the point where the RCIC System is required, physics testing and operator training will not place the plant in an unsafe condition.

F. Minimum Emergency Core and Containment Cooling System Availability

The purpose of Specification 4.5.D is to assure a minimum of emergency core cooling equipment is available at all times. If, for example, one core spray were out of service and the emergency bus which powered the opposite core spray were out of service, only two RHR pumps would be available. Likewise, if two RHR pumps were out of service and two RHR on the opposite side were also out of service, no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the Reactor Coolant System which could lead to draining the vessel. This work would include work on certain control rod drive components and Reactor Recirculation System. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other

3.5 BASES (Cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform its design function.

H. Average Planar Linear Heat Generation Rate (APLHGR)

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 50 Appendix K.

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 local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm 20^{\circ}$ 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 within the 10 CFR 50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.5.1 and 3.5.2.

I. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

K. 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 (AEC Regulatory Staff).

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

4.5 BASES

The testing interval for the Core and Containment Cooling Systems is based on a quantitative reliability analysis, industry practice, judgement, and practicality. The Emergency Core Cooling Systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems will be automatically actuated during a refueling outage. In the case of the Core Spray System, condensate storage tank water will be pumped to the vessel to verify the operability of the core spray header. To increase the availability of the individual components of the Core and Containment Cooling Systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise, the pumps and motor-operated valves are also tested each month to assure their operability. The combination automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC Systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided in the Core Spray System and LPCI System to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the interval between the monthly checks the status of the discharge piping is monitored on a continuous basis.

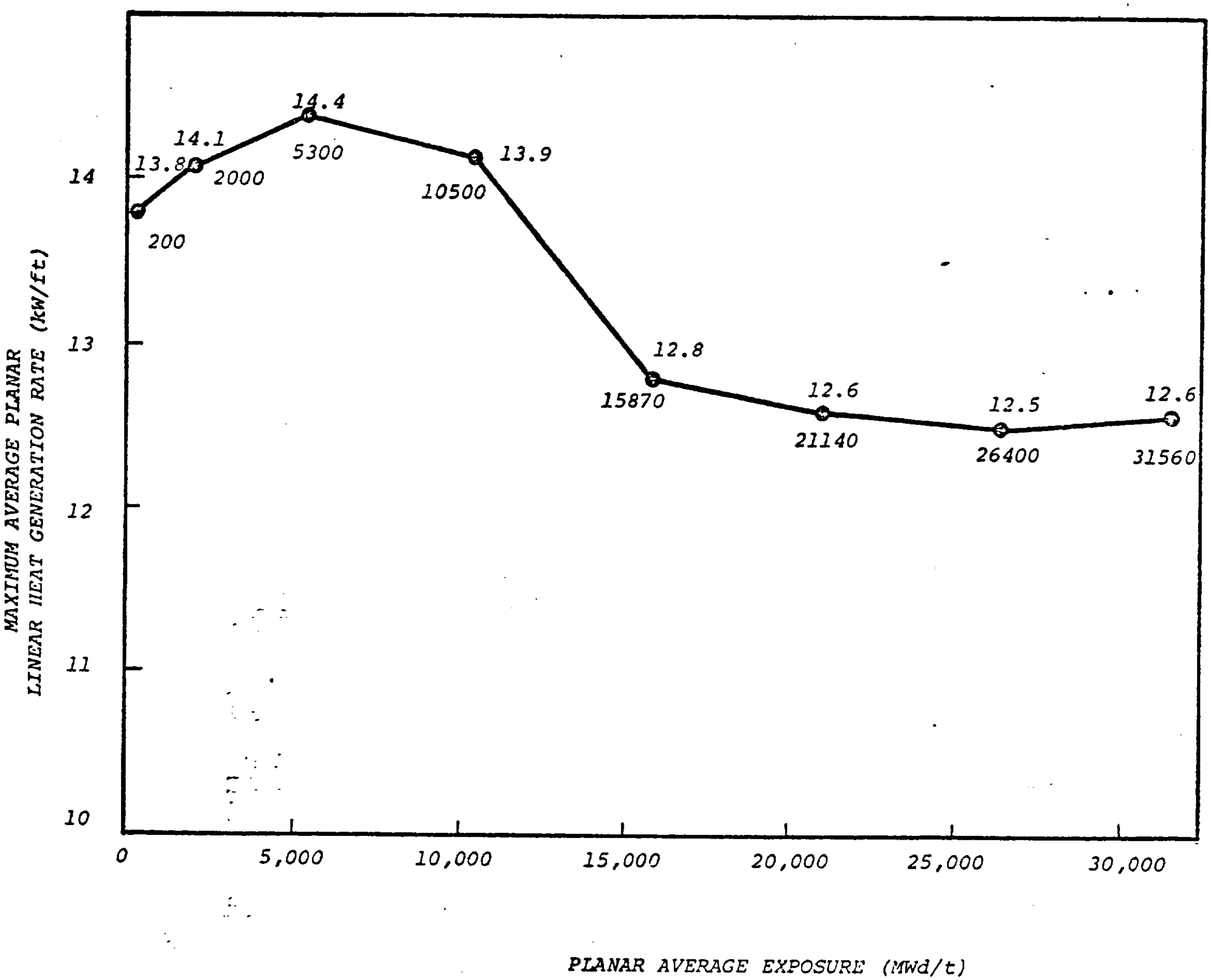


FIGURE 3.5-1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

INITIAL CORE - TYPE 1 & 3

Amendment No. 14

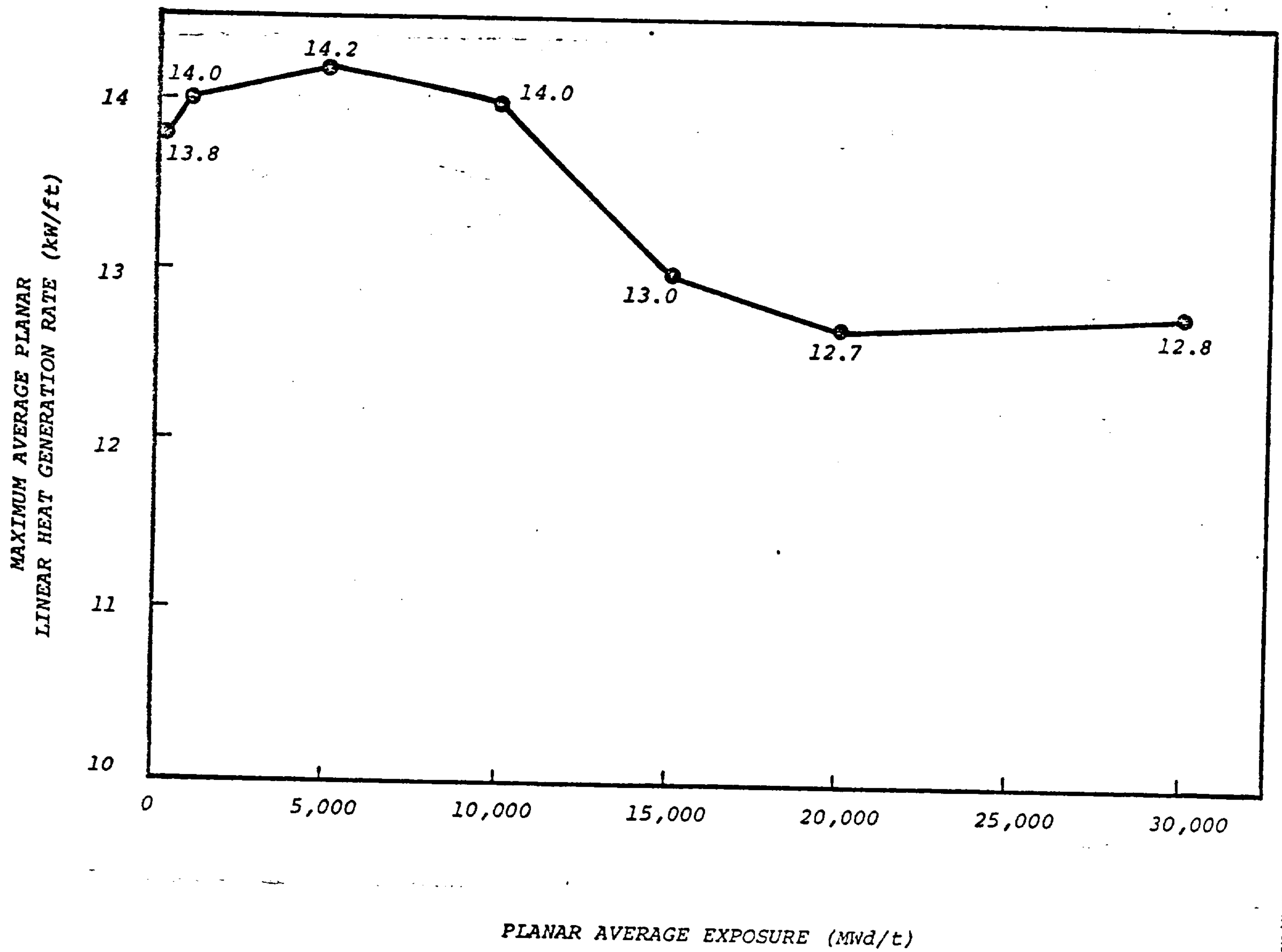


FIGURE 3.5-2 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

INITIAL CORE - TYPE 2

Amendment No. 14

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

A. Thermal Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a 1 hr period.

2. The reactor recirculation pumps shall not be operated unless the coolant temperatures between the upper and lower regions of the vessel are within 145°F.

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15 min intervals:

   a. Reactor coolant - upper vessel region

   b. Reactor coolant - lower vessel region

   c. Recirculation loops A and B

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 min intervals until three consecutive readings are within 5 degrees of each other.

1. The two recirculation loops have a flow imbalance of 15 percent or more when the pumps are operated at the same speed.

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10 percent.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the average of all jet pump differential pressures by more than 10 percent.

H. Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122 percent the speed of the slower pump when core power is 80 percent or more of rated power, or 135 percent the speed of the slower pump when core power is below 80 percent of rated power.

2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

3. The reactor shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.

H. Jet Pump Flow Mismatch

1. Recirculation pump speeds shall be checked and logged at least once/day.

3.6 and 4.6 BASES

A. Thermal Limitations

The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F/hr averaged over a period of 1 hr. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100° F/hr rate is limited provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hr applied continuously over a temperature range of 100°F to 546°F.

Calculated stresses were within ASME Boiler and 1965 ASME Boiler and Pressure Vessel Code, Section III, with 1966 addenda stress intensity and fatigue limits. The normal heating and cooling rate of 100°F/hr was also evaluated to assure protection against brittle fracture of the vessel shell remote from discontinuities. The rate meets the requirements of Appendix G to the Summer 1972 Edition of 1971 ASME III, throughout plant life, and is therefore satisfactory.

The limiting coolant temperature differential between the upper and lower regions of the reactor vessel, prior to recirculation pump operation, assures that the vessel bottom head region will not be warmed at an excessive rate due to rapid sweep-out of cold coolant in the vessel lower head region by recirculation pump operation (cold coolant can accumulate as a result of control drive inleakage and/or low recirculation flow rate during startup or hot standby). The limit on idle recirculation loop startup avoids high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

B. Pressurization Temperature

The Reactor Coolant System is a primary barrier against the release

3.9 LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEMS

Applicability:

Applies to the auxiliary electrical systems.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Normal and Reserve A-C Power Systems

The reactor shall not be made critical unless all of the following requirements are satisfied:

1. Power is available to the emergency buses from the following power sources:

   a. the two 115 kv lines and reserve station service transformers

   b. the two Emergency Diesel Generator Systems.

2. a. 4,160 v buses 10,500 and 10,600 are energized.

   b. 600 v buses 11,500, 12,500, 11,600 and 12,600 are energized.

4.9 SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEMS

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Swing Buses

a. Every two months the swing buses supplying power to the Low Pressure Coolant Injection System (LPCIS) valves shall be tested to assure that the transfer circuits operate as designed.

B. Emergency A-C Power System

Except when the reactor is in the cold shutdown or refueling modes with the head off, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1, 3.9.B.2, and 3.9.B.3.

1. From and after the time that incoming power is available from only one line or through only one reserve station service transformer, reactor operation is permissible for a period not to exceed seven days total for degradation of any combination of lines and transformers for any calendar month, provided that both Emergency Diesel Generator Systems are operable. At the end of the accumulated 7 days the reactor shall be placed in a cold condition within 24 hr.

2. From and after the time that incoming power is not available from any line or through neither reserve station service transformer, continued reactor operation is permissible for a period not to exceed 7 days, provided that both redundant

B. Emergency A-C Power System

1. Once each month, each pair of diesel generators which forms a redundant Emergency Diesel Generator System shall be manually initiated to demonstrate its ability to start, accelerate, and force parallel; after connection to the bus, the paralleled pair will be loaded to 5,200 KW, this load will be maintained until both generators are at steady state temperature conditions. During this period the generators' load sharing capability will be checked.

2. Once per month the diesel starting air compressors shall be checked for proper operation and their ability to recharge air receivers.

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated July 9 and December 23, 1975, the Power Authority of the State of New York, as owner, and Niagara Mohawk Power Corporation, as operator (the licensees), proposed changes to the Technical Specifications in Appendix A of Facility License No. DPR-59.

The proposed change would incorporate the "Acceptance Criteria for the Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (ECCS) as specified in Section 50.46 of Part 50 CFR into the operating license for the FitzPatrick Nuclear Power Plant. The licensees are presently authorized to operate at power levels up to 2436 megawatts thermal. The proposed action would result in a decrease in the power level amounting to less than 10 percent for no longer than 12 months. We have independently reviewed the expected environmental impact of the proposed action.

2. Environmental Impacts of Proposed Action

In the absence of any significant change in power levels, there would be no change in cooling water requirements. Further, there would be no change in radioactive effluents or thermal effluents from normal operation or post accident conditions. The restrictions on heat generation rates will require careful control of fuel operation history; however, there should be no reduction in total burnup resulting from the revised ECCS evaluation methods. It is not anticipated that the issuance of this change to the Appendix A Technical Specifications would affect the cost-benefit balance nor would it require changes in the Environmental Technical Specifications in Appendix B of the license.

No environmental impacts are expected other than those described in the Commission's Final Environmental Statement for the James A. FitzPatrick Nuclear Power Plant issued March 1973. The Commission's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected release rates from the total quantity of nuclear fuel within the reactor. The proposed action would not affect the total quantity of fuel used at FitzPatrick. No increases in radiation doses to humans or other biota are expected.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than those impacts described in the Final Environmental Statement, issued March 1973. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Date: MAR. 1 2 1976

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

INTRODUCTION

The FitzPatrick plant shut down on January 18, 1976, in order to plug the bypass flow holes in the lower core support plate to alleviate the problem of vibrating instrument tubes in the core which could in turn cause excessive wear on channel box corners. In addition, the licensee desired to modify the low pressure coolant injection system. We had previously approved these two modifications but not their use in plant operation.

The Power Authority of the State of New York has proposed to operate the FitzPatrick plant under the following conditions:

(1) with plugged bypass flow holes in the lower core support plate as requested in its submittal dated January 26, 1976;

(2) with limits based on the General Electric Thermal Analysis Basis (GETAB) as requested in its submittal dated February 11, 1976, and supplement dated February 19, 1976;

(3) with modified operating limits based on an acceptable evaluation model that conforms with Section 50.46 of 10 CFR Part 50 as requested in its submittal dated December 23, 1975, and supplements dated January 16, 1976 (Proprietary), January 23 and 29, 1976 and February 19, 1976; and

(4) with a modification to the low pressure coolant injection system (LPCIS) as requested in its submittal dated July 24, 1975, and supplements dated January 6, 8, 14, and 23, 1976.

EVALUATION

The proposed operation with plugs causes a slight increase in the core average void fraction; this effect is discussed under Nuclear Design. In the section of this evaluation entitled Mechanical Design the beneficial effect of the plugs in reducing channel box corner wear is described. The licensee submitted the analysis supporting a proposed General Electric Thermal Analysis Basis (GETAB) with GETAB based Technical Specifications and the loss-of-coolant accident analysis incorporating the effects of the plugs and the use of a modified low pressure coolant injection system. The GETAB analysis describes the safety limit and operating limit minimum critical power ratios (MCPR). The loss-of-coolant analysis is in conformance with Appendix K of 10 CFR Part 50. Both GETAB and LOCA analyses are based on the initial core loading of the FitzPatrick reactor with GE 7 x 7 fuel.

Nuclear Design

The primary nuclear effect caused by plugging the bypass flow holes is an increased bypass void fraction and a reduction in the average in-channel void fraction. The in- and out-of-channel void fraction changes give a net increase in the core average void fraction.

At steady state conditions, the increased bypass void fraction results in a small reduction in the maximum local peaking factor within a fuel bundle and an increase in the local bundle power calculational uncertainty. Another consequence of the reduced bypass flow is a small reduction in the infinite multiplication factor of uncontrolled fuel.

The presence of voids in the bypass region affects the relationship between the traversing incore probe (TIP) signal and the local bundle power. The TIP signal is reduced by the presence of voids and could lead to an underprediction of the peak heat flux. The relationship of the power in the four bundles surrounding a TIP instrument tube and the TIP signal as a function of bypass voids was determined by the General Electric Company (GE) by performing three group, two-dimensional diffusion theory calculations. A correction factor was developed and algorithms for computing the bypass void fraction and for making appropriate corrections in the local bundle power have been incorporated in the process computer.

The uncertainty in the local bundle power caused by bypass voids is taken into account in determining the minimum critical power ratio (MCPR) safety limit. The TIP uncertainty introduced by the bypass voids is zero in the bottom half of the core and increases from 3.98% at the core mid-place to 5.36% at the core exit.

After the bypass flow holes were plugged, the fuel was placed in its original core location. The following observations can be made:

(1) the control rod worths are not significantly changed and, consequently, the previous results of the control rod drop analysis remain valid,

(2) the shutdown margin will remain the same as previously analyzed,

(3) the standby liquid control system reactivity insertion rate and magnitude will not be affected.

We conclude that the analysis of the nuclear performance of the plant with plugged bypass holes is acceptable.

Mechanical Design

We previously issued a Safety Evaluation (14) for the installation of bypass flow hole plugs, but not for operation of the reactor with these plugs. Herein we address operation with plugged bypass flow holes.

The only mechanical design change in the reactor is the use of plugs to fill the bypass flow holes (13). The plug consists of two stainless steel parts (body and shaft) which are connected by an Inconel spring. The shoulder of the body rests on the top of the core plate along the rim of a one-inch bypass hole and is pressed down by the spring. An equal and opposite force is applied on the shaft. A stainless steel latch is connected to the bottom of the shaft by means of a pin. This latch is free to rotate about the pin and latches the shaft to the core plate. The spring exerts a minimum of 35 pounds on the body and latch and a maximum of 46 pounds (with the worst tolerance combination).

Removal of a plug can be accomplished by applying about 500 pounds of force and deforming the latch plastically. More than 10 plugs were removed in tests performed at the GE test facility with consistent latch deformations without damaging other parts.

Plugs identical to those installed in FitzPatrick have been installed in the Vermont Yankee, Duane Arnold and Pilgrim reactors. The plugs installed in Vermont Yankee were removed during a refueling operation after 10 months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be on the order of 20 to 32 psi. These loads together with the spring preload will produce yielding of the latch in bending but will be significantly below about 500 pounds of force necessary for removing the plug. The 1973 GE full scale flow mockup test shows that, with up to 40 psi differential pressure, there is negligible leakage flow through the plugged holes. No plug vibration was observed during the test and no apparent deformation on the latch was evident after the test. No fatigue and plastic strain ratcheting is expected since the plant power cycle during the anticipated service period will be minimal.

Stainless steel and Inconel are compatible with other reactor internals and are not expected to introduce any unusual oxidation and stress corrosion problems. The flux level at the core plate elevation is estimated to be quite low and an insignificant reduction in ductility due to irradiation is anticipated. GE has performed creep tests with both Inconel springs and stainless steel latches and found that stress relaxation or creep deformation were insignificant. The tests were performed at 550°F.

The licensees presented to the NRC staff a summary of channel inspections on BWR-2's and BWR-3's. These older plants have instrument tubes similar to FitzPatrick but no bypass flow holes in the core support plate. The bypass flow for these enters through clearances in the assembly and fittings, which is similar to the FitzPatrick configuration with plugged bypass holes. One hundred sixty-four channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in seven plants. No significant channel wear was observed at the corners adjacent to the instrument tubes.

General Electric has a design criteria for channel box wastage of 0.010 inches for the lower 80 inches of the channel and 0.020 inches for the remaining length. All of the channels (new and old) in the core meet this requirement. Channels with observed acceptable wear on the corner were not reinserted in the core next to an in-core instrument where additional wear could occur during subsequent reactor operation.

Based on a review of the design, the test rig, the installation methods and primarily the previously successful operating experience at Vermont Yankee and Pilgrim, we conclude that the plugs will not fail so as to result in loose parts in the core or result in unplugging of the bypass flow holes. Also, we conclude that the installed plugs will substantially reduce the instrument tube vibration, due to flow through the bypass holes, sufficient to preclude any unacceptable wear for at least the remainder of the present fuel cycle.

General Electric Thermal Analysis Basis (GETAB)

To apply GETAB to the Technical Specifications involves 1) establishing the fuel damage safety limit, 2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and 3) establishing limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied. We have evaluated and report herein the FitzPatrick Nuclear Plant developed thermal margins based on the NEDO-10958 report(1) and plant specific input information provided by the licensee.

Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR for the 7 x 7 fuel is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are expected to avoid boiling transition. The uncertainties in the core and system operating parameters and the GEXL correlation, Table 1-1 of the licensee's submittal, are treated in reference 2 for the unplugged core. Table 5-1 of reference 3 provides the same information for the plugged FitzPatrick core. Both of these references combine the uncertainties in the operating parameters with the relative bundle power distribution in the core to form the basis for the GETAB statistical determination of the safety limit Minimum Critical Power Ratio (MCPR). The tabulated lists of uncertainties for FitzPatrick Nuclear plant are the same or conservative with respect to those reported in NEDO-10958(1) and NEDO-20340(4) which are acceptable. Table 5-1 of the later submittal (reference 3) reflects acceptable inclusion in the statistical analysis of effects due to plugging of the core support plate bypass holes (i.e., inclusion of TIP reading uncertainty due to bypass region voids).

The reactor core selected for the generic GETAB statistical analyses that incorporate the operating parameters, fuel design (R factor*), and GEXL correlation uncertainties is a typical 251/764 core. This selected core is under the same reactor class as is the FitzPatrick core but the selected core is larger. Thus, the GETAB analysis results provide a fuel cladding integrity safety limit MCPR of 1.06 which is conservatively applied to the FitzPatrick reactor. Comparison of the licensee submitted bundle power distributions[2] used for the GETAB application and that for the actual operation of the FitzPatrick reactor illustrate use of more high power bundles in the GETAB analysis which result in a conservative value of 99.9% statistical limit MCPR.

We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable to prevent fuel damage for the FitzPatrick current fuel cycle.

Operating Limit MCPR

Various transient events will reduce the MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in minimum critical power ratio (MCPR). The licensee has submitted the results of the transient analyses which contribute a significant decrease in MCPR.[2,3,5] Types of transients evaluated were loss of flow, pressure and power increase, and coolant temperature decrease. The most limiting transients in the stated categories were 2-pump trip, load rejection without bypass, and loss of feedwater heating. Of these three, the most limiting transient was load rejection without bypass resulting in a ΔMCPR of 0.31 once proper inclusion of effects due to "core plugging" have been incorporated (reference 3). Addition of this ΔMCPR to the safety limit MCPR gives the minimum operating limit MCPR required to avoid violation of the safety limit, should this limiting transient occur.

The transient analyses were evaluated with the end-of-cycle scram reactivity insertion rates that include an acceptable design conservatism factor (see figure 7-1, reference 3). The licensee's submitted initial condition parameters [3] used for the worst operational transient analysis are acceptable. The initial MCPR assumed in the transient analyses was equal to or greater than the established operating limit MCPR of 1.37.

*The R factor is a parameter which characterizes the local peaking pattern with respect to the most limiting rod.

Conservatism was applied in the determination of the required operating limit MCPR because the axial and local peaking were assumed to take place at the beginning of the fuel cycle and the peak of the axial power shape was assumed to occur in the mid-plane (node 12; APF of 1.40). This is the worst consistent set of parameters that is supported by a GE study(1) which has shown the required operating MCPR to be a function of the location of axial peak. The required MCPR's are essentially independent of peak location for power distributions that peak in the middle and upper portions of the core. However, for power distributions that peak near the bottom of the core, the required MCPR is reduced.

The applied R factors of 1.098 (1.154 for low enrichment bundles) for 7 x 7 fuel are taken at the beginning of cycle to reasonably bound the expected operating conditions. During the cycle the local peaking and therefore the R factors are reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced and end-of-cycle R factors, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

Rod Withdrawal Error Transient

The licensee discussed the rod withdrawal error transient in terms of worst case conditions.(3,6) The analysis shows that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop rod withdrawal while the critical power ratio is still greater than the 1.06 MCPR safety limit, and the cladding plastic strain limit of one percent is not exceeded. We conclude that the consequences of this localized transient are acceptable.

Operating MCPR Limits for Less than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to Technical Specifications limiting conditions for operation, Figure 3.1.1. This requires the licensee to maintain the required operating MCPR greater than 1.37 times the K_f factor for core flows less than rated. The K_f factor curves were generically derived to assure that the most limiting transient occurring at less than rated flow will not result in a MCPR below the safety limit of 1.06. We conclude that the submitted safety analyses of abnormal operational transients for FitzPatrick Nuclear Power Plant are acceptable. The minimum operating limit MCPR established for FitzPatrick that is required to avoid violation of the Safety Limit MCPR, should the most limiting transient occur, is acceptable.

Overpressure Protection

The licensee submitted an overpressure analysis in order to demonstrate that an adequate margin exists below the ASME code allowable pressure of 110% of vessel design pressure. The transient was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed based on a 104% steady state power level with the end of cycle scram reactivity applicable to the initial (current) fuel cycle, no credit for relief valve operation, and all safety valves operable. The peak pressure at the bottom of the vessel was calculated to be 1310 psig yielding a margin of 65 psig below the allowable 1375 psig ASME code limit (110% of the 1250 psig design pressure). In addition, the licensee referenced results of a sensitivity study [7] performed for BWR-4 reactors indicating that for one failed safety valve the results would increase about 20 psi which would still leave a margin of 45 psi for the required analysis with one failed valve.

We find the overpressure analysis acceptable on the basis that the sensitivity study with one failed valve shows considerable margin below the allowable limit.

ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading "...the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On December 23, 1975, the licensee submitted an evaluation of the ECCS performance for the design basis piping break including the effects of plugged bypass flow holes and a modified LPCIS, for FitzPatrick[8] along with a proposed amendment requesting changes to the Technical Specifications to implement the results of the evaluation. The licensee incorporated further information relating to the details of the ECCS evaluation by letters dated January 16, 1976[9] and January 29, 1976,[10] (which referenced an earlier July 24, 1975[11] submittal) to show compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in our Safety Evaluation Report of the FitzPatrick Nuclear Power Plant dated December 27, 1974.

The background of our review of the General Electric (GE) ECCS model and their application to FitzPatrick is described in the Safety Evaluation Report (SER) for that facility dated December 27, 1974. The December 27, 1974 SER was issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the Status Report of October 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974, SER also describes the various changes required in the earlier GE evaluation model. Together the December 27, 1974 SER and the Status Report and its Supplement, describe an acceptable ECCS evaluation model and the basis for our acceptance of the model. The FitzPatrick evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computations, the FitzPatrick analysis was based on a modified version of the SAFE computer code, with explicit consideration of our recommended limitations. These are described in the December 27, 1974 SER. The FitzPatrick evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluations provided.

We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses were provided by incorporating by reference analyses which were provided for Brunswick 2, [11] the lead plant for FitzPatrick after the LPCI modification was made. These analyses supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation discharge line. These additional calculations provided further details with regard to the limiting location and size of break as well as worst single failure for the FitzPatrick design. The limiting break, which is the design basis accident, is the complete severence of the recirculation discharge line assuming a failure of the LPCI injection valve.

We have reviewed the evaluation of ECCS performance submitted by the Power Authority of the State of New York for FitzPatrick Nuclear Power Plant and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR 50.46(a). Therefore, operation of the reactor would meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures D5A and D5B of the Power Authority of the State of New York submittal of December 1975 (reference 8) and to a minimum critical power ratio (MCPR) greater than 1.18. We note that the analyses submitted properly reflect effects due to: (1) recent plugging of the bypass flow holes in the core support plate (i.e., predicted reflooding is somewhat delayed) and (2) due to the LPCI modification (i.e., single failure assumptions allowing an increase in the MAPLHGR limit). However, certain changes must be made to the proposed technical specifications to conform with the evaluation of ECCS performance.

Credit is now taken for flow from one LPCI loop for the largest suction line break. The other LPCI loop is lost due to assumed single failure. If no LPCI flow were available, as would be the case if the nonsingle failed loop were inoperable, the suction line break would be more limiting than the discharge line break which was analyzed herein as the limiting break. For this reason, the plant will be required to operate no more than seven days with one inoperable LPCI loop or pump.

The largest recirculation break area assumed in the evaluation was 2.336 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, a license condition has been added which prohibits reactor operation unless the valve in the equalizer line is closed.

The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, a more limiting technical specification limits operation of the reactor to a MCPR of 1.37 for 7 x 7 fuel based on consideration of a turbine trip transient with failure of bypass valves. A statement has been added to the bases for the MCPR limiting condition of operation indicating that the MCPR value used in the ECCS performance evaluation has been appropriately considered.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, a license condition has been added which prohibits reactor operation under such conditions.

The licensee submitted ECCS LOCA analysis is in conformance to the requirements of Appendix K to 10 CFR Part 50. The reactor operating restrictions based on the submitted analysis are noted elsewhere in this Safety Evaluation.

Low Pressure Coolant Injection System (LPCIS) Modification

We issued a Safety Evaluation (15) dated January 15, 1976, authorizing installation, but not use, of a modified LPCIS at FitzPatrick. The acceptability of the LPCIS modification was addressed in that document with the exception of the emergency electrical distribution system.

Our review of the swing bus arrangement currently installed in the FitzPatrick emergency electrical distribution system to the LPCIS has shown it to be questionable. There are certain undetectable failures within the transfer circuitry that, if present when the bus transfer were required, could prevent the bus from transferring to its alternate source. There are also certain single failures that could tie the two diesel generator sets together through either of the swing buses. We informed the licensee that this design was questionable and that separate and independent buses would be required to bring the 480-volt portion of the onsite emergency power system into conformance with the recommendations of Regulatory Guide 1.6.

In order to alleviate the problem of potential undetected failure prior to completion of any required electrical modifications, we shall place a technical specification requirement on these transfer circuits that they be tested bi-monthly.

The licensee, in a letter to NRC dated January 8, 1976, stated, "In the event that we are unable to demonstrate the present power supply (swing bus arrangement) to the modified LPCIS is safe and reliable and within the NRC regulations, the Authority will design and implement a revised power system no later than the first major refueling shutdown which is presently scheduled for late 1977." This commitment from the licensee with the added surveillance Technical Specifications on the existing swing bus arrangement, represents a satisfactory interim emergency power supply to the LPCIS.

The loop selection logic circuitry of the LPCIS has been removed from the control room panels. Removal of this logic circuitry allows both injection valves to open, given an accident signal, no matter where the pipe break is located. Opening both injection valves requires that the RHR crosstie valve remain closed during normal plant operations and accident conditions. The licensee has altered the keylock switch on the control room panel which operates the crosstie valve from keylock open to keylock close, and the crosstie valve circuit breaker at the motor control center cubical is padlocked open with the valve closed. An annunciator has been added to alarm whenever the crosstie valve is open. We find these changes to be an acceptable method of assuring that this valve will remain closed during normal plant operation and accident conditions and are, therefore, acceptable.

Due to the elimination of the loop selection logic, the accident initiation signals have been rewired to direct (1) both LPCI injection valves to open, (2) both recirculation loop discharge valves to close when reactor pressure decreases to an appropriate setting and (3) LPCI pumps to start from two divisions instead of one (i.e., each pump and valve will receive a one-out-of-two logic initiation) upon detection of accident conditions.

The LPCI system redundant injection valves, pumps and recirculation valves are controlled by a-c control power relays in their control circuitry. These relays are in turn controlled by redundant 125-volt d-c output relays provided in each actuation train in the LPCI logic panels. This assures that failure of the 120-volt d-c power supply of either train will not prevent operation of any valve and pump in either train. Separation has been provided within the logic panels and wiring between the two logic panels is run through separate conduit. Separation of A & B circuits is maintained by the conduit so that any assumed failure of a conduit run will not prevent the operation of the redundant or associated control systems. We conclude that these design changes do not compromise the separation and independence of the two safety trains and are acceptable.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: MAR. 1 2 1978

References

1. "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," NEDO-10958 and NEDE-10958.

2. FitzPatrick Nuclear Power Plant Void Reactivity Coefficients letter to Robert A. Purple, Chief, Operating Reactors Branch #1, NRC, from George T. Berry, Power Authority of the State of New York, August 1975.

3. NEDO-21166, James A. FitzPatrick Nuclear Power Plant Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged, February 11, 1976.

4. General Electric "Process Computer Performance Evaluation Accuracy," NEDO-20340, and Amendment 1, NEDO-20340-1, dated June, 1974 and December 1974, respectively.

5. Letter to Director of NRR from George T. Berry, Power Authority of the State of New York, February 5, 1976, with answer to January 30, 1976, NRC questions on GETAB transient analyses.

6. Letter to Director of NRR from George T. Berry, Power Authority of the State of New York, October 28, 1975.

7. Letter to Director, NRR, from Ivan F. Stuart, GE, Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressures to Valve Operability, December 23, 1975.

8. Letter from General Electric Co. to Mr. G. T. Berry, Power Authority of the State of New York, FitzPatrick Nuclear Plant ECCS Appendix K Analysis, December 23, 1975, transmitted to NRC with attached Analysis and Technical Specification changes.

9. Letter to Director of NRR from George T. Berry, Power Authority of the State of New York, January 16, 1976.

10. Letter to Director of NRR from George T. Berry, Power Authority of the State of New York, January 29, 1976.

11. Brunswick Unit 2, Licensee Submittal, License No. DPR-62, 10 CFR Part 50 Appendix K Calculations and Revised Technical Specifications, dated May 9, 1975.

References (Cont'd)

12. Letter to Robert A. Purple, Chief, Operating Reactors Branch #1, NRC, from George T. Berry, Power Authority of the State of New York, LPCI System Modification, July 24, 1975.

13. James A. FitzPatrick Nuclear Power Plant - Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged, NEDO-21166. January 1976.

14. Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 9 to Facility Operating License No. DPR-59, January 30, 1976.

15. Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 8 to Facility Operating License No. DPR-59, January 15, 1976.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND NEGATIVE DECLARATION

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-59 issued to Power Authority of the State of New York and Niagara Mohawk Corporation which revised Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant, located in Oswego County, New York. The amendment is effective as of its date of issuance.

The amendment modifies the provisions in the Technical Specifications relating to Limiting Conditions for Operation associated with the Emergency Core Cooling System (ECCS), with plugged bypass flow holes, and Reactor Core Critical Power Limits and provides for modification of the ECCS to improve its performance in accordance with the licensees' application for amendment dated July 9, 1975.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notices of Proposed Issuance of Amendment to Facility Operating License in connection with this action were published in the FEDERAL REGISTER on August 8, 1975 (40 F.R. 3289), January 9, 1976 (41 F.R. 1657), and January 19, 1976 (41 F.R. 2695). No request for a hearing or petition for leave to intervene was filed following the notices of the proposed actions.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the James A. FitzPatrick Nuclear Power Plant issued March 1973, and that a negative declaration to this effect is appropriate.

For further details with respect to this action, see (1) the application for amendment dated July 9, 1975, as supplemented July 24, 1975, August 1, 1975, September 12 and 22, 1975, October 28, 1975, December 23, 1975, January 6, 8, 14, 16, 23, 26, 27, and 29, 1976, and February 5 and 11, 1976, (2) Amendment No. 14 to License No. DPR-59, (3) Amendment Nos. 8 and 9, issued January 15 and 30, 1976, respectively, (4) the Commission's related Safety Evaluation, and (5) the Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Oswego City Library, 120 E. Second Street, Oswego, New York.

A copy of items (2), (3), (4), and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 12th day of March 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors