Attachment I to JPN-96-009

REVISED TECHNICAL SPECIFICATION PAGES

PROPOSED TECHNICAL SPECIFICATION CHANGES
REGARDING IMPLEMENTATION OF BWROG OPTION I-D
LONG-TERM SOLUTION FOR THERMAL HYDRAULIC STABILITY

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

LIST OF PAGE CHANGES

Implementation of BWROG Option 1-D Long-Term Solution
for Thermal Hydraulic Stability (JPTS-96-005)

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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 Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed.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 the Safety Limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Startup and 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 APRM system obtains neutron flux input signals from LPRMs (fission chambers) and is calibrated to indicate percent rated thermal power. The APRM scrams in the run mode are a flow referenced scram and a fixed high neutron flux scram. As power rises during transients, the instantaneous neutron flux (as a percentage of rated) will rise faster than the rate of heat transfer from the fuel (percentage of rated thermal power) due to the thermal time constant of the fuel and core thermal power will be less than the power indicated by the APRMs (neutron flux) at either scram setting.

The APRM flow referenced scram trip setting, nominally varies from 54% power at 0% recirculation flow to 120% power at 100% recirculation flow but is limited to 117% rated power. The flow referenced trip will result in a significantly earlier scram during slow thermal transients, such as the loss of 80°F feedwater heating event, than would result from the 120% fixed high neutron flux scram. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower MCPR Operating Limits if such a transient

2.1 BASES (Cont'd)

c. APRM Flux Scram Trip Setting (Run Mode) (cont'd)

is the limiting abnormal operational transient during a certain exposure interval in the cycle. The flow referenced trip also provides protection for power oscillations which may result from reactor thermal hydraulic instability.

The APRM fixed high neutron flux scram protects the reactor during fast power increase transients if credit is not taken for a direct (position) scram or flow referenced scram.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted as specified in Table 3.1-1 when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by either reducing the APRM scram and rod block settings or adjusting the indicated APRM signal to reflect the high peaking condition.

Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR will be greater than the Safety Limit when the transient is initiated from the MCPR operating limits specified in the Core Operating Limits Report.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus provides an added level of protection before APRM Scram. 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 withdrawal. The flow variable trip setting parallels that of the APRM Scram and provides margin to scram, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2. Reactor Water Low Level Scram Trip Setting

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.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum No. of Operable Instrument Channels Per Trip System (Notes 1 and 2)	Trip Function	Trip Level Setting	Mode in Which Function Must Be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (Note 3)
			Refuel (Note 7)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch	A
1	Manual Scram		X	X	X	2	A
3	IRM High Flux	≤96% (120/125) of full scale	X	X		8	A
3	IRM Inoperative		X	X		8	A
2	APRM Neutron Flux-Startup (Note 15)	≤15% Power	X	X		6	A
2	APRM Flow Referenced Neutron Flux (Not to exceed 117%) (Note 13)	(Note 12)			X	6	A or B
2	APRM Fixed High Neutron Flux	≤120% Power			X	6	A or B
2	APRM Inoperative	(Note 10)	X	X	X	6	A or B

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TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit specified in the Core Operating Limits Report.

13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained as specified in the Core Operating Limits Report.

14. Deleted.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is placed in the Run position.

16. Instrumentation common to PCIS.

3.5 (cont'd)

J. Thermal Hydraulic Stability

1. When the reactor is in the run mode:

   a. Under normal operating conditions the reactor shall not be intentionally operated within the Power/Flow Exclusion Region defined in the Core Operating Limits Report (COLR).

   b. If the reactor has entered the Power/Flow Exclusion Region, the operator shall immediately insert control rods and/or increase recirculation flow to establish operation outside the region.

K. Single-Loop Operation

1. The reactor may be started and operated, or reactor operation may continue, with a single Reactor Coolant System recirculation loop in operation. The requirements applicable to single-loop operation in Specifications 1.1.A, 2.1.A, 3.1.A, 3.1.B, 3.2.C, and 3.5.H shall be in effect within 8 hours, or the reactor shall be placed in at least the hot shutdown mode within the following 12 hours.

2. During resumption of two-loop operation following a period of single-loop operation, the discharge valve of the lower speed pump shall not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

3. With no Reactor Coolant System recirculation loop in service, the reactor shall be placed in at least the hot shutdown mode within 12 hours.

3.5 BASES (cont'd)

J. Thermal Hydraulic Stability

10 CFR 50, Appendix A, General Design Criterion 12 requires that power oscillations are either prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal hydraulic instability which results in power oscillations, a power/flow exclusion region to be avoided during normal operation is calculated using the approved methodology specified in Technical Specification 6.9(A)4. Since the exclusion region may change each fuel cycle, the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in the exclusion region and to immediately exit the region if entered. Entries into the exclusion region are not part of normal operation, but may result from an abnormal event, such as a single recirculation pump trip or loss of feedwater heating, or be required to prevent equipment damage. In these events, time spent within the exclusion region is minimized.

Although operator actions can prevent the occurrence of and protect the reactor from an instability, the APRM flow-biased reactor scram will suppress power oscillations prior to exceeding the fuel safety limit (MCPR). Reference 3.5.L.2 demonstrated that this protection is provided at a high statistical confidence level for core-wide mode oscillations and at a nominal statistical confidence level for regional mode oscillations. This reference also demonstrated that the core-wide mode of oscillation is preferred due to the large single-phase channel pressure drop associated with the small fuel inlet orifice diameters.

K. Single-Loop Operation

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

L. References

1. "FitzPatrick Nuclear Power Plant Single-Loop Operation", NEDO-24281, August 1980.

2. "Application of the 'Regional Exclusion with Flow-Biased APRM Neutron Flux Scram' Stability Solution (Option I-D) to the James A. FitzPatrick Nuclear Power Plant," GENE-637-044-0295, February 1995

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(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification 3.5.H;
- The Minimum Critical Power Ratio (MCPR) and MCPR low flow adjustment factor, K_f, of Specifications 3.1.B and 4.1.E;
- The Linear Heat Generation Rate (LHGR) of Specification 3.5.I;
- The Reactor Protection System (RPS) APRM flow biased trip settings of Table 3.1-1;
- The flow biased APRM and Rod Block Monitor (RBM) rod block settings of Table 3.2-3; and
- The Power/Flow Exclusion Region of Specification 3.5.J.

and shall be documented in the Core Operating Limits Report (COLR).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendments.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October, 1986 including latest errata and addenda.
3. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.
4. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1991.
5. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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Attachment II to JPN-96-009

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING IMPLEMENTATION OF BWROG OPTION I-D LONG-TERM SOLUTION FOR THERMAL HYDRAULIC STABILITY

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

I. DESCRIPTION OF THE PROPOSED CHANGES

The following proposed changes to the James A. FitzPatrick Technical Specifications establish operability requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with BWROG long-term solution Option I-D (References 1 and 2). Editorial changes are also made to support the revised specifications, improve readability of Bases sections, and enhance the presentation of requirements for single loop operation.

Reference 3 issued an amendment with similar Technical Specification changes to the Vermont Yankee Nuclear Power Station (VYNPS). VYNPS is the lead plant for the Option I-D stability solution.

Page ii

Replace SR 3.5.J with "DELETED." Add line for LCO (3.5) "K. Single-Loop Operation" with no associated SR, denoted by "NONE" on page 124a.

Page vii

Replace title for Figure 3.5-1 with "(Deleted)," delete page number.

Pages 17 and 18

Change BASES Section 2.1.c to read:

"The APRM system obtains neutron flux input signals from LPRMs (fission chambers) and is calibrated to indicate percent rated thermal power. The APRM scrams in the run mode are a flow referenced scram and a fixed high neutron flux scram. As power rises during transients, the instantaneous neutron flux (as a percentage of rated) will rise faster than the rate of heat transfer from the fuel (percentage of rated thermal power) due to the thermal time constant of the fuel and core thermal power will be less than the power indicated by the APRMs (neutron flux) at either scram setting.

The APRM flow referenced scram trip setting, nominally varies from 54% power at 0% recirculation flow to 120% power at 100% recirculation flow but is limited to 117% rated power. The flow referenced trip will result in a significantly earlier scram during slow thermal transients, such as the loss of 80 F feedwater heating event, than would result from the 120% fixed high neutron flux scram. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower MCPR Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle. The flow referenced trip also provides protection for power oscillations which may result from reactor thermal hydraulic instability.

The APRM fixed high neutron flux scram protects the reactor during fast power increase transients if credit is not taken for direct (position) scram or flow referenced scram.

increase transients if credit is not taken for direct (position) scram or flow referenced scram.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted as specified in Table 3.1-1 when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by either reducing the APRM scram and rod block settings or adjusting the indicated APRM signal to reflect the high peaking condition.

Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR will be greater than the Safety Limit when the transient is initiated from the MCPR operating limits specified in the Core Operating Limits Report."

Page 40

Delete reference to Note 14 for APRM Flow Referenced Neutron Flux. The revised specification reads, "APRM Flow Referenced Neutron Flux (Not to exceed 117%) (Note 13)."

Delete reference to Note 14 for APRM Fixed High Neutron Flux. The revised specification reads, "APRM Fixed High Neutron Flux."

Page 43a

Delete text for Note 14 and replace with "Deleted."

Pages 124a through 124c

Change LCO 3.5.J to read as follows:

"J. Thermal Hydraulic Stability

1. When the reactor is in the run mode:

   a. Under normal operating conditions the reactor shall not be intentionally operated within the Power/Flow Exclusion Region defined in the Core Operating Limits Report (COLR).

   b. If the reactor has entered the Power/Flow Exclusion Region, the operator shall immediately insert control rods and/or increase recirculation flow to establish operation outside the region."

Delete LCOs 3.5.J.2 and 3.5.J.3.

Delete SR 4.5.J.1.

Renumber and revise LCOs 3.5.J.4 through 3.5.J.6 to read:

"K. Single-Loop Operation

1. The reactor may be started and operated, or reactor operation may continue, with a single Reactor Coolant System recirculation loop in operation. The requirements applicable to single-loop operation in Specifications 1.1.A, 2.1.A, 3.1.A, 3.1.B, 3.2.C, and 3.5.H shall be in effect within 8 hours, or the reactor shall be placed in at least the hot shutdown mode within the following 12 hours.

2. During resumption of two-loop operation following a period of single-loop operation, the discharge valve of the lower speed pump shall not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

3. With no Reactor Coolant System recirculation loop in service, the reactor shall be placed in at least the hot shutdown mode within 12 hours."

Delete pages 124b and c.

Page 131

Revise Bases Section 3.5.J to read:

"J. Thermal Hydraulic Stability

10 CFR 50, Appendix A, General Design Criterion 12 requires that power oscillations are either prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal hydraulic instability which results in power oscillations, a power/flow exclusion region to be avoided during normal operation is calculated using the approved methodology specified in Technical Specification 6.9(A)4. Since the exclusion region may change each fuel cycle, the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in the exclusion region and to immediately exit the region if entered. Entries into the exclusion region are not part of normal operation, but may result from an abnormal event, such as a single recirculation pump trip or loss of feedwater heating, or be required to prevent equipment damage. In these events, time spent within the exclusion region is minimized.

Although operator actions can prevent the occurrence of and protect the reactor from an instability, the APRM flow-biased reactor scram will suppress power oscillations prior to exceeding the fuel safety limit (MCPR). Reference 3.5.L.2 demonstrated that this protection is provided at a high statistical confidence level for core-wide mode oscillations and at a nominal statistical confidence level for regional mode oscillations. This reference also demonstrated that the core-wide mode of oscillation is preferred due to the large single-phase channel pressure drop associated with the small fuel inlet orifice diameters."

Relocate the last paragraph of Bases Section 3.5.J to a revised Bases Section 3.5.K, renumber Bases Section 3.5.K to be Bases Section 3.5.L and add reference 3.5.L.2:

"K. Single-Loop Operation

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

L. References

1. "FitzPatrick Nuclear Power Plant Single-Loop Operation", NEDO-24281, August 1980.

2. "Application of the 'Regional Exclusion with Flow-Biased APRM Neutron Flux Scram' Stability Solution (Option I-D) to the James A. FitzPatrick Nuclear Power Plant," GENE-637-044-0295, February 1995."

Page 134

Delete Figure 3.5-1, mark page "(THIS PAGE INTENTIONALLY BLANK)."

Page 254-c through f

Change page number 254-c to 254c.
Add another bullet to section 6.9(A)4.a:

"• The Power/Flow Exclusion Region of Specification 3.5.J."

Add new sections 6.9(A)4.b.4 and 6.9(A)4.b.5:

"4. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1991.

5. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992."

Move sections 6.9(A)4.c and 6.9(A)4.d to page 254d. Change page number of following page to "254e,f."

II. PURPOSE OF THE PROPOSED CHANGES

The purpose of the proposed changes is to implement controls over plant operation which ensure compliance with General Design Criteria (GDC) 10 and 12 of 10 CFR 50 Appendix A.

Following the thermal hydraulic instability event at LaSalle in 1988 (Reference 4), a BWR Owners' Group committee was formed to obtain resolution of NRC concerns over compliance with GDC 10 and 12. The BWR Owners' Group developed several stability long-term solutions which addressed these issues, which were subsequently approved by the NRC (References 1 and 2).

NYPA has chosen to implement the Option I-D stability solution and has submitted a plant unique assessment demonstrating the suitability of the solution for FitzPatrick to the NRC (Reference 5). Option I-D requires use of the flow-biased Average Power Range Monitor high neutron flux scram without Simulated Thermal-Power Monitor (STPM), and a power/flow map exclusion region while adhering to certain power distribution limitations.

The purpose of changes related to removal of the STPM is to ensure the Technical Specifications reflect the plant configuration following modification of the APRMs to support the Option I-D stability solution.

The purpose of changes related to replacement of a generic restricted region (Figure 3.5-1, being removed from the Technical Specifications) which may be entered with appropriate monitoring, with a cycle specific exclusion region (to be provided in the COLR) which is not to be entered during normal operation, is to prevent the occurrence of thermal hydraulic oscillations while removing unnecessary operating limitations.

The purpose of the editorial changes separating the requirements for thermal hydraulic stability from the requirements for single loop operation is to enhance the clarity of the Technical Specifications.

The purpose of specifying establishing the thermal hydraulic stability exclusion region on a cyclic basis in the COLR is to ensure that the region is correct for the reload core design.

Other editorial changes made are the addition or deletion of blank pages as required by expanded or contracted text sections and revision of the table of contents to reflect changed section numbering.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

Changes to pages 17, 18, 40 and 43a reflect removal of the STPM function from the APRM flow-biased scram trip circuit. This allows the flow referenced scram to provide protection of the fuel safety limit (minimum critical power ratio, MCPR) for reactor instabilities. It is necessary to disable the filter of the STPM (which models the thermal time constant of the fuel) so that the scram trip circuit will provide adequate protection for thermal hydraulic oscillations. The density wave oscillations associated with thermal hydraulic instability have a natural frequency of 0.3 to 0.7 Hz, which corresponds to a period of 1.4 to 3 seconds, compared to a time constant of about 6 seconds for the STPM. When the APRM neutron flux signal is processed through the STPM, the magnitude of the oscillations associated with thermal hydraulic instability will be greatly reduced because the STPM time constant is significantly longer than the period of the oscillation. With the STPM processing removed from the APRM flow referenced scram trip input, the signal will not be reduced in magnitude, and a scram will occur in time to limit the ΔCPR to a value such that the MCPR safety limit is not violated.

The analyses presented in Reference 5 demonstrate that the flow-biased APRM scram will provide suppression of thermal hydraulic oscillations prior to the MCPR safety limit being violated. These analyses apply to core-wide oscillations with a high degree of statistical confidence, and to regional oscillations with a nominal degree of statistical confidence.

Removal of the STPM filter may result in reactor scrams which would not have occurred with the time constant in place. In the event of such a scram, compliance with Technical Specification limitations on Reactor Coolant System (RCS) heatup and cooldown rates, and design limitations on RCS thermal cycles will ensure RCS thermal stresses are adequately controlled and accounted for. Because the unfiltered APRM flow biased trip provides protection for the MCPR safety limit for potential reactor instabilities, a net safety benefit is gained.

Changes to pages 124a through 124c, 131 and 134 include those which provide increased protection against potential fuel damage as a result of unstable reactor operation. The present Technical Specifications allow operation within a generically defined region of the core power/flow map in which thermal hydraulic instability is thought to be possible, provided the operators utilize enhanced monitoring to detect an instability and provided action is taken to suppress an instability, should it occur. The proposed changes prohibit normal operation within a plant specific power/flow exclusion region in which instabilities are conservatively predicted to occur. If entry is made into the exclusion region as a result of a plant transient, the proposed changes require immediate exit by either increasing reactor water recirculation flow or by inserting control rods. Therefore, the proposed changes reduce the potential of occurrence of thermal hydraulic instabilities.

An editorial change was made to these pages to separate the controls required for protection against instability from those which are required to permit single loop operation. The controls for single loop operation were relocated to a separate section (3.5.K) of the Technical Specifications with no change in technical content. The power/flow exclusion region is not dependent on the number of recirculation loops in service, therefore it is not necessary to couple the requirements for single loop operation with those related to thermal hydraulic stability. This change improves the clarity of the Technical Specifications.

The change to page 254c requires that the power/flow exclusion region be established for each operating cycle in accordance with NRC approved methods and included in the Core Operating Limits Report. This ensures that the stability limits are appropriate for the current core configuration.

Editorial changes to the table of contents and page additions and deletions do not alter any operability or surveillance requirements contained in the Technical Specifications. Therefore, these changes have no effect on safety.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated because:

   The implementation of BWR Owners' Group long-term stability solution Option I-D at FitzPatrick does not modify the assumptions contained in the existing accident analysis. The use of an exclusion region and the operator actions required to avoid and minimize operation inside the region do not increase the possibility of an accident. Conditions of operation outside of the exclusion region are within the analytical envelope of the existing safety analysis. The operator action requirement to exit the exclusion region upon entry minimizes the possibility of an oscillation occurring. The actions to drive control rods and/or to increase recirculation flow to exit the region are maneuvers within the envelope of normal plant evolutions. The flow referenced scram has been analyzed and will provide automatic fuel protection in the event of an instability. Thus, each proposed operating requirement provides defense in depth for protection from an instability event while maintaining the existing assumptions of the accident analysis.

2. create the possibility of a new or different kind of accident from those previously evaluated because:

   The proposed operating requirements either mandate operation within the envelope of existing plant operating conditions or force specific operating maneuvers within those carried out in normal operation. Since operation of the plant with all of the proposed requirements are within the existing operating basis, an unanalyzed accident will not be created through implementation of the proposed change.

3. involve a significant reduction in the margin of safety because:

   Each of the proposed requirements for plant thermal hydraulic stability provides a means for fuel protection. The combination of avoiding possible unstable conditions and the automatic flow referenced reactor scram provides an in depth means for fuel protection. Therefore, the individual or combination of means to avoid and suppress an instability supplements the margin of safety.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not adversely affect the ALARA or Fire Protection Program at the FitzPatrick plant, nor will the changes impact the environment.

VI. CONCLUSION

Based on the discussions above, the implementation of the BWROG long-term stability solution Option I-D at FitzPatrick does not involve a significant hazards consideration, or an unreviewed safety question, and will not endanger the health and safety of the public. The Plant Operating Review Committee and Safety Review Committee have reviewed this proposed Technical Specification change and agree with this conclusion.

VII. REFERENCES

(1) BWR Owners' Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960-A, June 1991.

(2) BWR Owners' Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960-A, Supplement 1, March 1992

(3) NRC Letter, Daniel H. Dorman to Donald A. Reid, "Issuance of Amendment (TAC NO. M89201)," dated August 9, 1995

(4) NRC Information Notice No. 88-39: LaSalle Unit 2 Loss of Recirculation Pumps with Power Oscillation Event, June 15,1988

(5) NYPA Letter, William J. Cahill, Jr. to NRC (JPN-95-032), "Submittal of Plant Specific Licensing Topical Report for Long-Term Solution on Reactor Stability (Generic Letter 94-02)," dated June 29, 1995

Attachment III to JPN-96-009

MARKUP OF TECHNICAL SPECIFICATION PAGES

PROPOSED TECHNICAL SPECIFICATION CHANGES
REGARDING IMPLEMENTATION OF BWROG OPTION I-D
LONG-TERM SOLUTION FOR THERMAL HYDRAULIC STABILITY

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

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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 the Safety Limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Startup and 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)

6

~~The APRM flux scram trip in the run mode consists of a flow referenced scram setpoint and a fixed high neutron flux scram setpoint. The APRM flow referenced neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions. This prevents spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Examples of events which can result in momentary neutron flux spikes are momentary flow changes in the recirculation system flow, and small pressure disturbances during turbine stop valve and turbine control valve testing. These flux spikes represent no hazard to the fuel since they are only of a few seconds duration and less than 120% of rated thermal power.~~

6

~~The APRM flow referenced scram trip setting at full recirculation flow is adjustable up to 117% of~~

Insert a

Amendment No. ~~43~~,

Insert a:

The APRM system obtains neutron flux input signals from LPRMs (fission chambers) and is calibrated to indicate percent rated thermal power. The APRM scrams in the run mode are a flow referenced scram and a fixed high neutron flux scram. As power rises during transients, the instantaneous neutron flux (as a percentage of rated) will rise faster than the rate of heat transfer from the fuel (percentage of rated thermal power) due to the thermal time constant of the fuel and core thermal power will be less than the power indicated by the APRMs (neutron flux) at either scram setting.

The APRM flow referenced scram trip setting, nominally varies from 54% power at 0% recirculation flow to 120% power at 100% recirculation flow but is limited to 117% rated power. The flow referenced trip will result in a significantly earlier scram during slow thermal transients, such as the loss of 80°F feedwater heating event, than would result from the 120% fixed high neutron flux scram. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower MCPR Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle. The flow referenced trip also provides protection for power oscillations which may result from reactor thermal hydraulic instability.

The APRM fixed high neutron flux scram protects the reactor during fast power increase transients if credit is not taken for a direct (position) scram or flow referenced scram.

2.1 BASES (Cont'd)

c. APRM Flux Scram Trip Setting (Run Mode) (cont'd)

rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 80°F feedwater heating event, than would result with the 120% fixed high neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle.

The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced scram.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted as specified in Table 3.1-1 when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by either ~~(1)~~ reducing the APRM scram and rod block settings or ~~(2)~~ adjusting the indicated APRM signal to reflect the high peaking condition.

Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR will be greater than the Safety Limit when the transient is initiated from the MCPR operating limits specified in the Core Operating Limits Report.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus provides an added level of protection before APRM Scram. 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 withdrawal. The flow variable trip setting parallels that of the APRM Scram and provides margin to scram, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2. Reactor Water Low Level Scram Trip Setting

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.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum No. of Operable Instrument Channels Per Trip System (Notes 1 and 2)	Trip Function	Trip Level Setting	Mode in Which Function Must be Operable: Refuel (Note 7)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (Note 3)
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch	A
1	Manual Scram		X	X	X	2	A
3	IRM High Flux	≤ 96% (120/125) of full scale	X	X		8	A
3	IRM Inoperative		X	X		8	A
2	APRM Neutron Flux-Startup (Note 15)	≤ 15% Power	X	X		6	A
2	APRM Flow Referenced Neutron Flux (Not to exceed 117%) (~~Notes~~ 13 ~~and 14~~)	(Note 12)			X	6	A or B
2	APRM Fixed High Neutron Flux ~~(Note 14)~~	≤ 120% Power			X	6	A or B
2	APRM Inoperative	(Note 10)	X	X	X	6	A or B

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit specified in the Core Operating Limits Report.

13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained as specified in the Core Operating Limits Report.

14. ~~The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.~~ Deleted.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is placed in the Run position.

16. Instrumentation common to PCIS.

3.5 (cont'd)

J. Thermal Hydraulic Stability

1. Whenever the reactor is in the startup or run modes, two Reactor Coolant System recirculation loops shall be in operation, with:

   a. Total core flow greater than or equal to 45 percent of rated, or

   b. Thermal power less than or equal to the limit specified in Figure 3.5-1 (Line A).

   except as specified in Specifications 3.5.J.2 and 3.5.J.3.

2. With two Reactor Coolant System recirculation loops in operation and total core flow less than 45 percent of rated, and thermal power greater than the limit specified in Figure 3.5-1 (Line A); or with one Reactor Coolant System loop operating and thermal power greater than the limit specified in Figure 3.5-1 (Line A):

   a. Determine the APRM and LPRM noise levels:

      1. Within 2 hours after reaching steady-state within the regions of Figure 3.5-1 where monitoring is required, and at least once per 8 hours thereafter; and

Insert b

4.5 (cont'd)

J. Thermal Hydraulic Stability

1. Establish baseline APRM and LPRM neutron flux noise values within 2 hours of entering the region for which monitoring is required unless baselining has been performed since the last refueling outage. Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

6

Amendment No. ~~30~~, ~~98~~,

3.5 (cont'd)

2. Within 2 hours after completing an increase in thermal power of 5 percent or more of rated thermal power.

b. If the APRM and LPRM neutron flux noise levels are greater than 5 percent and greater than three times their established baseline noise levels, initiate corrective action within 15 minutes to restore the noise levels to within the required limits within 2 hours, by increasing core flow and/or reducing thermal power.

3. If during single-loop operation, core thermal power is greater than the limit defined by line A of Figure 3.5-1, and core flow is less than 39 percent, immediately initiate corrective action to restore core thermal power and/or core flow to within the limits, specified in Figure 3.5-1, by increasing core flow and/or initiating an orderly reduction of core thermal power by inserting control rods.

4. The requirements applicable to single-loop operation in Specifications 1.1.A, 2.1.A, 3.1.A, 3.1.B, 3.2.C and 3.5.H shall be in effect within 8 hours following the removal of one recirculation loop from service, or the reactor shall be placed in at least the hot shutdown condition within 12 hours.

3.5 (cont'd)

5. ~~During resumption of two-loop operation following a period of single-loop operation, the discharge valve of the low-speed pump shall not be opened unless the speed of the faster pump is less than 50 percent of it's rated speed.~~

6. ~~With no Reactor Coolant System Recirculation loop in service, the reactor shall be placed in Hot Shutdown within 12 hours.~~

Insert b:

J. Thermal Hydraulic Stability

1. When the reactor is in the run mode:

   a. Under normal operating conditions the reactor shall not be intentionally operated within the Power/Flow Exclusion Region defined in the Core Operating Limits Report (COLR).

   b. If the reactor has entered the Power/Flow Exclusion Region, the operator shall immediately insert control rods and/or increase recirculation flow to establish operation outside the region.

K. Single-Loop Operation

1. The reactor may be started and operated, or reactor operation may continue, with a single Reactor Coolant System recirculation loop in operation. The requirements applicable to single-loop operation in Specifications 1.1.A, 2.1.A, 3.1.A, 3.1.B, 3.2.C, and 3.5.H shall be in effect within 8 hours, or the reactor shall be placed in at least the hot shutdown mode within the following 12 hours.

2. During resumption of two-loop operation following a period of single-loop operation, the discharge valve of the lower speed pump shall not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

3. With no Reactor Coolant System recirculation loop in service, the reactor shall be placed in at least the hot shutdown mode within 12 hours.

3.5 BASES (cont'd)

J. Thermal Hydraulic Stability

~~Operation in certain regions of the power vs. flow curve have been identified as having a high potential for thermal hydraulic instability (Figure 3.5-1). These regions are located in the high power/low flow area of the curve and can be encountered during startup, shutdown, rod sequence exchange or recirculation pump trip. Operation in these regions is associated with higher than normal neutron flux noise levels. Increased awareness of LPRM and APRM signal noise when operating in these regions will identify instability and allow operator action to correct the problem. The neutron flux noise level, thermal power and core flow limits are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability", dated February 10, 1984.~~ ← Insert C

← K. Single-Loop Operation

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

L ~~K~~. References

1. "FitzPatrick Nuclear Power Plant Single-Loop Operation", NEDO-24281, August 1980.

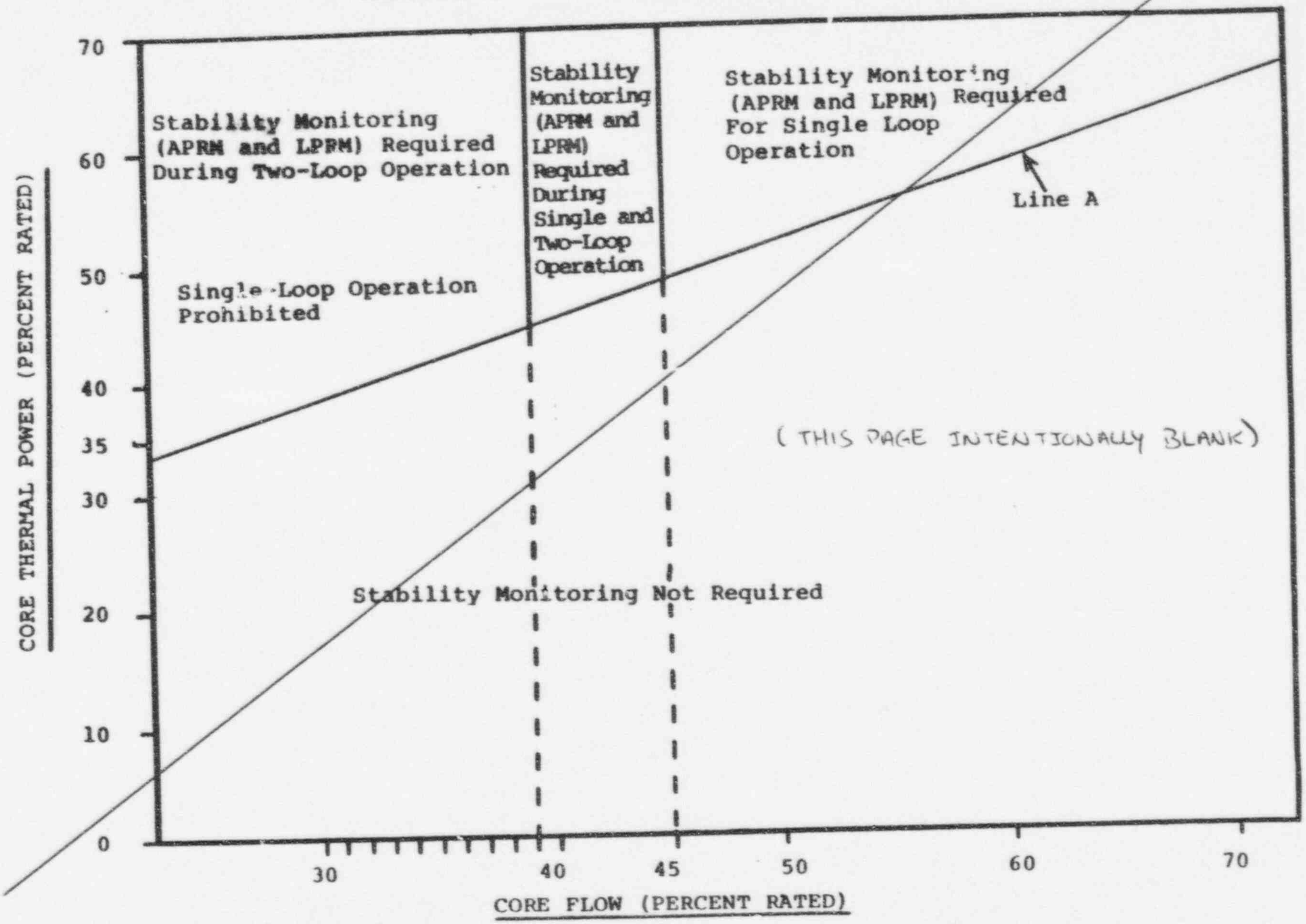
2. "Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) to the James A. FitzPatrick Nuclear Power Plant," GENE-637-044-0295, February 1995

Insert c:

10 CFR 50, Appendix A, General Design Criterion 12 requires that power oscillations are either prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal hydraulic instability which results in power oscillations, a power/flow exclusion region to be avoided during normal operation is calculated using the approved methodology specified in Technical Specification 6.9(A)4. Since the exclusion region may change each fuel cycle, the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in the exclusion region and to immediately exit the region if entered. Entries into the exclusion region are not part of normal operation, but may result from an abnormal event, such as a single recirculation pump trip or loss of feedwater heating, or be required to prevent equipment damage. In these events, time spent within the exclusion region is minimized.

Although operator actions can prevent the occurrence of and protect the reactor from an instability, the APRM flow-biased reactor scram will suppress power oscillations prior to exceeding the fuel safety limit (MCPR). Reference 3.5.L.2 demonstrated that this protection is provided at a high statistical confidence level for core-wide mode oscillations and at a nominal statistical confidence level for regional mode oscillations. This reference also demonstrated that the core-wide mode of oscillation is preferred due to the large single-phase channel pressure drop associated with the small fuel inlet orifice diameters.

Figure 3.5-1
Thermal Power and Core Flow Limits of Specifications 3.5.J.1, 3.5.J.2 and 3.5.J.3



(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification 3.5.H;
- The Minimum Critical Power Ratio (MCPR) and MCPR low flow adjustment factor, K_f, of Specifications 3.1.B and 4.1.E;
- The Linear Heat Generation Rate (LHGR) of Specification 3.5.I;
- The Reactor Protection System (RPS) APRM flow biased trip settings of Table 3.1-1; ~~and~~
- The flow biased APRM and Rod Block Monitor (RBM) rod block settings of Table 3.2-3; and
- The Power/Flow Exclusion Region of Specification 3.5.J.

and shall be documented in the Core Operating Limits Report (COLR).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendments.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October, 1986 including latest errata and addenda.
3. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

move to next page

4. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1991.

5. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992.

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