Yellow

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

JUL 23 1975

DISTRIBUTION:
Dockets (2)
NRC PDR
Local PDR
ORB#3 rdg
OELD
OI&E (3)
NDube
BJones (8)
JMMcGough
JSaltzman
SATeets
DJaffe
TVerdery
GLear
SKari
WOMiller
BScharf (15)
TJCarter
PCollins
SVarga
CHebron
AESteen
Abernathy
DEisenhut
BCRusche
ACRS (16)

Docket Nos. 50-277
and 50-278

Philadelphia Electric Company
Attn: Mr. Edward G. Bauer, Jr., Esquire
Vice President and General Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" issued by the Commission on July 23, 1975, for the Peach Bottom Atomic Power Station Units 2 and 3. The Order adds a provision to License Nos. DPR-44 and DPR-56 stating certain operating restrictions and surveillance requirements during power operation. The added provision is related to limiting channel box wear and its potential effects.

The Order permits Peach Bottom Units 2 and 3 to remain at design power and flow conditions for a total of 75 hours to allow the fuel preconditioning program to be completed. The operation at design power and flow will be done in conjunction with special surveillance required by the Order. The bases for allowing operation at design flow and power are that: (1) most likely, cracks do not yet exist in the channel boxes; (2) wear rate is relatively slow, even at 100 percent flow; (3) the likelihood of a thermally - challenging transient during such a brief time period is low; and (4) extra surveillance of core conditions will be employed.

O/P
1

LV

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosure:
Order for Modification of
License

OFFICE ➤	E.H.Verdery	OELD				
SURNAME ➤	G.Lear		K.R.Goller	A.Giambusso	E.G.Case	B.L.Rusche
DATE ➤	7/ /75	7/ /75	7/ /75	7/ /75	7/ /75	7/ /75

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

JUL 23 1975

Docket Nos. 50-277
and 50-278

Philadelphia Electric Company
Attn: Mr. Edward G. Bauer, Jr., Esquire
Vice President and General Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" issued by the Commission on July 23, 1975, for the Peach Bottom Atomic Power Station Units 2 and 3. The Order adds a provision to License Nos. DPR-44 and DPR-56 stating certain operating restrictions and surveillance requirements during power operation. The added provision is related to limiting channel box wear and its potential effects.

The Order permits Peach Bottom Units 2 and 3 to remain at design power and flow conditions for a total of 75 hours to allow the fuel preconditioning program to be completed. The operation at design power and flow will be done in conjunction with special surveillance required by the Order. The bases for allowing operation at design flow and power are that: (1) most likely, cracks do not yet exist in the channel boxes; (2) wear rate is relatively slow, even at 100 percent flow; (3) the likelihood of a thermally - challenging transient during such a brief time period is low; and (4) extra surveillance of core conditions will be employed.

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

JUL 23 1975

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosure:
1. Order for Modification of License
2. Safety Evaluation

cc: w/enclosures

Eugene J. Bradley
Philadelphia Electric Company
Assistant General Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Anthony Z. Roisman, Esquire
Berlin, Roisman & Kessler
1712 N Street, N. W.
Washington, D. C. 20036

Albert R. Steel, Chairman
Board of Supervisors
Peach Bottom Township
R. D. #1
Delta, Pennsylvania 17314

Wilmer P. Bolton
Chairman, Board of Supervisors
Drumore Township
R. D. #1
Holtwood, Pennsylvania 17532

Mr. R. A. Heiss, Coordinator
Pennsylvania State Clearinghouse
Governor's Office of State Planning
and Development
P. O. Box 1323
Harrisburg, Pennsylvania 17120

Raymond L. Hovis, Esquire
35 South Duke Street
York, Pennsylvania 17401

Troy B. Conner, Jr.
Conner, Hadlock & Knotts
1747 Pennsylvania Avenue, NW
Washington, D. C. 20006

W. W. Anderson, Esquire
Deputy Attorney General
Department of Justice
Second Floor - Capitol Annex
Harrisburg, Pennsylvania 17120

John B. Griffith, Esquire
Special Assistant Attorney
General, Maryland
Annapolis, Maryland 31401

Warren Rich, Esquire
Special Assistant Attorney
General, Maryland
Annapolis, Maryland 21401

Mr. Robert Blanco
Environmental Protection Agency
Region III Office
Curtis Building
6th and Walnut Street
Philadelphia, Pennsylvania 19106

Mrs. Louisa R. Yeagley
Martin Memorial Library
159 E. Market Street
York, Pennsylvania 17401

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
PHILADELPHIA ELECTRIC COMPANY	)	Docket Nos. 50-277
	)	50-278
(Peach Bottom Atomic Power Station,	)	
Units 2 and 3)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

Philadelphia Electric Company (PECO or Licensee) is the holder of Facility Operating License Nos. DPR-44 and DPR-56, which respectively authorize operation of Peach Bottom Atomic Power Station Units 2 and 3 (Units 2 and 3 or the Facilities) at steady-state reactor core power levels not in excess of 3293 megawatts thermal (rated power). The Facilities are boiling water reactors (BWR) located at the Licensee's site in Peach Bottom, York County, Pennsylvania.

II.

1. The Licensee's Facilities are two of eleven United States BWR facilities which are similarly designed in that each design provides for coolant flow through bypass holes in the reactor lower core support plate. Fuel inspections conducted in April 1975 in a similarly designed foreign BWR revealed significant wear and some cracking of several

Zircaloy fuel channel boxes.[1/] The discovery of this damage had been preceded by anomalous incore nuclear detector instrument readings from the traversing incore probe (TIP),[2/] the anomaly consisting of a band of "noise" with a characteristic frequency, superimposed on the normal signal. Both the TIP noise and the associated channel box damage appeared to be the result of vibration of the instrument tubes produced by coolant flow through the bypass holes.

2. In the period since April 1975, the eleven affected BWRs have been under surveillance by the NRC Staff, by GE, and by the individual reactor operators. Based on the experience of the foreign reactor, the NRC Staff has been of the view that a noise-to-signal ratio of 0.06 or more indicates the need for remedial action in the form of reduced core flow with a resultant reduction in core power. TIP readings at or above the 0.06 criterion were found in April 1975 at

---

1/ This information was conveyed to the Staff of the Nuclear Regulatory Commission (hereinafter referred to as the "NRC Staff") by the General Electric Company (GE) by telephone calls on April 17, 18, and 21, 1975, and by letter dated April 22, 1975. Copies of the April 22 letter and copies of all other documents referred to in this order are available or are being made available for public inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and the Commission's Local Public Document Room, Martin Memorial Library, 159 E. Market Street, York, Pennsylvania.

2/ Ibid.

the Cooper Station, and in May 1975 at the Duane Arnold Energy Center and at the Peach Bottom Atomic Power Station Unit 3. The operators of the three facilities agreed -- later confirmed by NRC Order -- to limit operation to core flow and power levels not exceeding 50% of design levels without prior written approval of the NRC Staff.[3/] PECO agreed by letter on June 6, 1975 to similarly limit operation of Peach Bottom Atomic Power Station, Unit 2.

3. The 50% power and 50% flow limitations were interim measures to provide assurance of safety while further data were being developed in tests, studies and investigations then in progress. The 50% power limitation was derived from the 50% flow limitation on the basis of certain inherent operating characteristics of BWRs.[4/] The flow limitation in turn was based on the results of tests conducted by GE in 1973 in connection with channel box damage due to control curtain vibration which had been observed in the Vermont Nuclear Power Station, in the

---

3/ For the details of these actions, see Order for Modification of License, In the Matter of Nebraska Public Power District (Cooper Nuclear Station), Docket No. 50-298, dated April 26, 1975; Order for Modification of License, In the Matter of Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (Duane Arnold Energy Center), Docket No. 50-331, dated May 21, 1975; and Order for Modification of License, In the Matter of Philadelphia Electric Company (Peach Bottom Atomic Station, Unit 3) Docket No. 50-278, dated June 2, 1975.

4/ See Safety Evaluation by the Director of Licensing, U.S. Atomic Energy Commission, Relating to Channel Box Wear in the Vermont Yankee Nuclear Power Station (Docket No. 50-271) and the Pilgrim Nuclear Power Station (Docket No. 50-293) dated October 26, 1973, pp.16-18.

Pilgrim Nuclear Power Station, and in a foreign reactor.[5/] The 1973 GE data was supported by observations made at the Cooper Station over a short period prior to the issuance of the NRC Staff's April 26 order while the Facility was operating at a limited flow of about 55% of full core flow. Similar observations were made in May 1975 in tests conducted at the Duane Arnold reactor.

4. Through an exchange of letters dated June 2, 1975, the operators of Duane Arnold requested and the NRC Staff approved a plan to operate that reactor for a short period of time to obtain TIP readings at core power levels up to 100% of rated power and core flow rates up to 100% of design flow rate. The objective of these measurements was to permit a determination, following a shutdown and inspection of the reactor, whether TIP noise data and channel box damage data could be correlated. After receiving and analyzing the Duane Arnold data, the NRC Staff concluded that such a correlation does indeed exist. The correlation also supports continued use of the 0.06 noise-to-signal ratio as a test of the need for remedial action.

5/ For details concerning the earlier matter affecting the Vermont Yankee and Pilgrim facilities see the AEC Regulatory Staff's related Safety Evaluation, n. 4, supra.

5. In the period since April 1975 there have been additional studies concerning the relationship of TIP noise to coolant flow rate. New relevant data are available from accelerometer tests, from TIP readings at Cooper and Duane Arnold, and from experiments conducted by GE on prototype TIP tube - channel box configurations. These data all point to the conclusion that TIP noise is significantly attenuated, if not arrested, at a coolant flow rate of about 40% of design flow. The NRC Staff has therefore concluded that a 40% flow rate limitation should apply to sustained operation of a reactor that has reached the action threshold of a 0.06 noise-to-signal ratio. However, considering inter alia the rate at which channel box wear occurs, limited operation at higher flows (for, e.g., the conduct of tests) may be permitted without endangering the health and safety of the public.

6. Investigations have also continued with respect to the appropriateness of the 50% core power limit on affected reactors. GE prepared, and forwarded to the NRC Staff by letter dated July 11, 1975, a proposal to substitute a maximum bundle power limit for the overall 50% core power limit now in use. GE's July 11 submission included supporting analyses performed with the assumption of a coolant flow of 40% of design flow, and with the further, conservative assumption of a cracked channel box.

7. By letter dated July 17, 1975, the Licensee formally proposed a plan, previously discussed with the NRC Staff, for operation of the Facilities in the short term, and for a determination of the conditions, if any, under which operation should be allowed in the longer term. The plan entails a reduction of coolant flow to 40% of design flow and a substitution of a 3.35 MWt maximum bundle power limit for the present 50% limit on core power for Units 2 and 3. Operation within these limits would continue, under the plan, for the balance of a period of approximately 45 equivalent full flow days from June 21, 1975 for Unit 2 and June 2, 1975 for Unit 3. During that time the Licensee would operate the Facility at rated flow and power for a limited time to obtain TIP traces, and to conduct accelerometer tests. Additionally, the Licensee would return the units to rated power for a short period of time to complete the fuel preconditioning program. At the end of approximately 45 effective full flow days from June 21 and June 2, respectively, the Licensee would shutdown Units 2 and 3 unless the efficacy of the 40% flow limit had been verified by accelerometer data. Finally, limits on flow and bundle power would be re-evaluated in light of data collected during the interim period under consideration.

8. The Licensee's plan has been reviewed by the NRC Staff. As discussed in the NRC Staff's Safety Evaluation Report on the matter dated July 22, 1975, and in a letter to the Licensee dated July 23, 1975, the Facilities can

be operated without endangering the health and safety of the public at the proposed core flow and maximum bundle power limits for a period of approximately 45 effective full flow days from June 21, 1975 and June 2, 1975, respectively. The operation of the Facilities at higher flow and power levels is acceptable under the same standard for certain tests, including the 75 hours of operation at rated flow and power to collect flow vibration data for Units 2 and 3, and complete a fuel pre-conditioning program for Units 2 and 3. The NRC Staff believes the Licensee's plan described above is appropriate under the circumstances and should be confirmed by NRC order.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulation in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License Nos. DPR-44 and DPR-56 are hereby amended by adding to each license the provisions set out in Appendix A hereto. IT IS FURTHER ORDERED THAT Facility Operating License No. DPR-56 is amended by deleting the provision added by the Commission's Order for Modification of License dated June 2, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Benard C. Rusche

Benard C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 23rd day of July, 1975

APPENDIX A

1. The Licensee shall not, without prior written approval of the Director, Office of Nuclear Reactor Regulation, operate the facility at a flow rate exceeding 40 percent of rated flow, or at a maximum bundle power exceeding 3.35 MWt or at a maximum average planar linear heat generation rate (MAPLHGR) exceeding 85 percent of the limit established by the Order for Modification of License dated December 27, 1974, except that:

   a. The flow limit may be exceeded, up to a maximum of 50 percent of rated flow, for such period of time as is necessary to establish the appropriate mechanical limit setpoint on the pump speed control in compliance with paragraph 2 below.

   b. The flow and bundle power limits may be exceeded by such amount and for such period of time as is necessary to establish appropriate control rod patterns after undergoing a change in power or change in flow, and to accommodate xenon buildup.

   c. The flow and bundle power limits may be exceeded by such amount and for such period of time as is necessary to perform the control rod operability requirements of Section 4.3.A.2.a of the Technical Specifications, Appendix A to Operating License Nos. DPR-44 and DPR-56.

d. The flow and bundle power limits may be exceeded by such amount and for such period of time as is necessary to establish appropriate conditions for and to perform traversing incore probe (TIP) maps for the purpose of establishing an accelerometer test program, and to establish appropriate conditions for and to conduct the accelerometer test program.

e. The operation authorized by subparagraph d. above may be extended to a total of 75 hours as necessary to allow the fuel preconditioning program to be completed.

2. The Licensee shall adjust the mechanical limits on the pump speed control to preclude an unrestricted flow increase above 50 percent for operation with flow rates limited to 40 percent of rated flow.

3. The Licensee shall shut down the facility following 45 equivalent full flow days from the "reference date", as hereinafter defined, unless within such period a traversing incore probe surveillance program as specified in Section VIII of the U.S. Nuclear Regulatory Commission's Safety Evaluation Report, In the Matter of BWR Channel Box Wear dated July 22, 1975, has been completed and accelerometer tests performed with appropriate requirements specified in Section VII of the Evaluation of Channel Box Wear dated July 22, 1975 have been completed which

indicate the efficacy of the 40 percent flow limit. The "reference date" is June 21, 1975 for Unit 2 and June 2, 1975 for Unit 3.

4. The Licensee shall inform the Director, Office of Nuclear Reactor Regulation, of the details of the proposed accelerometer test program prior to installation of the accelerometers.

5. The Licensee shall, during power operation with flow rates equal to 40 percent of rated or during operations with flow rates greater than 40 percent as allowed by paragraph 1 above, obtain unfiltered TIP traces as specified in Section VIII of the Evaluation of Channel Box Wear dated July 22, 1975, such that all operable TIP positions are traversed at least once during the course of each thirty day period and shall report the results thereof.

6. The Licensee shall submit brief analyses of TIP readings, including sample traces, to the Director, Nuclear Reactor Regulation within 30 days of obtaining such TIP readings.

7. The Licensee shall submit an analysis of the accelerometer data (a) within 10 days of obtaining the first indication of channel box impact, and (b) within 10 days of completing the accelerometer tests performed for an interval of approximately 10 effective full flow days.

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

BWR CHANNEL BOX WEAR

July 22, 1975

EVALUATION OF BWR CHANNEL BOX WEAR

I. INTRODUCTION

Fuel inspection in a foreign BWR has revealed significant wear and some cracking of several Zircaloy fuel channel boxes; significant wear has also been observed at the Duane Arnold facility in Iowa. This wear is believed to result from flow induced vibration of the instrument tubes which are located between some channel boxes in the bypass region. To reduce the vibration of the instrument tubes and to preclude further channel box wall thinning, a substantial reduction in the core flow rate is required. This report presents an evaluation of and basis for approval for the fuel element thermal limits at reduced flow. Also, an evaluation of further channel box wear at reduced flow is discussed.

The inspection of certain channel boxes has indicated that the magnitude of the wear can be correlated to the signal noise of the Traversing Incore Probe (TIP). To obtain additional confirmatory data and to determine the threshold flow rate below which significant vibration ceases, a program which obtained TIP traces at various flow rates is discussed. In addition, the need for obtaining accelerometer readings at selected locations to aid in the control of channel wear is discussed.

Reactor operation at increased flow rates (up to 100%) is permissible for short time periods to obtain data related to the TIP and accelerometer surveillance program. The basis for operation in this mode is presented.

II. OBSERVATIONS AT DUANE ARNOLD

During the June channel box inspection at the Duane Arnold reactor in excess of 70 percent of the channels adjacent to instrument tubes were inspected. The inspected channels were selected to be representative of those channels sustaining the worst corner wear. Two thirds of all the channels associated with bypass flow holes were inspected. All the channels associated with significant TIP noise* were inspected during the outage. The maximum TIP noise indication immediately preceding the outage was 8 percent with many indications at 6 percent and below. For each inspected channel, the width of the wear marks were measured and the depth of the wear was conservatively inferred. Neither through-wall wear nor a crack was reported on any of the channels. Inspection of the most severly worn channel is continuing.

---

*TIP Noise is defined as the ratio of fluctuations in the TIP signal as traced on its x-y recorder, in the frequency range of interest (generally 1-3 Hz), divided by the mean value of the signal.

III. REDUCTION OF VIBRATION WITH FLOW

The data available from the inspection of channel box wear in the Duane Arnold reactor has been correlated with the associated TIP noise. To date, the TIP noise is still the best indicator of the magnitude of channel box wear. In other words, as the channel box corners are worn, there is more room for the TIP to move and as a result, there is a greater change in the TIP signal. It has been observed that the forced amplitude is more than linearly dependent upon the flow rate (generally, the forced amplitude varies with the flow rate squared). Thus, the flow rate can be used as the parameter to control the vibrational amplitude of the TIP tube for the purpose of essentially eliminating further channel box corner wear.

General Electric has experimentally confirmed the reduction in vibrational amplitude with decreasing flow specifically for their prototypical TIP tube-channel box configurations. Their test facility employed accelerometers attached to the instrumentation tube. The impact signals from these sensors diminished significantly with decreasing flow, and were observed to cease below a threshold flow rate of approximately 45 percent of full flow.

As mentioned previously, there were pre-shutdown measurements at the Duane Arnold reactor. The TIP noise levels were observed at several coolant flow rates. These have been correlated and show the characteristic reduction in TIP noise with decreasing flow rate. Separate calculations were made to determine the minimum TIP noise level to be expected for the Duane Arnold reactor independent of flow rate. This minimum calculated noise level corresponds to the maximum observed TIP noise level at a coolant flow rate of 40 percent of full flow.

Finally, TIP traces were taken at the Cooper reactor at several levels of power and flow. It was determined that no vibrational noise in a TIP trace was observed at 43 percent flow while the same TIP exhibited noise at both 69 percent flow and greater flows.

From the above discussion, we conclude that channel box wear due to TIP tube vibration is significantly attenuated for sustained operation at 40 percent of the rated coolant rate. Although instrument tube wear on the channel box could cause penetration of channel corners if operation at rated power and flow rate continued for extended periods of time. Wall penetration requires extended periods of wearing; it is a long term phenomenon, and rapid penetration of the channel box corners should not be expected if operation is limited to lower flow rates.

Thus, sustained operations should be conducted at or below 40 percent flow. Short operational periods at coolant velocities approaching rated flow rate can not be expected to cause rapid through-wall penetration. However, additional surveillance must be associated with these short periods of higher flow operations, since incremental wear may occur.

The restriction of sustained operations at 40 percent flow, once a TIP trace exceeds the 6% criterion is designed to preclude gross channel box penetration and the undesirable side effects associated with wear-through, such as loose pieces of Zircaloy, or fuel rod wear-through.

IV. THERMAL EVALUATION AT REDUCED FLOW

General Electric presented in a generic report[1] a procedure for determining acceptable operating limits for those BWR-4's that may have cracked channel boxes.* The major assumptions used in this study are that the highest powered assembly in the core has a channel box with a crack of specified size and location, and that the core flow is 40 percent of the design value. Analyses were performed to determine the flow through the bundle as a function of elevation and subsequently the critical power in the affected assembly based on the GEXL correlation[2]. The thermal limits are to be based on these calculations and specific minimum critical power ratios (MCPR) for each plant that were previously submitted to NRC by the utilities.

The hydraulic analysis[1] examined an assembly with a cracked channel in a core of "uncracked" assemblies with a core flow of 40 percent and power of 60 percent. Calculated flow rates in the lower portion of the "cracked" assembly were ∿20 percent higher than those in a similar "uncracked" assembly, while the flows in the upper portion of the assembly above the crack were ∿20 percent lower. It was noted that these results were independent of crack sizes down to about 3 square inches. These calculations were based on nominal hydraulic resistance correlations, however, the inlet enthalpy was conservatively based on design power conditions.

---

*It is our present opinion that it is very unlikely that these facilities have through-wall penetration; this assumption was postulated due to the lack of fully definitive wear predictions.

The calculated assembly flow rate was used to determine the boiling length-quality relationship that is necessary for evaluating the assembly critical power from the GEXL correlation[2]. A symmetrical axial power shape was assumed with a peak of 1.4, which is conservative for a "cracked" channel that would have more voids and lower power in the upper portion of the assembly above the crack. The calculated critical power for a "cracked" assembly is 4.6 MW for a reactor at 40 percent flow. This value is about 10 percent lower than that for a similar "uncracked" assembly and about 10 percent higher than for an assembly with a crack along its entire length.

The ratio (MCPR) of assembly critical power determined as outlined above and the bundle allowable power is based on analyses of the abnormal operating transients for a particular plant. Operating limit MCPR's, previously determined for the plants based on design power conditions, are composed of a safety limit MCPR and a ΔMCPR.

The safety limit MCPR is a statistical combination of the uncertainties in plant operating conditions, manufacturing tolerances, and uncertainties in the GEXL correlation. The presence of cracked channels would increase the uncertainties and the safety limit over that for a core with whole channels. However, this perturbation is compensated by the assumption that the cracked channel occurs in the highest powered assembly and is not treated in a statistical fashion in the overall thermal margin determination.

The ΔMCPRs are based on the largest change in thermal margin calculated for the abnormal operating transients (turbine trip without bypass, loss of feedwater heater, etc.) initiated from design power conditions. Previous analyses with uncracked channels have shown a significant reduction in the ΔMCPR when the transient is initiated from reduced flow and power conditions. The effect of the "cracked" channel on the MCPR is judged to be less than the effect of reduced power level. As a result, the application of ΔMCPRs based on full power operation is considered conservative when applied to operation at reduced flow and power operation.

A transient associated with reduced flow operation is an unrestrained flow-power increase. Under these conditions, the bundle power increases more rapidly than the critical assembly power resulting in a reduction in thermal margin. Correction factors (K_f) have been derived for partial flow operation in normal operating reactors. These correction factors compensate for possible unrestrained flow-power increases by increasing the operating limit MCPR at reduced flow conditions. It has been proposed to adjust the mechanical limits on the pump speed control to preclude an unrestrained flow increase above 50 percent when operating at 40 percent core flow with assumed "cracked" channels. Under these restraints, the maximum core power increase would be about 10 percent for an unrestrained flow increase of 10 percent, while the critical power

increase would be about 5 percent. It has been proposed conservatively that this possible transient be compensated by using a K_f factor of 1.15 which is based on a flow increase from 40 to 102.5 percent with "uncracked" assemblies.

Bundle power was proposed in Reference 1 as the thermal limit for reactor operation with assumed "cracked" assemblies. The bundle power limit would be established by reducing the critical power, calculated for "cracked" assemblies at reduced flow, by the operating limit MCPR for a plant, the flow correction factor, K_f, of 1.15, and allowing for the fraction of energy deposited in the channel. The staff has concluded that this definition of bundle power will result in acceptable thermal margins for reactors so affected.

A conservative LOCA evaluation was performed in Reference 1 that assumed immediate loss of nucleate boiling and subsequent pool boiling until core uncovery. The results indicated a maximum reduction of 15 percent in the Appendix K MAPLHGR limits. These results are not limiting because the reactor will be operating approximately 40 percent below full power conditions.

Our regulations concerning thermal limits during steady state and transient operation (MCPR) and accident (MAPLHGR) are thus satisfied during operation at 40 percent flow, and the associated bundle power limit.

V. POWER OPERATION ABOVE THE BUNDLE POWER LIMIT

The bundle power limits described elsewhere in this report are based on the conservative assumption that a leakage area exists in the side of a channel box, with associated coolant diversion and thermal penalty.

We believe that these bundle power limits may be exceeded for short periods of time, for good cause, on the basis that (1) most likely, cracks do not yet exist in the channel boxes, (2) wear rate is relatively slow, even at 100 percent flow, (3) the likelihood of a thermally-challenging transient during brief periods is low, and (4) extra surveillance of core conditions will be employed.

Specifically, we approve (1) operating at 100 percent flow and power for a time sufficient to gather additional incore information to guide the placement of excore vibration detection devices (e.g., accelerometers) and (2) additional operating capability up to 100 percent flow and power for the purposes of calibrations and sensitivity studies with the devices mentioned in Part (1) above.

At this time, we can not generally approve operating above the bundle power limit for other purposes, such as critical need for power for brief periods. However, any request for such flexibility would be considered in the context of the arguments mentioned above.

VI. STARTUP

The Technical Specifications presently define the Average Power Range Monitor (APRM) flow biased scram setpoint and rod block setpoint. Proposed operation, at core flow of 40 percent rated and core power of approximately 60 percent of rated, may result in rod blocks during reactor startup in trying to achieve the proper control rod pattern. To eliminate these rod blocks while maintaining the current Technical Specification limits, the core flow may be increased so that the rod pattern may be ajusted to accommodate xenon buildup and redistribution, and system related flow perturbations. As the proper control rod pattern is established and the system power is stabilized, the reactor core flow must be returned to 40 percent and the power to about 60 percent.

VII. SURVEILLANCE WITH ACCELEROMETERS

GE demonstrated that when there is a significant instrument tube-channel box impact both strain gages and accelerometers can detect the signal. In their flow test facility at San Jose, the LPRM tube was instrumented with both strain gages and accelerometers along the length of the tube. The flow-induced impact signals were distinguishable from the background noise. A marked reduction in g-level was also observed when flow was reduced. We were also informed that at certain flow rates, the impact signal ceased. Thus, we believe that the installation of accelerometers at each BWR should be required if the plant is to operate beyond 45 EFFD (effective full flow days), or above 40% flow or both. The accelerometers should be employed to establish a threshold flow rate below which significant impact ceases. The flow rate may be plant dependent, and may be generally higher than 40 percent flow. We believe that continued surveillance is desirable to guard against any anomaly after a reduction in flow.

We recommend that the number of accelerometers be approximately eight or more. At least the following types of incore monitoring tubes should be instrumented:

	TYPE	QUANTITY	NO. OF BYPASS HOLES	COMMENTS
a.	LPRM	2-4	Any	Tubes with Noisy TIP Signal
b.	LPRM	2	3	-
c.	LPRM	1	2	Asymmetric
d.	LPRM	1	4	-
e.	SRM & IRM	2-0		Desirable but not necessary

Location of tubes to be instrumented may be selected at random throughout the core except as (a) above.

Approximately ten effective full flow days (EFFD) is recommended as a minimum interval for continuous surveillance of the accelerometer response at the beginning of operations with the accelerometers. During periods where the flow rate is in excess of that established in the threshold for impact, data shall be collected from the accelerometers at least daily. Periodic analyses should be made and forwarded to NRC.

VIII. SURVEILLANCE WITH TIP SUBSYSTEM

Added surveillance is necessary when significant channel box wear is suspected to occur at a plant. The Traversing Incore Probe (TIP) subsystem and other subsystems will be required to perform this surveillance. An unfiltered recording of the TIP reading has been shown, by observations at a domestic plant, able to provide an indication of those locations where the vibration of the instrument tube is causing potentially severe channel box wear. In particular, it has been observed at the domestic plant that the approximately 2.5 Hz frequency of vibration of the instrument tube is superimposed as noise content on those TIP traces for which channel box wear was observed. This approximately 2.5 Hz noise content of the TIP traces has been shown to be strongly correlated to the observed wear, i.e., the greater the measured channel box corner wear the greater the TIP noise content.

Measurements made with the TIP neutron detectors can be made at any combination of reactor power and core flow rate. At lower reactor power and flow rate, it is expected that the force of the instrument tube vibration will be reduced and, correspondingly, channel box corner wear. Thus, the TIP traces will show a minimum noise content due to instrument tube vibration. Added measurements will be necessary for power operation above the bundle power limit (See Section V). Surveillance at both the lower flow rate and power level and higher flow rate and

and power level will provide an indication of no further or lessened degradation of the channel boxes.

The surveillance program with the TIP subsystem is listed below:

1. Each TIP unit should be calibrated by insertion of a 0.1 Hz sinewave into the TIP flux amplifier, of sufficient amplitude to product a 10 percent peak-to-peak oscillation about the 50 percent level on the TIP recorder chart. Without changing input amplitude, the calibration should be repeated at frequencies of approximately 0.5, 1, 2.5, 3, 4, and 5 Hz. The full-scale calibration current (gain factor) should be recorded for a TIP amplifier. The response correction factor should be obtained by dividing the recorder trace amplitude at 0.1 Hz by the width at 2.5 Hz.* If the TIP amplifier is recalibrated such that the calibration current of the TIP amplifier is changed by more than 20 percent, the TIP frequency should be repeated and a new response calibration factor should be computed.

---

*Since the most complete set of wear measurements exist for the Duane Arnold facility, TIP readings at other facilities will be normalized to Duane Arnold.

2. During power operation, unfiltered TIP traces should be obtained such that all operable TIP positions are traversed at least once during the course of each thirty day period.

   (a) For the purpose of obtaining TIP traces, power operation is defined as operation for core flow rates equal to or greater than 40 percent of rated.

   (b) The TIP traces for a given core flow rate and power should all be done over a period of not more than eight hours of elapsed time.

   (c) If, during any thirty day reporting period, the core flow rate is changed by more than 15 percent of rated flow, TIP traces should be taken at these different flow rates.

   (d) Each TIP trace should be clearly marked with the plant name, date, TIP machine used, TIP speed, flow, reactor power, control notches for adjacent rods, and estimated noise content (see Step 3).

3. The noise bandwidth (peak-to-peak) to signal amplitude ratio should be estimated for the noisiest 10 inch section of axial fuel length of each TIP trace; the estimated noise bandwidth to signal amplitude ratio should be done for noise content with a 1 to 3 Hz range.

4. If the normalized (to Duane Arnold) 100 percent rated core flow and reactor power noise bandwidth to signal amplitude ratio exceeds 0.08, or has increased by more than 50 percent from a previous observation, for a given 10 inch fuel section, the NRC staff should be immediately notified.

   (a) The normalized noise bandwidth to signal amplitude ratio is defined as the estimated ratio (Step 3) times the 2.5 Hz response correction factor (Step 1) divided by 1.3 (the Duane Arnold response factor).

   (b) If the reactor core flow rate is less than 100 percent of rated, the normalized noise bandwidth to signal amplitude ratio should be extrapolated to the 100 percent flow condition by adding 0.04 times the quantity 100 minus the core flow rate expressed as a percentage.

5. The TIP traces and the results obtained from the TIP traces should be filed at the plant and be available for inspection. The noisiest TIP traces should be periodically forwarded to NRC, with a brief analysis of the noise calculations.

IX. CONCLUSIONS

The NRC staff concludes that plants operating at reduced flow due to greater than 6 percent TIP noise will not present an undue risk to the health and safety of the public provided that:

1) The bundle power limits are calculated in accordance with Reference 1 and discussed above.

2) The mechanical limits on the pump speed control be adjusted to preclude transient operation significantly above the reduced flow limit.

3) The MAPLHGR limits currently in force be reduced by 15 percent.

4) Surveillance with accelerometers and TIP detectors is accomplished.

It is further concluded that operation at higher flow rates for short periods of time is acceptable for surveillance purposes, and as necessary during cold start prior to xenon equilibrium.

REFERENCES

1. Letter from R. Engel (General Electric) to V. Stello (NRC), dated July 11, 1975.

2. General Electric BWR Thermal Basis (GETAB): Data, Correlation, and Design Application, NEDO-10958 (November 1973).