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May 21, 1975

Docket No. 50-331

Iowa Electric Light & Power Company
ATTN: Mr. Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" issued by the Commission on May 21, 1975, for the Duane Arnold Energy Center. The Order added a provision to License No. DPR-49 stating that you shall not operate the facility at core power levels exceeding 50% of rated power or core flow rates exceeding 50% of design flow rate without prior written approval by the Director, Office of Nuclear Reactor Regulation. A copy of the Order is being filed with the Office of the Federal Register for publication.

Also enclosed for your information are (1) a copy of the Safety Evaluation issued by us on October 26, 1973 for the Vermont Yankee and Pilgrim facilities, (2) a copy of a General Electric Company letter dated April 22, 1975, and (3) a copy of the Order for Modification of License for the Nebraska facility which are referenced in the subject Order.

Sincerely,

/S/

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

Enclosures:
1. Order for Modification of License dtd May 21, 1975
2. Safety Evaluation dtd October 26, 1973 re Vermont Yankee and Pilgrim facilities
3. GE ltr dtd April 22, 1975
4. Order for Modification for Nebraska facility dtd April 26, 1975

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(3)

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

ORDER FOR MODIFICATION OF LICENSE

I.

Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (Licensees) are the holders of Facility Operating License No. DPR-49 which authorizes operation of the Duane Arnold Energy Center (the Facility) at steady-state reactor core power levels not in excess of 1658 megawatts thermal (rated power). The Facility is a boiling water reactor (BWR) located at the Licensees' site near Palo in Linn County, Iowa.

II.

1. By telephone calls on April 17, 18 and 21, 1975, and by letter of April 22, 1975, the NRC Staff was informed by the General Electric Company (GE) that fuel inspections in a foreign BWR had revealed significant wear and some cracking of several zircaloy fuel channel boxes. The discovery of this damage had been preceded by anomalous

signals of certain incore neutron detector instruments. The Staff has tentatively concluded that damage to the fuel element channel boxes and the anomalous instrument readings may be due to vibration of the instrument tubes which may be produced by coolant flow through bypass holes in the reactor lower core support plate. To determine whether or not any BWRs within the United States could be subject to similar damage, GE notified the operators of eleven United States plants having a similar design that have utilized bypass holes in the reactor lower core support plate and asked that they inspect the incore neutron detector instrument readings from the traversing incore probe (TIP).

2. GE was informed, and subsequently informed the NRC Staff at a meeting held on April 24, 1975, that anomalous readings, similar to those experienced in the foreign reactor, were found at the Cooper Nuclear Station (the Cooper Station) operated under Facility Operating License No. DPR-46 by the Nebraska Public Power District (NPPD). Such anomalous readings are indicative of possible damage or cracking of the channel box. With this indication of possible damage existing in the core, further operation under similar conditions could lead to further core damage, and if continued, could result in an unsafe condition. In view

of the foregoing, NPPD agreed on or about April 26, 1975 that it would not operate the Cooper Station at core power levels exceeding 50% of rated power or core flow rates exceeding 50% of design flow rate without prior written approval of the NRC Staff. The NRC Staff confirmed this agreement by an Order dated April 26, 1975.[1/]

3. The NRC Staff, by letter dated April 26, 1975, also informed NPPD and the operators of the other ten nuclear power reactors referred to in paragraph 1 above that, pursuant to 10 CFR §50.54(f), they were required to take specific action to assess the potential for damage to the fuel element channel boxes in each reactor and report the results to the NRC Staff. Specifically, the letter required the Licensees of the eleven nuclear reactors to: "(1) Review all the results from TIP or LPRM monitoring that has been performed within the last three months to identify any anomalous behavior, and report the findings within 7 days, (2) During power operation of your reactor, following receipt of this letter, and at intervals not to exceed one day, obtain unfiltered TIP monitoring traces such that during the course of each seven-day period all operable TIP positions are traversed at least once, (3) If

---

1/ 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.

special studies or tests of neutron flux behavior other than that described in paragraphs 1, 2 are being conducted with the LPRM subsystem or the TIP subsystem, provide the results to NRC within ten days of the initiation of the tests or studies, (4) If in your examination of the TIP monitoring traces required in 1 or 2 above, you observe that the ratio of noise band width to signal amplitude, for signals of frequency greater than one Hz, exceeds 0.06 over any ten inches of axial core length, inform NRC immediately by telephone, (5) For successive TIP traces at the same instrument tube location, if the noise band width changes by more than 50% peak to peak over an axial core length of one foot, the NRC shall be informed immediately by telephone."

4. On May 6, 1975 the Licensees informed the NRC Staff that prior to a shutdown for maintenance work on May 5, 1975, the TIP monitoring traces from the Duane Arnold Energy Center (Facility) had displayed a ratio of noise level width to signal amplitude exceeding 0.06 for signals of frequency greater than one Hz. During the slow power ascension following the shutdown, the Licensees performed tests to monitor the TIP noise and conducted special tests to determine the signal characteristics of the Low Power Range Monitors (LPRMs). These tests showed that at core flow rates of approximately 55%, the anomalies in the TIP traces were no longer observed, substantiating the belief derived from earlier

experience[2/] that, at flows of approximately 50% of full core flow, the flow through the bypass holes were reduced sufficiently to substantially reduce excessive vibration of the instrument thimbles located in the bypass region. This, in turn, should reduce further channel box damage. On May 17, 1975, the Licensees notified the NRC Staff by telephone that the reactor had reached 100% flow and 96% power and that a TIP trace taken on May 17, 1975 had displayed a ratio of noise band width to signal amplitude of 0.0625. On October 8, 1974 a similar TIP trace taken in the same location indicated no noise, i.e., displayed no significant ratio of noise band width to signal amplitude. This indicates that in the interim period noise at the Facility increased significantly.

2/ Tests to monitor TIP noise and to determine the signal characteristics of the LPRMs were also performed during April 1975 at the Cooper Station. These tests are similar to those performed by the Licensees and the results are consistent. In addition, in late 1973 and early 1974, channel box damage due to control curtain vibration was observed in the Vermont Yankee Nuclear Power Station, in the Pilgrim Nuclear Power Station, and in a foreign reactor. In those instances, excessive vibrations of certain temporary control curtains located in the bypass region between the channel boxes had caused the channel box damage. In connection with the Vermont Yankee and Pilgrim facilities, tests at a GE flowtest facility demonstrated that the vibration of the control curtains in the bypass region resulting from flow through bypass flow holes was substantially reduced at flow conditions of approximately 50% full flow. See Safety Evaluation by the Directorate 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. It should be noted that the bypass holes in the Pilgrim Nuclear Station were plugged in August 1974.

5. The Licensees' May 17 report of the anomalous TIP trace reading exceeded the 0.06 criterion established by the NRC Staff in its letter of April 26. Moreover, the May 17 anomalous TIP trace reading is similar to those experienced at the foreign reactor and at the Cooper Station. As indicated above, such anomalous readings are indicative of possible damage or cracking of the channel box. With this indication of possible damage existing in the core, further operation of the Facility under similar conditions could lead to further core damage, and if continued, may result in an unsafe condition. As indicated above, further tests are being conducted by GE, and additional studies and investigations by GE and others are underway. Nonetheless, as an interim measure, in order to minimize the possibility of any further damage to the fuel channel boxes at the Facility, the NRC Staff has concluded that core flow should not be allowed to exceed 50% of design flow. This, in turn, will require facility power to be reduced to approximately 50% of full power. Operation at this power level will provide assurance of facility safety while further data are being developed.

6. The Licensees have agreed that they will not operate the facility at core power levels exceeding 50% of rated power or core flow rates

exceeding 50% of design flow rate without prior written approval of the NRC Staff. The NRC Staff believes that the Licensees' action, under the circumstances, is appropriate and that this action should be confirmed by NRC Order.

7. Copies of the following documents are available for public inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., 20555 and are being placed in the Commission's Local Public Document Room, Reference Service, Cedar Rapids Public Library, 426 Third Avenue, S.E., Cedar Rapids, Iowa: (1) Safety Evaluation by the Directorate 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; (2) Letter dated April 22, 1975, from the General Electric Company to the NRC Staff; (3) Order for Modification of License, In the Matter of Nebraska Public Power District (Cooper Nuclear Station), Docket No. 50-298, dated April 26, 1975. Copies of the letter dated April 26, 1975 from the NRC Staff to the Iowa Electric Light and Power Company are available for public inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., 20555, and the Commission's Local Public Document Room, Reference Service, Cedar Rapids Public Library, 426 Third Avenue,

S.E., Cedar Rapids, Iowa. In addition, copies of the Traversing Incore Probe (TIP) traces submitted to the NRC Staff by the Licensees, including the TIP traces dated May 16-17, 1975, are being placed in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., 20555 and in the Commission's Local Public Document Room, Reference Service, Cedar Rapids Public Library, 426 Third Avenue, S.E., Cedar Rapids, Iowa.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commisson's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-49 is hereby amended by adding the following new provision:

> By reason of the circumstances outlined in the Order for Modification of License, dated May 21, 1975, the Licensees shall not operate the Facility at core power levels exceeding 50% of rated power or core flow rates exceeding 50% of design flow rate without prior written approval of the Director, Office of Nuclear Reactor Regulation.

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 21st day of May, 1975.

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

RELATING TO

CHANNEL BOX WEAR

IN THE

VERMONT YANKEE NUCLEAR POWER STATION
(Docket 50-271)

AND THE

PILGRIM NUCLEAR POWER STATION
(Docket 50-293)

October 26, 1973

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1.0 INTRODUCTION

The fuel bundles of boiling water reactors are encased in square Zircaloy-4 tubes known as channel boxes (Fig. 1) which separate parallel coolant flow paths in the core, provide guide surfaces for the motion of control rods, and protect fuel bundles during handling. Each channel box slides over the surface (lower tie plate surface, Fig. 1) of a fitting which rests on the core support plate. The zone between the channel box assemblies is referred to as the bypass region and, in all boiling water reactors, contains the control rods and in-core instrumentation. In some facilities, the bypass region also contains temporary control curtains (Fig. 2), which are removed after initial operation of the first core and are used to provide supplemental control.

The principal portion of the coolant flow is directed through the channel box assemblies. In earlier designs, which includes most of the operating boiling water reactors, leakage between the lower tie plate and the channel box is the only source of bypass flow. However, in other facilities, this leakage has been restricted and additional bypass flow is provided by "bypass flow holes" in the core support plate.

Figure 3 depicts a cross-section of a fuel cell containing all of the elements discussed above. This particular arrangement, combining bypass flow holes with temporary control curtains, is found

FIGURE 1

FUEL CHANNEL BOX

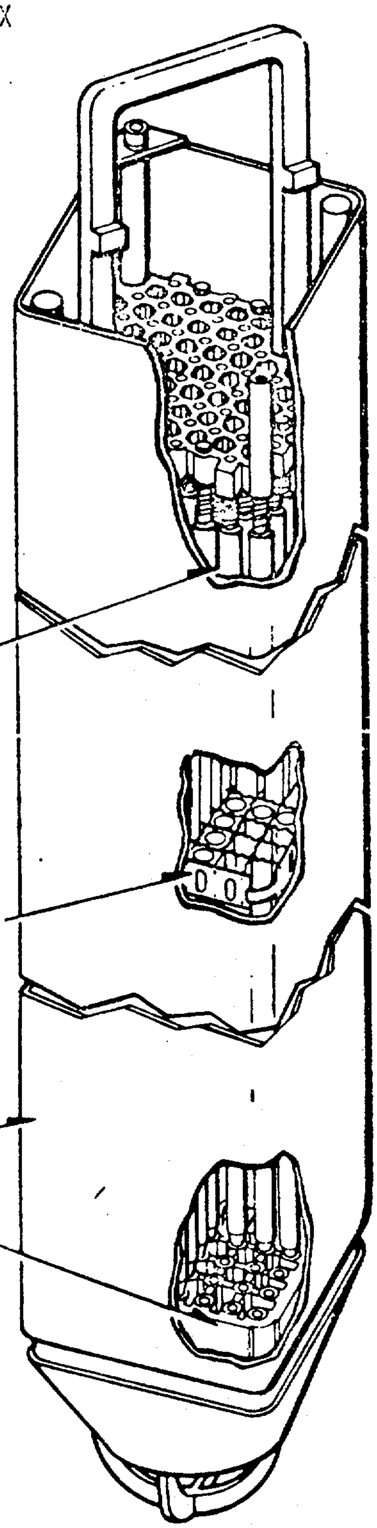


FIGURE 2

TEMPORARY CONTROL CURTAIN

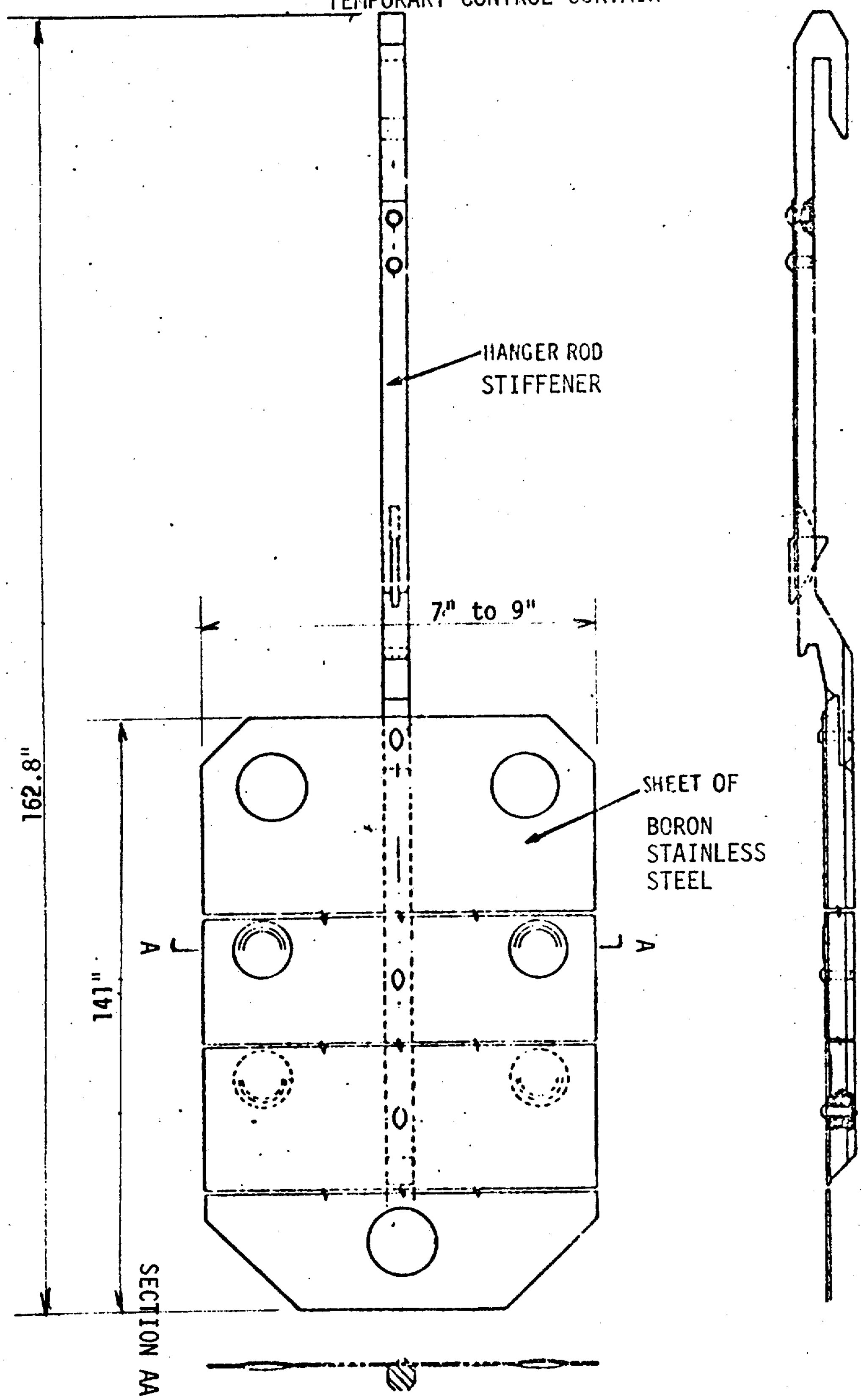
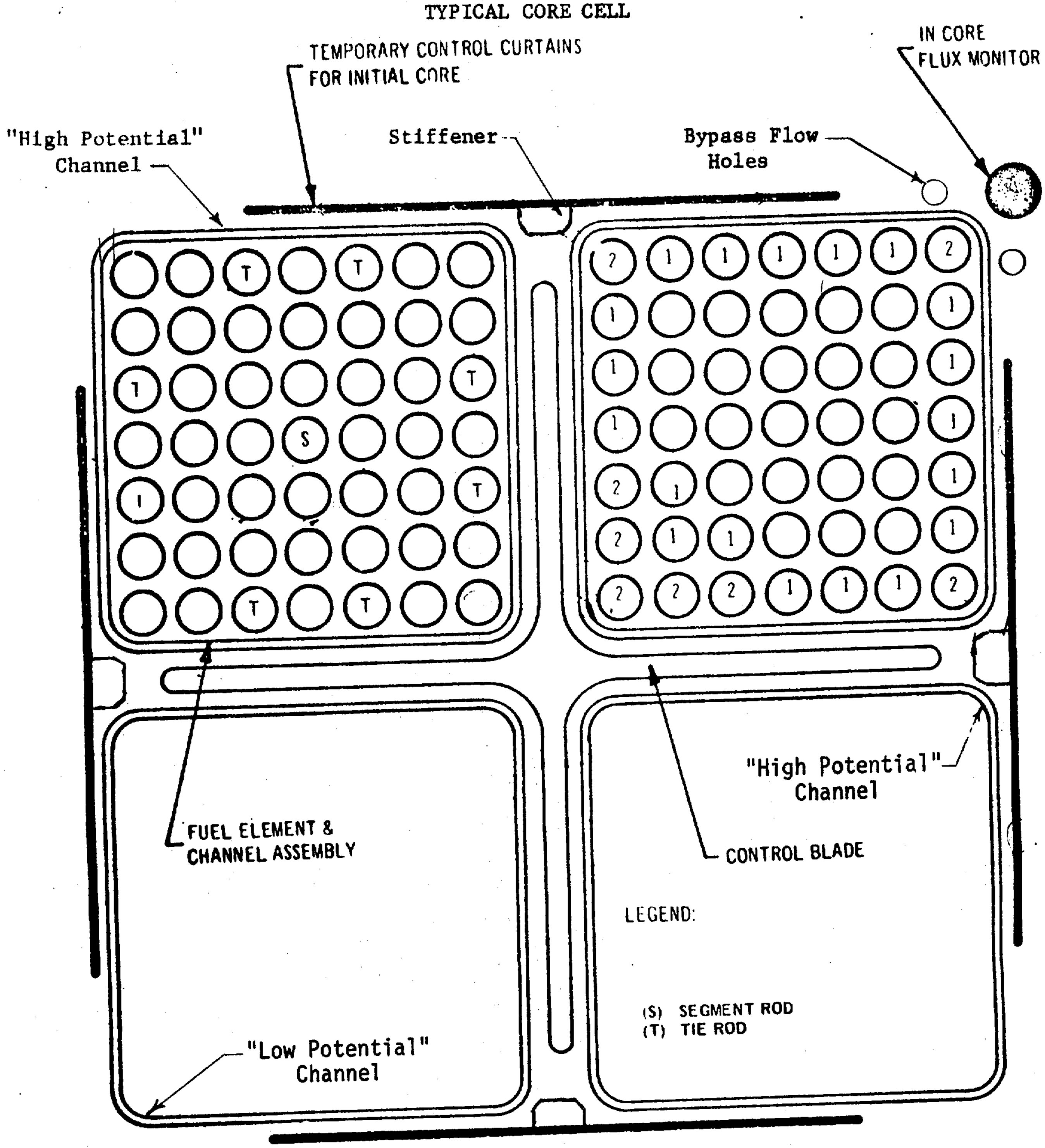


FIGURE 3

TYPICAL CORE CELL



in only two operating U.S. reactors, the Vermont Yankee Nuclear Power Station and the Pilgrim Nuclear Power Station. The KKM reactor in Switzerland is the only other facility which contains both of these design features.

In August 1973, extensive wear on the corners of some fuel assembly channel boxes was observed during an inspection of the KKM reactor (Ref. 1). Similar wear has recently been observed in Vermont Yankee. The wear is confined in all cases to the lower part of the affected channel boxes. In some instances the wear has created slots up to one foot in length and more, and up to 1/2 inch wide. For reasons discussed below, the observed channel box wear and cracks have been determined to be the result of vibration induced in temporary control curtains by the flow through the bypass flow holes. Accordingly, similar wear is presumed to exist in the Pilgrim reactor.

The AEC Regulatory staff and the licensees of the Vermont Yankee and Pilgrim facilities have been investigating the cause, consequences and possible correction of this condition. Vermont Yankee is not now operating and will not operate while this matter is under consideration. The facility was shut down on September 28, 1973 for installation of an augmented off-gas system and for other reasons unrelated to fuel channel box damage. On October 16, 1973, the licensee represented to the AEC Regulatory staff that Vermont Yankee would remain shut down until the fuel channel box damage had been repaired and the cause of the damage corrected (Ref. 2).

The Pilgrim reactor is now operating subject to stringent, 50% reductions in core flow rate and core power. These limitations were first adopted by the licensee on the recommendation of the General Electric Company on October 5, 1973, and then imposed by the AEC Regulatory staff, in a letter dated October 16, 1973 (Ref. 3), on the condition that this mode of operation be continued only for a limited period of about sixty days. The purpose of this Safety Evaluation is to set forth the considerations underlying the staff's approval of such interim operation of Pilgrim.

The matter of channel box wear is also the subject of a "Joint Petition for Immediate and Indefinite Shutdown of Vermont Yankee Nuclear Power Station and Pilgrim Nuclear Power Station," (Ref. 4) which was filed with the Commission on October 15, 1973 and supplemented by an affidavit filed on October 19, 1973 (Ref. 5). By order dated October 23, 1973, (Ref. 6) the Commission treated the petition as a request for the issuance of an order to show cause pursuant to 10 CFR §2.202 and instructed the Director of Regulation to, among other things, determine whether further action, including any shutdown, is appropriate as an emergency matter, and to announce that determination also, together with supporting reasons. This Safety Evaluation sets forth the staff's reasons for concluding, in view of the limitations now in effect for the Pilgrim facility, that there is no need at this time for further action, such as a shutdown, as an emergency matter.

2.0 BACKGROUND

2.1 Observations

2.1.1 KKM Reactor

Extensive wear on fuel assembly channel boxes was first observed during a routine shutdown of the KKM boiling water reactor at Mühleberg, Switzerland in August, 1973. Fuel assemblies were also removed from the core and placed in the spent fuel storage pool at that time. Routine inspection of the fuel assemblies revealed wear of varying degrees at different locations. Of the 228 channels in the core, the results of inspections of 210 channels are available. In general, minor wear or burnishing was seen on the sides of 73 channels which is attributed to rubbing by the blade of the control curtain or the control rod roller. Burnishing of the corners of 13 channels opposite the location of the incore flux monitors was also observed.

Wear on the lower portion of one corner of 43 channels was reported. This wear varied from minor to extensive. In eight of the 43 cases the channel wall was worn through, resulting in a narrow vertical slot. These slots varied in length from less than one inch to 1-1/2 feet long and were up to 1/2 inch wide. Additional wear was evident above these slots and in some cases extended as far as five feet up the corner. Some channels had tight horizontal cracks up to three inches long at the top and bottom of the slots. Pieces of up to three square inches in area were missing from two channels.

Four of the fuel assemblies with worn-through channels were inspected for fuel rod damage. Although some minor wear on the fuel rod spacer grid was seen, no damage in the form of either wear or overheating was seen. One assembly, which had clad perforation as indicated by leakage of fission products, was inspected. One rod in the interior of this bundle had splits in the clad which are characteristic of pellet-clad interaction. The channel of this assembly had no visible wear.

2.1.2 Vermont Yankee

In February 1973, the Vermont Yankee (VY) reactor was shutdown for refueling. Fuel assemblies were removed from the core and visually inspected in a manner similar to the procedure which was used during the initial phase of the refueling at KKM. No fuel assembly channel wear was reported.

The Vermont Yankee reactor is now undergoing a scheduled six week shutdown which began on September 27, 1973. Although one purpose of the shutdown was to reconstitute and relocate fuel assemblies and remove some of the temporary control curtains, the experience at the KKM reactor resulted in a program to inspect the fuel assembly channel boxes. The extent and degree of channel wear and damage observed in the Vermont Yankee core is consistent with that observed in the KKM reactor. Since the cause of the wear, as discussed in the next sections had been determined, only those channels with the potential for wear and a small sampling of the remaining channels have been inspected. To date, 133 of the 368 channels in the core have been inspected. Of the channels that were judged not to have a potential for wear, no wear was observed. Of the channels judged to have the potential for wear, 63 had evidence of wear to varying degrees. Of these, 24 had been worn through at one corner. The extent of wear as indicated by the size of the slots and wear marks was similar to that observed in the KKM reactor. Tight, horizontal cracks emanating from the top and bottom of the slots were also observed. In no case were pieces of channel wall missing.

2.1.3 Other Reactors

A continuing program of inspection similar to that employed at KKM, has been in effect at other boiling water reactors since 1962. These reactors are similar in design to Vermont Yankee, Pilgrim and KKM in that they have temporary control curtains, but they do not have bypass flow holes in the core support plate. Over 5500 channels have been visually inspected. Detailed inspections of 51 channels from four reactors have also been made. Although one channel with a crack of undetermined origin was found, no significant wear was observed on any channel.

2.2 Cause

On the basis of the observation of the damaged channels at Vermont Yankee and KKM and the results of simulation tests at a

General Electric facility, the wear observed on the corners of channel boxes is attributable to vibration of the temporary control curtains. The high velocity flow exiting from the bypass flow holes in the core support plate impinges on the control curtain blade causing the curtain to vibrate. Asymmetric displacement of the blade by the flow causes the control curtain stiffener to contact the corner of an adjacent channel. The continued vibration of the stainless steel stiffener in contact with the channel causes wear and fretting of the Zircaloy channel.

In both the VY and KKM reactors severe wear or breaching of the channel corner was observed on only those channel corners that were adjacent to the stiffeners on those curtains that were themselves adjacent to bypass flow holes. In addition, the severely worn corners were always adjacent to the side of the stiffener farthest from the bypass flow holes. The corner of channels adjacent to stiffeners but on the side nearest the bypass holes exhibited only minor wear. Therefore, the observed wear is entirely consistent with the previously stated mechanism.

The configuration of control curtains, bypass flow holes and fuel assemblies has been simulated by General Electric in a test facility (Figure 4). The facility consists of twelve fuel assembly channels, four temporary control curtains, and a support plate with four, one-inch-diameter flow bypass holes (Figure 5). A pump and piping provide

Figure 4

GE FLOW TEST FACILITY ELEVATION

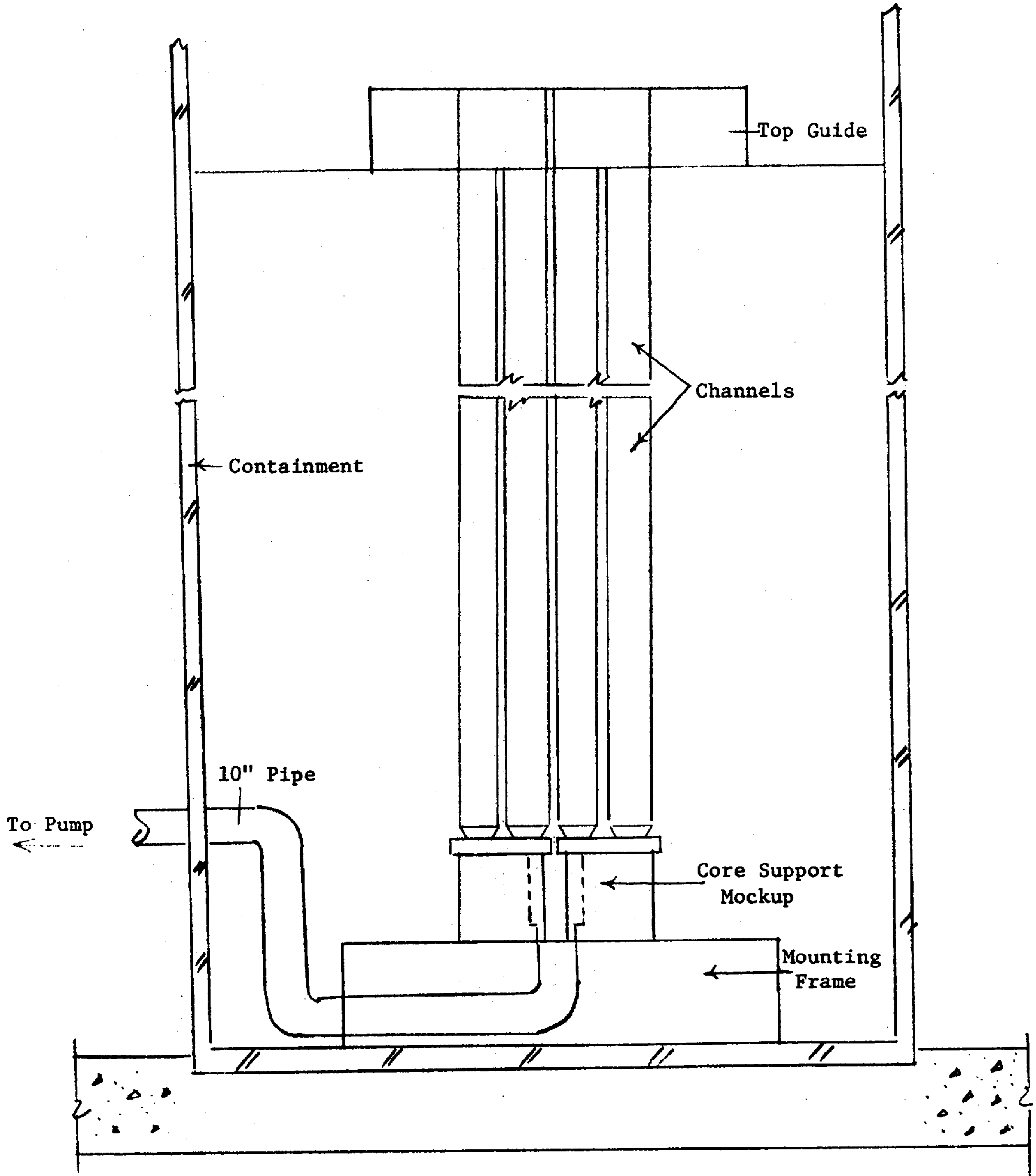
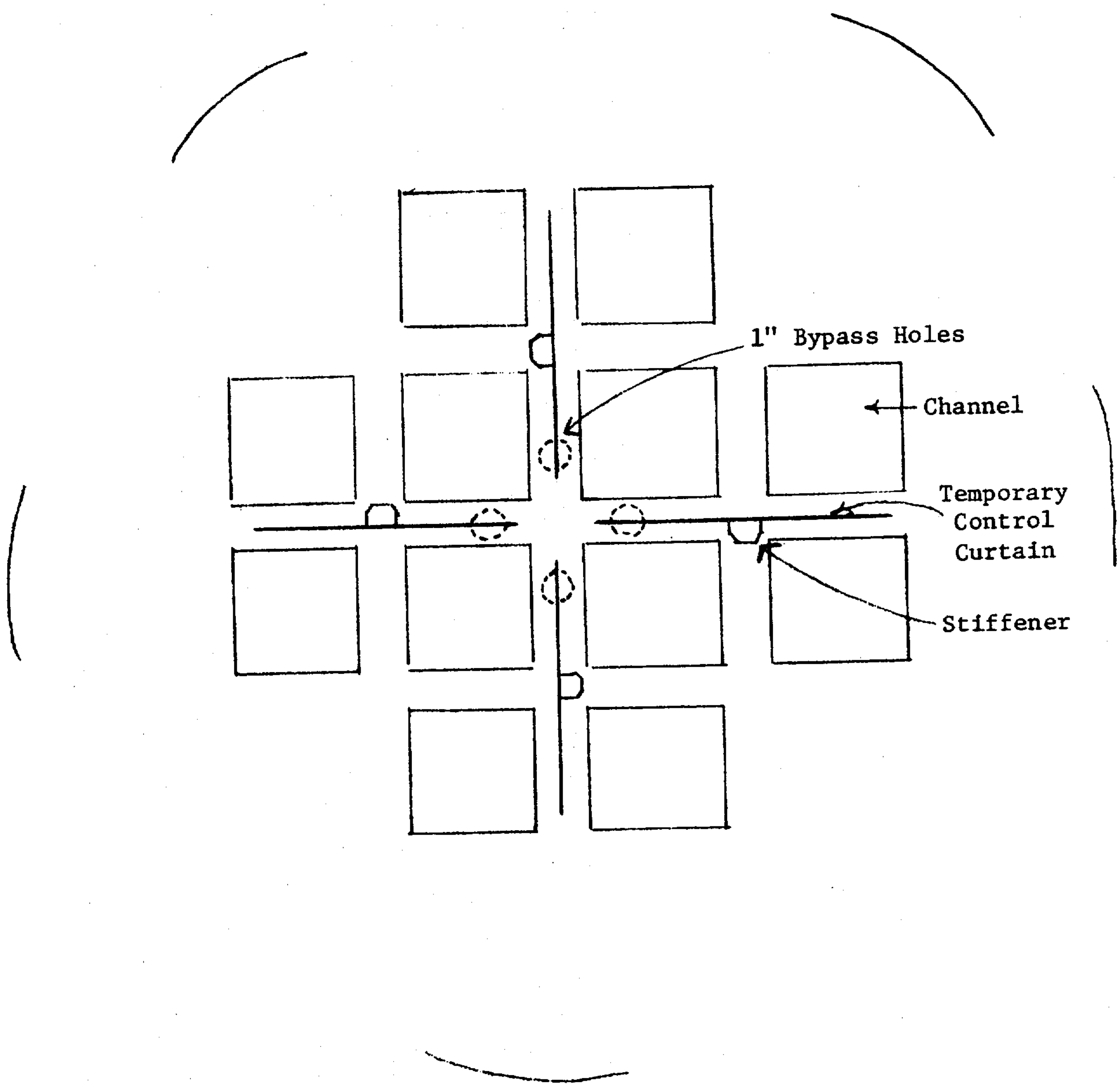


Figure 5

GE FLOW TEST FACILITY
PLAN



flow through the bypass holes equivalent to the bypass flow in a reactor operating at up to 120% of rated core flow. The channels contain no fuel rods and no flow is directed up through the channels. Proximity transducers were mounted on the bottom of three channels and the top of one channel.

Although only preliminary test results are available, both visual observation and the transducer output confirmed that curtain vibration occurs. Visual observation showed the curtains are displaced in a direction opposite from the flow holes. The degree of displacement as well as the amplitude of vibration decreased with decreased bypass flow. No evidence of resonant vibration was observed.

2.3 Consequences

The primary function of the channel box is to separate two parallel flow paths: a high flow channel upward through the bundle and a low-flow bypass region in which the control rods move. The channel also provides a bearing surface for the control rods and protects the fuel rods during handling. The channels also act as a heat sink in the event of a loss-of-coolant accident.

The following sections contain an evaluation of the effect of holes in channels on thermal-hydraulic margin, the effect of cracked channels on control rod operation, and the effect of potentially loose zircaloy pieces on thermal-hydraulic margins in localized blocked zones.

2.4 Affected Reactors

Only those reactors that have both bypass flow holes and control curtains have the potential for damaging vibrations. This is confirmed by inspection of fuel channels in reactors that have control curtains, but no bypass flow holes. Further confirmation comes from the observation that channels in KKM and VY located adjacent to control curtains that were not adjacent to bypass flow holes had no damage.

Of U. S. boiling water reactors, only the Vermont Yankee and Pilgrim reactors have both control curtains and bypass flow holes and are, therefore, subject to vibration induced channel damage. Although earlier designs, such as Oyster Creek and Monticello employ temporary control curtains, they do not have bypass flow holes. Therefore these designs are not subject to the channel damage observed in KKM and VY. More recent designs, such as Quad Cities and Browns Ferry, have bypass flow holes but employ gadolinia in the fuel pellets for reactivity control rather than temporary control curtains. Therefore, these designs are also not subject to the vibration induced channel damage observed in VY and KKM.

The Pilgrim design incorporates both features necessary to produce vibration damage, and the operating history of Pilgrim is intermediate between that of KKM and Vermont Yankee as can be seen from the tabulations below. The channel wear can be expected to be similar to that already observed.

Table 2.1

Operating Times

Reactor	Operating Time (mos)	Approximate Fuel Exposure (MWD/T)
KKM	14	6200
Vermont Yankee	9	3300
Pilgrim	11	4500

Table 2.2

Operating Flows

Percentage Rated Flow	Days of Operation		
	KKM	Pilgrim	VY
116-120%	1	0	0
101-115%	37	0	0
91-100%	300	220	127
Other	90	15	143

As indicated in the above table, Pilgrim has been operating for a longer time than the Vermont Yankee but for a shorter time than KKM. The time during which the Pilgram core flow exceeded 90% rated is also intermediate between KKM and Pilgrim. At the refueling outage in August and September 1973, KKM had already accumulated an operating history which exceeds the accumulated history on the Pilgrim Station by more than 60 days, the time period specified in our October 16, 1973 letter. Since the bypass flow holes are the same size in all of these three reactors, but the Pilgrim core pressure drop is less than in KKM or VY, the bypass flow velocity was lower in Pilgrim. Since the Pilgrim core flow is limited to 50% of rated, the bypass velocity will be less than half of that experienced by KKM or VY. Lower velocities result in less vibration and less channel damage. Consequently, the operating data now available can be used to predict that channel damage in Pilgrim beyond the currently imposed 60-day time limit would not exceed that observed in KKM. Extrapolation beyond the KKM experience has not been evaluated.

3.0 SAFETY ANALYSIS

The Pilgrim Station has not been shutdown and inspected to establish that the channel boxes have been subjected to the wear observed in the KKM and Vermont Yankee reactors. However, both the staff and the licensee have assured that the type and extent of channel box wear described in Section 2.0 is also present in the Pilgrim reactor. The following safety analysis is predicated on that assumption.

The objective of the safety analysis is to demonstrate that operational restrictions on the facility, 50% of rated power and 50% of rated flow, are adequate to compensate for predicted effects of channel box wear. Steady state operation, effects of potential transients and postulated accidents were considered in the analyses. The substantial reduction in reactor power level simplified the analyses in that very conservative assumptions could be postulated to demonstrate that appropriate criteria are met.

3.1 Steady State Operation

The rationale for selecting the 50% of rated power level was derived from consideration of certain inherent operating characteristics of a BWR. For example, operation of the reactor during natural circulation flow is restricted to a power level of about 55% of rated power by the plant protection system instrumentation (flow biased scram). Normal operation using the recirculation pumps permits variation of reactor power by adjusting pump speed and varying the coolant flow rate. This power-flow relationship establishes the normal control characteristic for the plant. The intersection of the normal flow control characteristic with the natural circulation flow characteristic occurs at about 50% of rated power and about 30% of rated flow. A primary coolant flow rate of 50% of rated flow is provided by operation of the recirculation pumps, at the minimum flow rate for which stable two-pump operation was established.

Operation of the plant under natural circulation conditions would provide considerable thermal-hydraulic margin, namely, a minimum critical heat flux ratio (MCHFR) of about five, as compared with a design value of about two for full-power operation. Operation at a flow rate within the channel boxes greater than the natural circulation flow rate would further increase the thermal hydraulic margin. However, operation of the facility at the higher flow rate causes the ambient pressure within the channel boxes to exceed the pressure in the spaces between boxes. Hence, the potential for leakage of coolant from the channel boxes through slots would be possible, resulting in lowered flow rate inside the channel box and a decrease in the thermal-hydraulic margin. The balance between these interactive phenomena has not been quantified, but the bounding assumption is that all of the increased coolant provided by turning on the pumps leaks to the bypass zone. In this case the reactor is still operating in the acceptable natural circulation mode. Flow blockage that might be caused by a loose piece of Zircaloy would have at worst a highly localized effect on MCHFR. In our judgment this would not lead to a cladding failure event, since the channel box thickness is small with respect to the flow channel width, and at most only a portion of any fuel rod could be affected.

3.2 Transients

Operation of Pilgrim at 50% of rated flow and power will result in substantial improvements in thermal margins, even with defective channel boxes. For limiting plant transients that result in flow decreases, such as pump trip or seizure, the plant is already near the natural circulation point, for which MCHFR would be greater than 4.0. Transients that lead to flow increase, e.g., failure in the recirculation pump speed control, would bring about a power increase, but since the plant would be operating on the normal flow control characteristic, then it would be operating in a mode previously analyzed and accepted.

Other plant transients, such as loss of load or loss of feedwater flow would produce less severe transients since the MCHFR at the start of the transient is much larger than design value (about five versus two in normal operation). Pressurization type transients such as turbine trip, main steamline isolation valve closure, or loss of condenser vacuum would not be affected by postulated holes in the channel boxes, since transients such as these have been shown to be acceptable with the assumption that natural circulation operation is the initial condition. Operation at 50% of rated flow should produce even more conservative results. The same reasoning leads to the conclusion that the postulated transients analyzed in the FSAR remain as the limiting situation, since the steady-state MCHFR is so much higher at 50%.

Pilgrim has already undergone stability tests at 50% power, and in natural circulation conditions, and was found to be stable. The presence of slotted channel walls would not affect the results of those tests, since little or no flow bypass from coolant channel is expected during natural circulation.

3.3 Accident Analysis

The staff has reconsidered the impact that the slotted channel boxes would have on accident analyses. Of the accidents considered only the loss-of-coolant accident (LOCA) is affected. The consequences of the rod drop accident, most recently reviewed in Section 4.4 of Ref. 7, are not considered to be affected since the slotted channel boxes have no effect on control rod worths, and the analyses presume an adiabatic heatup.

The staff analysis of the LOCA was performed with the computer code MOXY (Reference 8). Three MOXY calculations were performed, with the principal parameter of each calculation being the peak cladding temperature (PCT). The object of the calculations was to determine the effect that degraded heat transfer, such as caused by loose pieces of Zircaloy or excessive flow bypass, would have on the LOCA calculation. A summary of the calculations is presented in Table 3.1.

Table 3.1

FUEL ELEMENT HEATUP CALCULATIONS

Case	APLHGR* kw/ft	$\bar{h}$ gap ** BTH/hr-ft^2°F	PCT***
1	11.1	350	2050
2	5.55	145	1525
3	5.55	145	2027

*APLHGR - Average planar linear heat generation rate.

**$\bar{h}$ gap - Average gap conductance.

***PCT - Peak cladding temperature.

The staff's LOCA calculations for Pilgrim were based on heating rates appropriate to Pilgrim and in some cases, based on input data from other plants. Local peaking factors from plant "F" of Reference 9 were used, and sink temperatures were estimated. The staff concludes that the input data used for Pilgrim were conservative.

The Pilgrim calculations showed:

Case 1. At rated power and flow the PCT was 2050°F.

Case 2. At 50% flow and power the PCT was 1525°F.

Case 3. At 50% flow and power, and with a degraded heat transfer coefficient, the PCT was 2027°F.

The degraded heat transfer coefficient was assumed to be decreased linearly to zero from the initial steady-state value, in the first second of the blowdown. This is clearly conservative for a jet-pump plant which maintains substantial core flow for several seconds after the postulated break. It is, therefore, concluded that analysis of a LOCA with very conservative assumptions regarding heat transfer effects for worn-through channel boxes yields acceptable results at 50% power and flow.

The mechanical performance of the defective channel boxes was evaluated by the staff. During steady-state operation with 50% flow and power, there is a net outward pressure differential across the channel wall. The pressure is predicted to be mostly outward during a postulated LOCA, which minimizes the potential for channel collapse onto the fuel assembly.

In the remote possibility of a slight inward pressure during a LOCA, the magnitude is expected to be less than 1 psi. The staff's calculation indicates that the fuel assembly, including the spacer grid, has sufficient strength to sustain such an inward pressure, without structural dependence on the channel box.

The Vermont Yankee plant has operated with holes in channels at 100% flow and 75% power with a higher outward pressure differential than Pilgrim at 50% of rated flow. There have been no difficulties experienced with control rod movement attributable to excessive deflection of channel boxes. With the reduction of power and flow to 50% in Pilgrim plant, further degradation, i.e., wear and deflection, is either stopped or significantly slowed. This was verified from the mockup flow test discussed earlier (Sec. 2.0). Regular exercising of the control rods insure that there are always sufficient numbers of control rods that can scram without channel interference. The Pilgrim plant is being operated so that control rods in the potentially damaged regions are always inserted at least 3 feet into the core, well beyond the position of the expected channel box damage.

The differential pressure across the channel will be further reduced in magnitude due to the initially lower power and flow and by the presence of any cracks. During a LOCA, it is expected that the differential pressure will decrease from the 2 psi steady state value.

During power operations the channel boxes have accumulated neutron fluence which alters the Zircaloy ductility. Although the total ductility is reduced, most of the remaining ductility is available as localized strain. This relatively large localized strain is beneficial in rapidly attenuating stress concentrations at the tip of any cracks. The accelerated in-reactor creep of Zircaloy-4 will also contribute to the alleviation of stresses concentrated at the tip of cracks. Subsequently, brittle self-propagation of channel cracks is unlikely. However, further ductile extension of a crack is possible. The ductile extension of the crack and opening of the slot will serve to decrease any stresses in those channel boxes due to the lateral pressure gradients. The maximum ductile crack propagation was 4 inches in KKM, which ran at several times the pressure differential for the 50% power and flow conditions in Pilgrim. The maximum ductile crack propagation was 1-1/2 inches in Vermont Yankee, again with 4 times the pressure differential currently in Pilgrim. It is not likely, nor has it been observed, that the worn-through channel will further crack to the point of losing its geometric integrity. Thus the channel box should still be able to perform its functions as described in Section 2.3.

4.0 CONCLUSIONS

From our understanding of the wearing mechanism, both we and the licensee assume that the channel boxes in Pilgrim have sustained similar damage. The effect of this damage has been adequately accounted for by limiting operations to 50% of rated flow rate and 50% of rated power. With these limits the plant has been shown to comply with the ECCS Interim Acceptance Criteria.

Based on the analysis of fuel channel box damage described above, there is reasonable assurance that Pilgrim Nuclear Power Station may be operated, subject to 50% reductions in rated power and rated core flow, for a limited period of about sixty days from October 16, 1973, without endangering the health and safety of the public. As noted in Section 1.0, the Vermont Yankee Nuclear

Power Station is not operating and will not operate while the matter of channel box damage in that facility remains under consideration by the staff.

There have been two alternative types of vibration control devices proposed by GE. At present, the licensees of Pilgrim and Vermont Yankee and the Regulatory staff are evaluating these vibration control devices.

REFERENCES

1. "Recent Problems at the Mühleberg Nuclear Power Station (GE-BWR)," Letter from P. Courvoisier, Abteilung für die Sicherheit der Kernanlagen to Dr. Clifford Beck, dated October 11, 1973.

2. Note to Files from Assistant Deputy Director for Technical Review, dated October 16, 1973.

3. Letter to Boston Edison Company from AEC, V. A. Moore, Assistant Director for Boiling Water Reactors, dated October 16, 1973, Docket No. 50-293.

4. "Joint Petition for Immediate and Indefinite Shutdown of Vermont Yankee Nuclear Power Station and Pilgrim Nuclear Power Station," filed October 15, 1973.

5. "Affidavit of Henry W. Kendall, Ph.D.," received October 23, 1973 by the AEC.

6. <u>Memorandum and Order</u> In the Matter of Vermont Yankee Nuclear Power Station, dated October 23, 1973, Docket Nos. 50-271 and 50-293.

7. "Technical Report on Densification of General Electric Reactor Fuels" report by the Regulatory Staff of the U.S. Atomic Energy Commission, August 23, 1973.

8. D. R. Evans, "MOXY: A Digital Computer Code for Core Heat Transfer Analysis," IN-1392, 1970.

9. "Densification Considerations in BWR Fuel Design and Performance" NEDM-10735, Supplement 6, January 1973.

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A. PHILIP BRAY
GENERAL MANAGER
BWR PROJECTS DEPARTMENT

April 22, 1975

RECEIVED APR 25 1975

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Case:

This will document the recent telephone calls to you and your staff on April 17, 18 and 21, 1975 informing you of the discovery of fuel channel damage in a General Electric designed boiling water reactor (BWR) located in a foreign country.

At this time, inspection is only in the early phases following plant shutdown. Initial expectation is that the damage could have resulted from vibrational contact between the in-core local power range monitor (LPRM) tubing and adjacent fuel channels, causing significant wear on the fuel channels. Indication of an anomalous condition at the affected plant was first detected by the observance of unusual "noise" indicative of tube vibration during traversing in-core probe (TIP) calibrations of the LPRM.

No similar extensive channel damage has ever been observed in a domestic BWR, thus the possibility exists that this is a unique plant condition. However, we have advised BWR plant operators to examine recent TIP traces or take readings if appropriate to determine if "noise" characteristic of the affected plant is observed.

The significance of this condition, if it exists in a BWR to the extent that fuel channel wear progresses to the point of through wall openings is similar to that identified in earlier core designs where temporary absorber curtain elements caused channel wear. Documentation of this curtain effect can be found in dockets 50-271 (Vermont Yankee) and 50-293 (Pilgrim).

We have scheduled a meeting with your staff on April 24, 1975 to discuss the state of information on this matter. We will continue to evaluate information as it is received and advise you of our conclusions.

Sincerely,

A. Philip Bray

/lml

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of NEBRASKA PUBLIC POWER DISTRICT (Cooper Nuclear Station)	)))))	Docket No. 50-298

ORDER FOR MODIFICATION OF LICENSE

I.

Nebraska Public Power District (the Licensee) is the holder of Facility Operating License No. DPR-46 which authorizes operation of the Cooper Nuclear Station (the Cooper Station or the facility) at steady-state reactor core power levels not in excess of 2381 megawatts thermal (rated power). The facility is a boiling water reactor (BWR) located at the Licensee's site near Brownville in Nemaha County, Nebraska.

II.

1. By telephone calls on April 17, 18 and 21, 1975, and by letter of April 22, 1975,[1/] the NRC Staff was informed by the General

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1/ Copies of this letter are being placed 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, Auburn Public Library, 1118 15th Street, Auburn, Nebraska.

Electric Company (GE) that fuel inspections in a foreign BWR had revealed significant wear and some cracking of several zircaloy fuel channel boxes. The discovery of this damage had been preceded by anomalous signals of certain incore neutron detector instruments. The NRC Staff has tentatively concluded that damage to the fuel element channel boxes and the anomalous instrument readings may be due to vibration of the instrument tubes, produced by coolant flow through bypass holes in the reactor lower core support plates. To determine whether or not any BWRs within the United States could be subject to similar damage, GE notified the operators of eleven United States plants having a similar design that utilized bypass holes in the reactor lower core support plate and asked that they inspect the incore neutron detector instrument readings from the traversing incore probe. GE was informed, and subsequently informed the NRC Staff at a meeting held on April 24, 1975, that anomalous readings, similar to those experienced in the foreign reactor, were found at the Cooper Station.

2. The fuel bundles of BWRs are encased in square zircaloy tubes known as channel boxes which separate parallel coolant flow paths in the core, provide guide surfaces for the motion of control rods, protect fuel bundles during handling, and provide load support during transient loadings that may result from operational abnormalities and accident conditions. The incore instrumentation in which the anomalous readings were found is located in the zone between channel box assemblies known as the bypass region. Flow in this region is provided by clearance between the lower tie plate and the channel box assembly and, in eleven reactors, is also provided by bypass flow holes in the lower core support plate. As noted, it is believed that the channel box damage experienced at the foreign reactor may be the result of vibration of the instrument tube which, in turn, was caused by flow through the bypass holes.

3. Of the operating BWRs in the United States having a design which utilizes bypass holes, only the Cooper Station has reported anomalous readings from the traversing incore probe which are indicative of damage or cracking of the channel box. With this indication of possible damage existing in the core, further

operation under similar conditions may lead to further core damage and, if continued, may result in an unsafe condition.

4. In late 1973 and early 1974, similar channel box damage was observed in the Vermont Yankee Nuclear Power Station, in the Pilgrim Nuclear Power Station, and in a foreign reactor. In those instances, excessive vibrations of certain temporary control curtains, which at that time were also located in the bypass region between the channel boxes a few inches from the instrument tubes, had caused the channel box damage. In connection with the Vermont Yankee and Pilgrim facilities, tests at a GE flowtest facility demonstrated that the vibration of the control curtains in the bypass region resulting from flow through bypass flow holes was substantially reduced at flow conditions of approximately 50% full flow.[2/]

5. Recently, the Licensee, at GE's request, operated the Cooper Station for a short period of time at a limited flow of about 55% of full core flow. Under these conditions, the anomalies

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2/ See Safety Evaluation by the Directorate 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. Copies of this document are being placed for public inspection in the public document rooms identified in footnote 1 above.

in the TIP trace were no longer observed, substantiating the belief derived from earlier experience that, at flows of approximately 50% of full core flow, the flow through the bypass holes was reduced sufficiently to substantially reduce excessive vibration of the instrument thimbles located in the bypass region. This, in turn, should reduce further channel box damage.

6. Further tests are being conducted by GE, and additional studies and investigations by GE and others are underway, including further inspection of the condition of the core in the foreign reactor. Nonetheless, as an interim measure, while these further data are being developed, in order to minimize the possibility of any further damage to the fuel channel boxes at the Cooper Station, the NRC Staff has concluded that core flow should not be allowed to exceed 50% of design flow. This, in turn, will require facility power to be reduced to approximately 50% of full power. Operation at this power level will provide assurance of facility safety while further data are being developed.

7. The Licensee has agreed that it will not operate the facility at core power levels exceeding 50% of rated power or core flow rates exceeding 50% of design flow rate without prior written approval of the NRC. The NRC Staff believes that

the Licensee's action, under the circumstances, is appropriate and that this action should be confirmed by NRC Order.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-46 is hereby amended by adding the following new provision:

> By reason of the circumstances outlined in the Order for Modification of License, dated April 26, 1975, the Licensee shall not operate the facility at core power levels exceeding 50% of rated power or core flow rates exceeding 50% of design flow rate without prior written approval of the Director, Office of Nuclear Reactor Regulation.

Respectfully submitted,

Benard C. Rusche

Benard C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 26th day of April, 1975.