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ENCLOSURES:
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.....re their 6-2-75 ltr.....(40 cys enclrec'd)**

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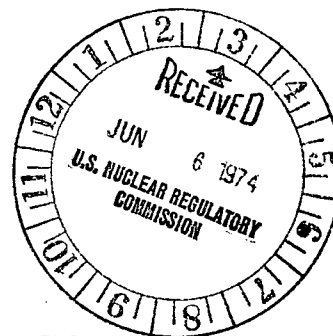
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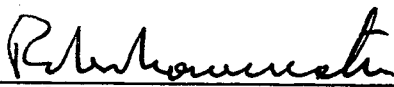
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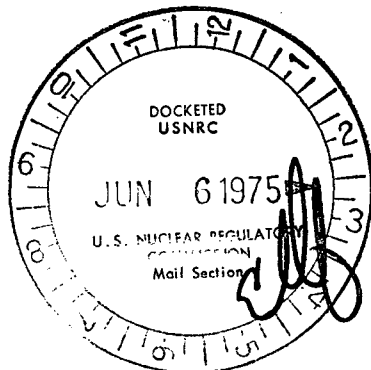
Mr. Charles W. Sandford's letter of June 2, 1975, indicated that further data would be submitted shortly regarding Iowa Electric's planned program for inspection of the Duane Arnold Energy Center core internals and any necessary repairs. Transmitted herewith are forty copies of a document entitled "Duane Arnold Energy Center Channel Inspection Program and Core By-Pass Flow Hole Plug Mechanism Design." Additional information will be provided when available.

Sincerely,

Lowenstein, Newman, Reis & Axelrad
Attorneys for
Iowa Electric Light and Power
Company


By Robert Lowenstein

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6165

June 5, 1975

DUANE ARNOLD
ENERGY CENTER

CHANNEL INSPECTION
PROGRAM AND CORE
BYPASS FLOW HOLE PLUG
MECHANICAL DESIGN

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1. INTRODUCTION AND SUMMARY

During a recent outage at a foreign plant, it was determined that some fuel assembly channels exhibit severe corner wear adjacent to in-core neutron monitor and startup source locations. The most severe wear corresponds to the location of the local power range monitor (LPRM). Less severe channel wear was found at areas which correspond to the source range monitor (SRM), intermediate range monitor (IRM) and startup source locations.

It was postulated, and subsequently confirmed, that the severe wear was caused by vibration of the in-core tubes due primarily to a high-velocity jet of water flowing through the bypass flow holes in the lower core plate. This caused the tubes to wear against the channel corner. In some cases, the wear was observed to penetrate the channel wall.

Duane Arnold Energy Center (DAEC) is similar to the foreign plant in that they are General Electric Boiling Water Reactors (BWR's) which incorporate bypass flow holes in the lower core plate. This report describes the actions that are being taken at DAEC to check for the occurrence of and to eliminate the possible recurrence of significant channel wear during the remainder of this fuel cycle.

The following sections of this report describe the diagnostic inspection program that will be performed to determine the extent of channel wear. Also included is the mechanical analysis performed in the development of a plug design for the bypass flow holes at DAEC to eliminate the primary cause.

In addition to plugging the bypass flow holes at DAEC, all channels which experienced corner wear greater than 10 mils in the lower half or 21 mils in the upper half of the channel will be replaced with new channels. This limit is consistent with the allowable on-site handling scratches or wear that any channel can experience during normal fuel handling operations and allow 4 years of additional duty.

2. CHANNEL INSPECTION PROGRAM

2.1 INITIAL DISCOVERY OF OCCURRENCE

During December 1974, an unusual transversing in-core probe (TIP) trace was noted at a foreign plant. Evaluation of this trace and subsequent diagnostic tests led to the conclusion that there was a high probability that a neighboring channel had developed a hole. Prior to the reactor shutdown, a second location developed similar noise characteristics on its TIP traces.

Inspection of the channels surrounding these locations at a subsequent outage confirmed the hypothesis. In order to assess the probable cause and amount of damage, a detailed inspection plan was developed and implemented. The results of this inspection are presented in Section 2.3.

Recent TIP traces taken at DAEC contain less noise content than the traces at the foreign plant. However, there are indications of LPRM vibrations which have led to the decision to run baseline tests, shut the reactor down, perform a diagnostic inspection, and, if necessary, plug the bypass flow holes. The following sections describe previous channel inspection experience, inspection methods and the DAEC diagnostic inspection program.

2.2 INSPECTION METHODS

2.2.1 Channels

Fuel assemblies are removed diagonally from each area surrounding the particular in-core monitor. These assemblies may be transferred to the pool storage rack or moved directly to the fuel preparation machine as required. A 360 degree detailed visual inspection of the full length of each channel is performed by the use of a borescope to determine any abnormal conditions. Photographs are taken of typical abnormal observations as required.

Channel corner wear depth due to interaction of the adjacent in-core instrument is conservatively estimated by observing the width of such wear and correlating that to the theoretical depth which would be obtained if the in-core instrument wore directly inward (no circular wiping motion). The theoretical depths obtained for such direct bearing (no circular motion) are presented on Figure 2-1. Since lateral motion of the in-core instrument is quite likely, based on observations which typically show more than one channel affected at each in-core location (indicative of the in-core moving in a circular or partial circular pattern), the wear depths so calculated predict a deeper penetration than that which is likely to occur and thus the model used is conservative. This is substantiated by some observations at the foreign plant which show a wear width of 3/8 inch (without perforation of channel) which would correlate to about a 90 mil wear depth; this, however, is greater than the channel wall thickness of 80 mils.

The wear widths are estimated by fixturing an optical standard adjacent to the channel within the field of vision of the objective lens of a borescope. The optical standard is a rod having a series of discrete diameters as shown below.

<u>Standard Diameter</u>		<u>Equivalent</u>
<u>mils</u>	<u>inches</u>	<u>Maximum Wear Depth (mils)</u>
45	$\sim 3/64$	1.3
93	3/32	5.5
125	1/8	10
187	3/16	22

The results of past inspections have shown that this method of estimating wear depth is fast and has good sensitivity up to about 30 to 40 mils of calculated depth. Above 40 mils calculated depth, the sensitivity of the method tends to drop off since relatively small changes in width tend to give relatively large changes in additional depth of apparent wear.

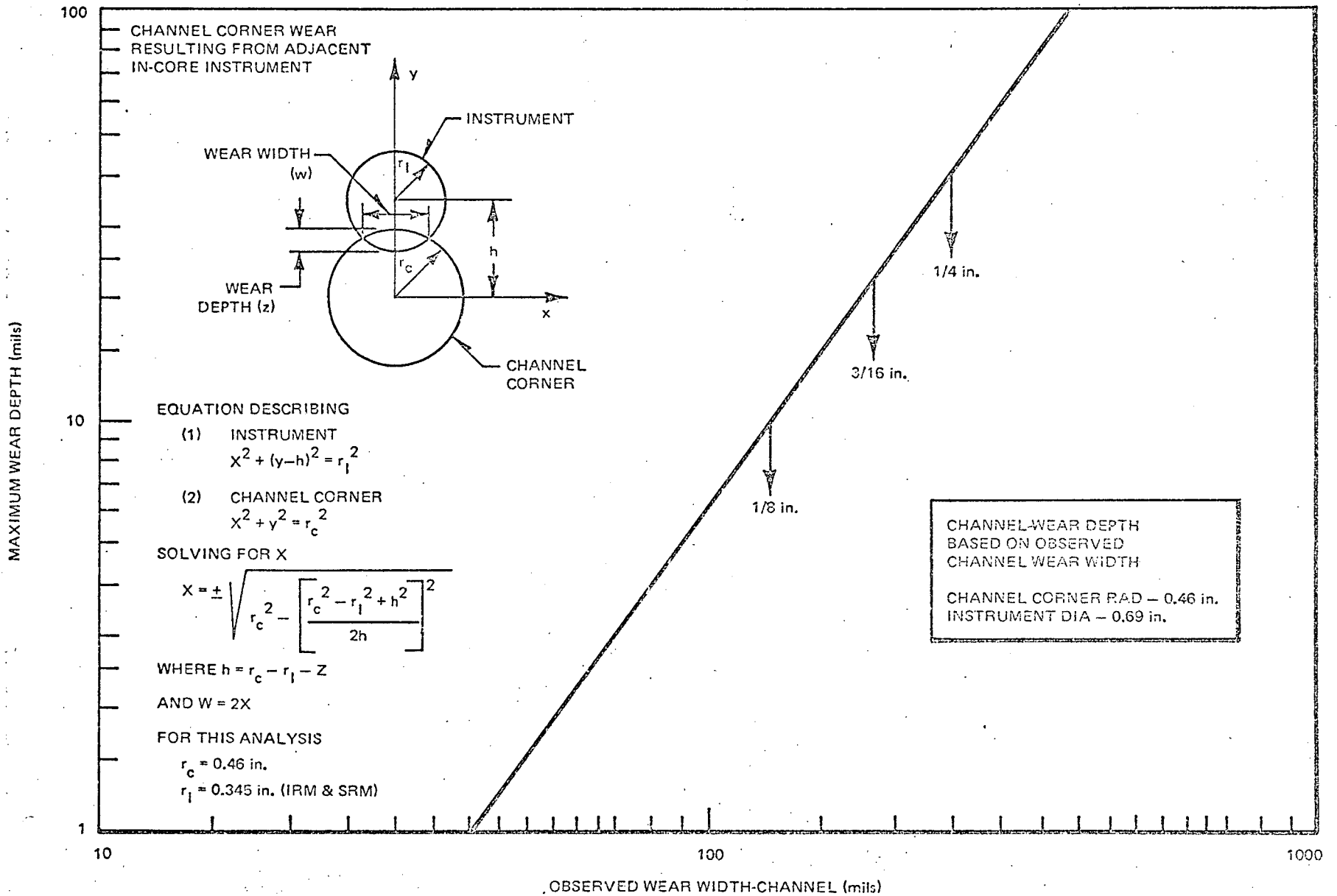


Figure 2-1. Channel Corner Wear Resulting from Adjacent In-Core Instrument

2.2.2 Reactor Internals

Prior to the removal of any fuel assemblies, the relative heights of all fuel assemblies are compared to assure full seating. Next all fuel assemblies are checked to assure correct orientation.

Incores are inspected using a remote TV camera. The borescope is used to supplement TV inspection when necessary to obtain high resolution capability.

During an in-core inspection, the engagement of the in-core detector plunger with the notch of the top guide is checked. The plunger is assumed to be fully engaged if the nitrided plunger's diametral change is not seen below the bottom of the top guide.

Next, the full length of the incore is traversed down to the orificed fuel support noting and video recording the in-core position relative to the adjacent fuel bundles looking for wear areas, scratches, or gouges. This process is performed for the four quadrants of the incore.

Finally, the areas where the incore enters the in-core guide tube at the core plate are inspected and video recorded. The relative position of the in-core spacer ferrule to the top of the guide tube and the top of guide tube elevation relative to the core plate is determined, and the location and the number of core bypass holes are confirmed.

2.3 INSPECTION RESULTS - FOREIGN PLANT

During the foreign plant channel inspection program, 197 channels received a detailed visual inspection. The results of this inspection are given in Table 2-1.

The inspection results confirmed the severe channel corner wear corresponded to LPRM, SRM, IRM and startup source locations. No unusual channel wear was found at other locations. Of the channels inspected, including all channels surrounding in-core tubes, three were found with through wall wear. Two were located around the LPRM with the first unusual TIP trace and one next to the second.

TABLE 2-1
FOREIGN PLANT CHANNEL
INSPECTION SUMMARY

<u>Type of Incore</u>	<u>Number Inspected</u>	<u>Estimated Wear (mils)</u>			
		<u>Through Wall Wear</u>	<u>30-70</u>	<u>10-30</u>	<u><10</u>
LPRM	124	3	39	50	32
Source	20	-	8	9	3
SRM	16	-	1	6	9
IRM	32	-	3	11	18
None	5	-	-	-	5
Total	197	3	51	76	67

2.4 DIAGNOSTIC INSPECTION PROGRAM - DAEC

2.4.1 Channels

The purpose of the channel visual inspection is to determine if any channel wear is present due to possible interaction between in-core monitors and adjacent channels, and if so, to determine the magnitude and extent of such wear.

The channel visual inspection will be performed in two phases, diagnostic and general, if required. The diagnostic phase is planned such that a representative sampling of the channels surrounding each in-core type of monitor is visually inspected. The general phase of channel inspection will be performed only if the results of the diagnostic phase indicate that channel wear could be expected in some or all of the remaining channels adjacent to incores which were not inspected during the diagnostic phase of inspection.

The detailed inspection sequence is based on a selection of fuel assemblies with respect to the various in-core monitors so that a continuous representative sampling is maintained throughout the entire inspection phase. Initially, the channels associated with those LPRM's giving the highest degree of noise or highest density of 2.5 Hz vibration will be inspected. There are 24 channels in this category. These channels are identified as Priority 1 on Table 2-2.

If unacceptable wear is not detected in this sampling of channels, then an additional 12 channels will be selected for visual inspection from IRM, SRM, and source locations. If still no unacceptable wear is detected then all channels will be acceptable for continued reactor operation. These channels are identified as Priority 2 on Table 2-2.

If unacceptable wear is detected on either the first sample of 24 channels or the second sample of 12 channels then all channels adjacent to in-core monitors will be considered suspect and removed from further operation until such time that detail inspection can be performed on such channels.

TABLE 2-2

SEQUENCE AND PRIORITY OF CHANNEL DIAGNOSTIC VISUAL INSPECTION

<u>Priority</u>	<u>Number of Assemblies To Be Inspected</u>	<u>In-Core Monitor Location</u>	<u>Type of (1) Incore</u>	<u>LPRM Noise Analysis</u>	
				<u>Noisy TIP</u>	<u>PSD (2)</u>
1	4	16-09	L-3	X	X
1	4	16-33	L-4	X	X
1	4	32-17	L-4	X	X
1	4	24-17	L-4	-	X
1	4	24-33	L-4	-	X
1	4	24-09	L-3	-	-

TABLE 2-2 (Continued)

<u>Priority</u>	<u>Number of Assemblies To Be Inspected</u>	<u>In-Core Monitor Location</u>	<u>Type of In-Core</u> ⁽¹⁾	<u>LPRM Noise Analysis</u>	
				<u>Noisy TIP</u>	<u>PSD</u> ⁽²⁾
2	2	08-37	IRM	N/A ⁽³⁾	N/A
2	2	24-37	SRM	N/A	N/A
2	2	12-21	Source	N/A	N/A
2	2	08-13	IRM	N/A	N/A
2	2	32-21	SRM	N/A	N/A
2	2	16-37	Source	N/A	N/A

Summary of Sampling:

	<u>L-4</u>	<u>L-3</u>	<u>L-1</u>	<u>L-0</u>	<u>IRM</u>	<u>SRM</u>	<u>Source</u>
In-Core type channels adjacent to incores	40	8	12	20	24	16	16
Channels to be inspected	16	8	0	0	4	4	4
Priority for inspection	1	1	-	-	2	2	2

(1) L refers to LPRM; the number following refers to the number of bypass flow holes around the LPRM.

(2) Refers to high density of 2.5 Hz vibration.

(3) N/A - not applicable.

The extent of the general inspection will be limited to those channels suspected to be damaged based on information obtained during the diagnostic phase. The inspections to be performed during the general phase will be limited to visual inspection of only those channel areas likely to be worn. It is thus expected that if the general inspection is implemented, only the corner adjacent to the incore will be examined; a full length inspection of this corner will be performed.

The maximum allowable depth of wear at the channel corner considered acceptable is 10 mils in the lower 80 inches of channel and 21 mils in the upper half of the channel. Channels with observed acceptable wear on the corner shall not be reinserted in the core next to an in-core instrument where additional wear could occur during subsequent reactor operation.

A depth of 20 mils is the maximum allowable wear on the channel sides (flats). Wear to this depth has not been seen on any channel inspected to date. Some marks are typically observed on the channel sides due to the control blade roller, but these characteristically have very little depth and have always been estimated to be less than 5 mils deep.

The criteria for judging the allowable wear is based on fatigue crack initiation. Stress rupture damage accumulation is found to be insignificant at the operating conditions of interest. The pressure duty considered consisted of low frequency pressure variations occurring from power level changes and the potential occurrence of abnormal operational transients. The wear limits are determined to allow 4 years of additional duty: 1 year at a radial power of 1.4 and 3 years at a factor of 1.2 considering that additional wear does not take place. In the fatigue analysis an axially directed scratch 1 mil in depth and with a 1/2 mil radius is assumed to be on the inside surface of the corner at which the wear is postulated. A notch sensitivity factor of 1 is used in the fatigue analysis. The conservatism of the assumption on the presence of a scratch together with the large notch sensitivity factor chosen and the remaining margin predict that the allowable wear values are conservative.

2.4.2 Reactor Internals

The purpose of this inspection is to examine reactor internals in order to determine the cause of the in-core/channel wear problem and to assess any damage caused by the problem. Any abnormality or item of importance shall be video recorded and the TV monitor picture photographed. This inspection is optional except in core locations where a channel with through wall wear has been observed or at that location which exhibited the highest degree of channel wear if no channel perforation was observed. Past experience has shown that channels exhibit significantly more wear than the in-core monitors due to the difference in materials.

If no rejectable conditions are present on any in-core instrument examined then no additional in-core examination is required. If rejectable conditions are observed by TV on any in-core instrument (LPRM, SRM, IRM, or source holder) confirmation is to be made by detailed borescopic inspection. If a reject condition is confirmed, additional in-core inspection of instruments that are adjacent to similarly damaged channels and progressively less damaged channels is performed. In-core instrument inspection is discontinued when no further rejectable conditions are observed.

Under core plate inspection was performed previously at a reactor that had three channels with through wall wear. The results of this inspection revealed nothing unusual under the core plate. Based on past results, this inspection can be omitted providing that the extent of channel damage or in-core instrument damage is no worse than previous experience.

The prime wear criteria for the LPRM cover tube is that it not be penetrated. The cover tube performs an enclosing, not structural, function.

Allowable Maximum Wear Values, inches

<u>LPRM Diameter</u>		
<u>Inches</u>	<u>Depth</u>	<u>Band Width for Depth Estimate⁽¹⁾</u>
0.700	0.020	0.375
0.750	0.039	0.500

(1) Wear band width is the maximum continuous width of wear indication. Burnish marks should not be considered as wear marks unless the line of demarkation between burnishing and actual wear cannot be determined.

Scratches and gouges that do not penetrate the cover tube of the LPRM will have no effect on the ability of the cover tube to perform its required function. Examination for scratches and gouges need only be made at locations where actual channel through wall wear has occurred because only at these locations does a mechanism for producing scratches and gouges exist.

The IRM and SRM dry tubes are a vessel pressure boundary and bear the ASME code stamp. Thus, wear is an important consideration. The wear criteria were established taking the ASME code requirements into consideration. Maximum depth of wear allowed is 0.020 inch. This wear depth can be estimated by assuming that a wear band width of 0.375 inch continuous width is 0.020 inch deep. Burnish marks need not be considered as wear marks unless a line of demarkation between burnishing and actual wear cannot be determined.

Examination for scratches and gouges is only made at locations where fuel channel through wall wear has occurred. At those locations, a scratch or gouge depth limit of 0.015 inch maximum is applied. Depth may be determined by replicas, by ultrasonic or eddy current examination, by comparison with standardized scratches, or by other suitable means. At locations where no fuel channel through wall wear exists, scratch and gouge examination need not be made because the mechanism for producing the scratches and gouges is absent.

The prime wear criteria of the source holder is that the outer tube not be penetrated. The cover of the source holder performs an enclosing, not structural, function. The recommended maximum allowable depth of wear is 0.020 inch. This depth can be estimated by assuming that a wear band width of 0.375 inch continuous width is 0.020 inch deep. Burnish marks should not be considered as wear marks unless a line of demarkation between burnishing and actual wear cannot be determined. Scratches and gouges must not penetrate the cover tube of the source holder. Other than this, they have no effect on source performance. Examination for scratches and gouges need only be made at locations where actual fuel channel through wall wear has occurred because only at these locations does a mechanism for scratches and gouges exist.

2.5 PREVIOUS EXPERIENCE - PLANTS WITHOUT BYPASS FLOW HOLES

General Electric has maintained an on-going program of obtaining information on the performance of channels in operating reactors. This information has been obtained from general visual observations of channels during refueling outages and from a comprehensive channel surveillance program which includes detailed visual examinations and dimensional measurements of a selected number of channels at specific lead performance plants.

The general observations are performed using binoculars during the normal operations of refueling outages. These observations are generally made while channeled fuel assemblies are being moved to and from the core and maneuvered during the out-of-core sipping operation in the fuel storage pool. Observations are also made in the fuel preparation machines when dechanneling defective or spent fuel and channeling of fresh fuel with irradiated channels. Visibility of the channel exteriors during these operations is adequate to observe significantly unusual channel surface conditions.

At the present time, there are 14 BWR's which do not have bypass flow holes in the lower core plate. Over 5000 channels from these reactors have been visually observed during the refueling operations described above. To date, there have been no reported indications of channel wear on any of these channels.

In addition to the general visual observations, detailed information has been obtained from the channel surveillance program which incorporates actual removal of channels from the fuel bundles and placing them in a special fixture which permits detailed observations to be made. In the fixture, the channel exterior is closely examined visually, and dimensions taken using special gauging tools.

At the present time, 143 channels surrounding in-core tubes from five plants have been inspected, and no indication of channel wear has been observed. The results of these inspections are presented in Table 2-3.

TABLE 2-3

SUMMARY OF CHANNELS INSPECTEDPLANTS WITHOUT BYPASS FLOW HOLES

<u>Channels Inspected</u>							<u>Exposure Range (GWD/t)</u>
<u>BWR</u>	<u>Plant</u>	<u>Total</u>	<u>LPRM</u>	<u>IRM</u>	<u>SRM</u>	<u>Source</u>	
2	A	29	4	1	1	1	6-17
	B	41	4	7	2	8	10-19
	C	34	11	3	5	0	11-17
3	D	25	6	3	1	1	11-15
	E	14	5	1	0	0	12-13
<u>Total Inspections</u>		<u>143</u>	<u>30</u>	<u>15</u>	<u>9</u>	<u>10</u>	<u>6-19</u>

Results and Conclusions

- No channel wear of any kind observed on channels.
- Minor marks (no depth perceptible) sometimes observed where curtains were or had been. (Noted on channel sides).
- BWR-2 and BWR-3 plants which do not have lower core plate bypass holes do not exhibit in-core instrument or channel wear.

3. DESCRIPTION OF CORE SUPPORT PLUG

3.1 CRITERIA

The following criteria were applied to the design of the core support plug:

- a. The core support plugs were originally designed to fit into and limit flow through the bypass flow holes to prevent jet impingement on the temporary control curtains.
- b. The plug must be able to withstand an operating pressure differential of 23 psi as well as a worst-case differential pressure of 32 psi corresponding to the peak accident differential pressure.
- c. Once installed, the plug must be held securely in place until it is permanently removed.

3.2 MECHANICAL DESIGN DESCRIPTION

The plug consists of five basic parts, as shown in Figure 3-1. Identical plugs have previously been installed at Vermont Yankee, Pilgrim, and KKM. The body provides a means of guiding the device into the bypass flow holes as well as a shoulder to support the plug and form a seal against water flow. The shaft extends through the body. A knob is provided at the top of the shaft to provide a means of grabbing the plug during installation and extraction. At the bottom, the latch is attached to the shaft by a pin. The latch is free to rotate during installation. The spring acts against the body and shaft during normal operation to provide the force necessary to offset the pressure differential acting on the body.

During installation, the plug has its latch rotated 90 degrees from its installed position and is withdrawn and locked in the body. The shaft is gripped by the installation tool, and the plug is inserted into the bypass flow holes. The body engages the rim of the hole. The shaft is pushed to its full extension, thus lowering and unlocking the latch below the underside of the core plate. The latch then rotates 90 degrees and bears on the bottom of the core plate. After insertion, the plug is pulled with about 30-pound force to test the placement.

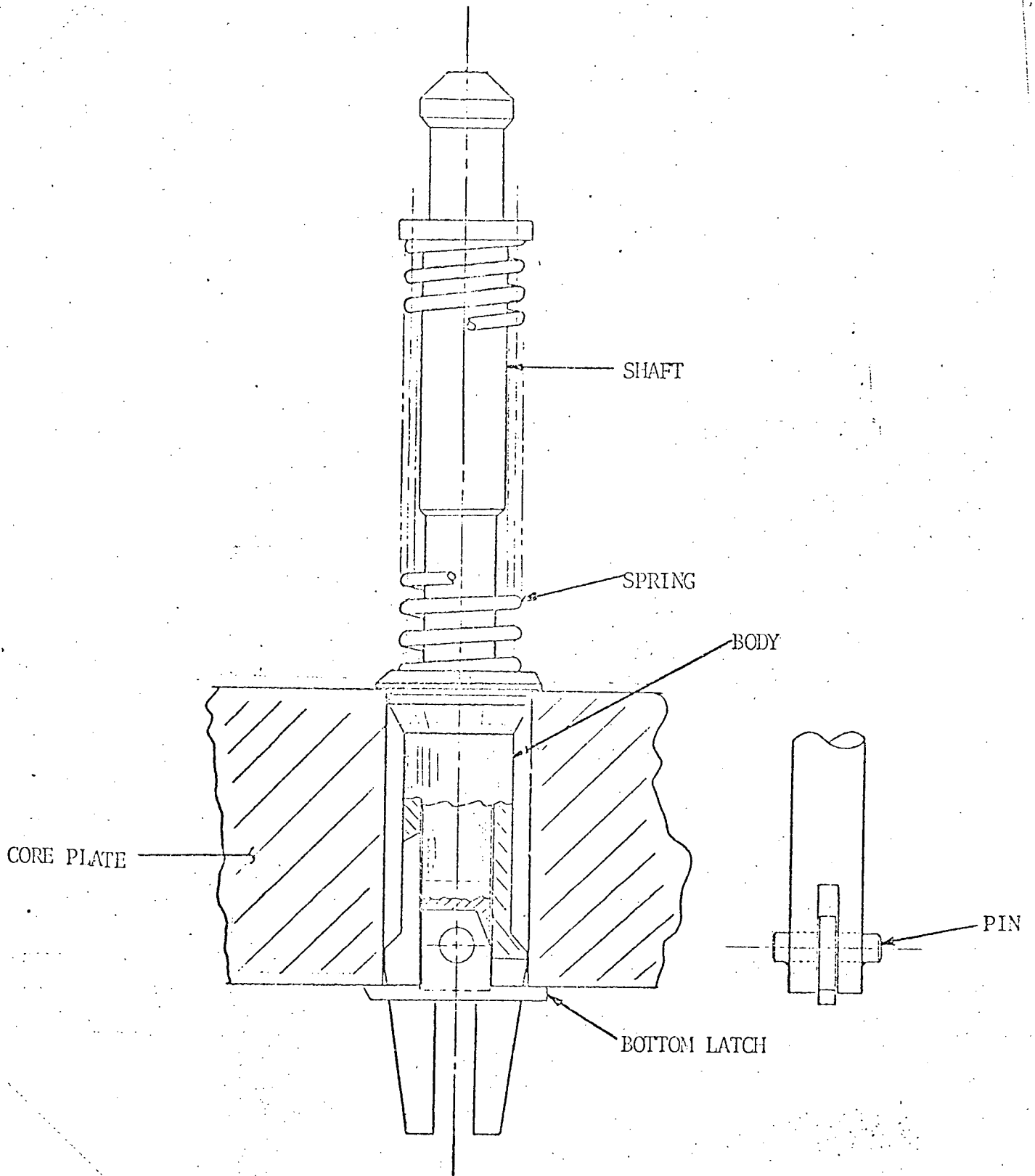


Figure 3-1. Plug Installed in Core Plate

The plug can be removed by gripping the top of the shaft with an extracting tool and applying a force of about 500 pounds. The latch's legs will be plastically deformed and the entire plug withdrawn. The plugs previously installed at Vermont Yankee were removed with no abnormalities of loose pieces reported. The force required for removal varied from 500 to 1300 pounds.

3.3 MATERIAL DESCRIPTION

The spring material is Inconel-X-750, spring-temper-conditioned with age-hardened heat treatment. All other parts are Type-304 stainless steel.

4. MECHANICAL ANALYSES

4.1 STRESS AND FATIGUE ANALYSIS

4.1.1 Summary

A stress analysis was performed on the plug in the core support plate. Normal operating conditions, pressure and thermal transients, and installation/removal operations were considered in the analysis. The results show acceptable stress levels in all plug components during normal operation and pressure and thermal transients. Plug assembly cycles produce extreme fibre torsional shear stresses in the spring near yield strength. Spring tests show that a loss of spring free length results from the assembly cycle but is predictably small and will not detract from the functional adequacy of the spring.

Some load relaxation will be experienced after plug installation. Elevated temperature testing has shown this relaxation to be a small percentage of the total preload. Creep in the stainless steel latch was experimentally investigated and was found to have negligible effect on plug preload. The combined effect on plug preload of the plug assembly cycles, the spring relaxation, and the latch creep will result in a final operating pre-load margin of nearly three times the operating static pressure differential across the plug, and a margin of 2 over the worst-case transient static pressure differential.

The analysis for the load required to extract the plug from the core plate shows that the assembly will maintain its integrity during this operation.

4.1.2 Component Stress

The Inconel-X-750 compression spring, with an initial maximum free length of 4.45 inches, is compressed prior to plug installation to a length of 1.88 inches, producing an elastically calculated, extreme fibre torsional shear stress including transverse shear effects of 156 ksi. This stress level is near

the yield strength of the spring and will produce a slight redistribution of stress across the wire section. In addition, the springs are compressed to the 1.88-inch length during plug assembly and during installation. In service, the spring is compressed from the 4.45-inch free length a maximum of 1.667 inches, producing an extreme fibre torsional shear stress of 112.2 ksi which includes the transverse shear effect and the stress concentration caused by the curvature of the coil. The in-service spring compression produces a maximum preload of 52.0 pounds. This preload is reacted in compression between the body and the core plate top surface, and in tension through the shaft, pin, and latch, the latch then bearing against the core plate bottom surface. Component stresses due to the preload and the normal operating pressure are as follows:

- o Bearing stress/body on core plate = 552 psi
- o Tensile stress/shaft minimum section = 1689 psi
- o Bearing stress/pin in shaft hole = 844 psi
- o Shear stress/pin in shaft hole = 925 psi
- o Bending stress/spring on shaft retainer = 3305 psi
- o Pin bending stress = 2058 psi
- o Pin average transverse shear stress = 542 psi
- o Bearing stress/pin in latch hole = 2499 psi
- o Tensile stress/latch minimum area = 2222 psi
- o Shear stress/pin in latch hole = 1623 psi
- o Bearing stress/latch on core plate = 2179 psi
- o Bending stress/latch leg = 23,126 psi
- o Latch average transverse shear stress = 2248 psi

The preceding stress values include the effect of a 23 psi pressure differential across the plug, representing rated flow conditions. A worst-case differential pressure across the plug of 32 psi corresponding to an accident will result in stresses 3% higher than those shown, or in the case of the latch leg, bending stress of 23,807 psi. Unless the differential pressure across the net area of the body exceeds the spring preload, the additional force in the spring due to the

pressure approaches zero. For the 32-psi differential pressure, the force on the body is 19 pounds. For a minimum preload of 37.5 pounds, which is based on the minimum 4.2-inch free length, the preload will resist this worst-case static pressure differential with no plug lifting or increased spring loading. A creep test of the latch at 550°F with a constant 46 pounds applied load showed negligible latch creep and therefore, negligible contribution to preload relaxation.

To remove the plug assembly from the core plate, a force is slowly applied to the top of the shaft; the latch legs, which are the most highly stressed area along the load path, bend plastically inward, allowing the assembly to be extracted from the hole in the core plate. Extraction tests have shown that a 490-pound load is required. This load is a factor of 9 above the combined maximum preload plus operating differential pressure load used to derive the stresses listed above. After increasing the applicable stresses by 9, it can be seen that the elastically calculated latch leg stress exceeds the ultimate strength, while the maximum primary stress in the remaining locations (shear tear-out stress at the latch hole) is limited to less than the yield strength. These results have been demonstrated in the extraction tests wherein the stainless steel latch legs deform plastically and no evidence of yielding is found at other locations.

4.1.3 Allowable Stresses

Tests of the Inconel-X-750 spring material (spring temper condition with heat treatment at 1200°F for 4 hours and air cooled) show an ultimate tensile strength of 264 ksi. Based on Huntington Alloy Products Division data for compression springs of this material and heat treatment, a yield strength in shear at 550°F of 158 ksi has been established. This value is based on shearing yield's being 70% of tensile yield, and tensile yield's being 90% of tensile ultimate.

For the annealed Type-304 stainless (ASTM A276) at 550°F, the ASME Section III S_M value is 16.9 ksi, $1.5 S_M$ is 25.35 ksi, and the S_Y is 18.8 ksi.

4.1.4 Comparison of Calculated and Allowable Stresses

The operating shear stress in the Inconel-X-750 spring was shown to be 112.2 ksi. This is the maximum level prior to any spring load relaxation and remains essentially unchanged during pressure transients. This level represents 71% of the shearing yield and is in the range of maximum stress at temperature as recommended by the International Nickel Company, Huntington Alloy Products Division.

The maximum operating stress in the Type-304 stainless components occurs in the latch legs where the primary bending intensity is 23,126 psi. During worst-case pressure transients, this stress has been shown to increase to 23.8 ksi. The 1.5 S_M limit on primary bending of 25,350 psi is met in both cases.

4.1.5 Fatigue

No significant plug component stress cycles can be identified for use in a fatigue evaluation. As previously shown, pressure transients produce negligible stress cycles. The most severe hypothetical temperature distribution that can reasonably occur in the assembly produces negligible thermal stresses (core plate at 550°F, plug assembly at 100°F). Stresses produced as a result of the dissimilar spring and plug materials are also negligible. As discussed in Section 4.2, Vibration, no plug vibration is to be expected.

4.2 VIBRATION OF PLUGS

4.2.1 Pump Pulsations

Recirculation pump shaft frequency at rated conditions is about 28 Hz (1670 rpm). This frequency is not observed in the reactor. The frequency observed is blade frequency and vane frequency, which are five times the shaft frequency or about 140 Hz. The natural frequency of the plug body oscillating vertically on the spring has been calculated at 38 Hz. Since the pump pulsation frequency is so far removed from the body frequency, body resonance is precluded.

4.2.2 Self-Excited Motion

Pressure difference = 23 psi (normal operation, across plug)

Net plug area = 0.59 in.²

Therefore, force on body = 13.6 lb

If the body should vibrate at its 38 Hz natural frequency the 13.6 pound force on the body would fluctuate with a half-sine shape since the body can move upward only.

A Fourier decomposition shows the first periodic Fourier force amplitude is 6.8 pounds, with period $T = 1/38$ or the body natural period. This is the primary component of the oscillatory pressure equivalent to the 13.6 pound half-sine force. The minimum spring preload downward on the body is 37.5 pounds or 5.5 times the equivalent periodic Fourier force amplitude. Therefore, a load amplification factor of 5.5 or greater in the body/spring system would be required for self-excited motion.

Note that this amplification factor requirement represents a lower bound solution in that the most unfavorable conditions of the flow are assumed. The major assumptions are: (a) when the plug body is lifted from the core plate, differential fluid pressure acting on the body immediately relieves to zero, and (b) that the existing leakage is neglected. If assumption (a) is not fulfilled, (i.e., if the differential pressure does not diminish to zero), the net oscillating force acting on the body will be less than 13.6 pounds and the first Fourier component will be less than 6.8 pounds. Therefore, a system amplification factor of greater than 5.5 is required to sustain the self-excited motion. Also, with the present leakage, the pressure difference across the core plate at the bypass flow hole becomes smaller due to the transformation of the pressure head into velocity head. In this case, the amplification factor has to be greater than 5.5 for self-excited motion.

Also note that since the net force due to pressure acting on the plug (13.6 pounds) is less than the preload (37.5 pounds), to initiate the self-excited motion a proper initial disturbance is required. This can be in the form of a sudden large force to overcome the preload, or a proper dynamic excitation with proper frequency and amplitude to overcome the preload. However, once the preload is overcome initially, the force due to pressure difference with a sufficiently large (>5.5) amplification factor of the system will suffice to maintain the motion as self-excited.

4.2.3 Conclusions

No self-excited motion is observed in the Full-Scale Mock-Up Test Facility; i.e., no initial disturbance large enough to initiate self-excited motion exists in the simulated condition. An examination of pressure expected in the reactor under normal and upset operational conditions reveals no disturbance of sufficient magnitude to initiate self-excited motion. Therefore, it is concluded that no such motion can occur for in-reactor service.

4.3 WEAR

A room-temperature flow test of a full-scale prototype plug assembly was conducted to assess vibration-wear conditions that would occur during reactor service. The plug assembly was inserted through a nominal 1-inch-diameter hole in a 2-inch-thick plate in the full-scale flow test facility fabricated to simulate the hole geometry of the reactor pressure vessel core support plate.

The flow was adjusted to give a pressure differential across the test assembly of 17 ± 1 psig. The leakage past the plug was 0.4 gpm. Test duration was 24 hours.

Monitoring of the plug during the flow test disclosed no evidence of vibration. Measurement of plug components before and after the flow test disclosed no evidence of wear; thus the flow test results are consistent with the conclusions of Subsection 4.2.3 and project satisfactory performance of the plug assembly for reactor service.