8005120216

NUREG-0641

Control Rod Guide Tube Wear in Operating Reactors

Operating Experience Report

Prepared by R. Riggs

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

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Manuscript Completed: December 1979
Date Published: April 1980

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ABSTRACT

Evidence of control rod guide tube wear has been observed in operating pressurized water reactors. The cause of this wear is identified as flow-induced vibration of the control rods. Repetitive contact of the control rod cladding material against the softer Zircaloy guide tube causes wear degradation of the guide tube. The wear is local and confined to the axial position of the control rod tips that are "parked" in the fully withdrawn position.

Guide tube surveillance and inspection results obtained from various PWR facilities indicate that significant guide tube wear is isolated to the facilities designed by Combustion Engineering.

This report describes the measures being taken by both the industry and the NRC to deal with this matter.

The staff presents its technical positions and requirements for continuing programs to provide confirmatory evidence supporting final resolution of this wear for all PWR facilities. This report describes the staff's positions supporting continued operation of the plants as of December 1979 pending completion of this generic effort.

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ACKNOWLEDGMENT

I would like to acknowledge the technical efforts of H. Levin and R. Johnson during the early stages of this review. In addition, the coordinating link in the staff review and interfaces with the licensees and vendors is credited to E. Conner. Last, but certainly not least, I acknowledge the valued suggestions by F. Coffman and M. Singh on the organization of this report, and the typing effort of P. Jackson.

NOMENCLATURE

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
BG&E	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combustion Engineering
CEA	control element assembly
CEDM	control element drive mechanism
CRA	control rod assembly
CRGT	control rod guide tube
ECT	eddy current test
EFPH	effective full-power hours
FP&L	Florida Power and Light Company
GTA	guide tube assembly
ID	measure of inner diameter
LOCA	loss-of-coolant accident
MY	Maine Yankee Atomic Power Company
NDT	nondestructive test
NNECO	Northeast Nuclear Company
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OD	measure of outer diameter
OPPD	Omaha Public Power District
RCCA	rod cluster control assembly
SSE	safe shutdown earthquake
UEF	upper end fitting

CONTROL ROD GUIDE TUBE WEAR IN OPERATING REACTORS

1. COMBUSTION ENGINEERING NSSS FACILITIES

1.1 Introduction - Chronology of Control Rod Guide Tube Wear

1.1.1 Initial Indications of Control Rod Guide Tube Wear

On December 13, 1977, during the first refueling outage, cracks were observed at Millstone Unit 2 of Northeast Nuclear Company (NNECO) in the guide tubes of a number of fuel assemblies. The affected fuel assemblies during this first cycle of operation were all under control element assemblies (CEAs).

Appendix A gives a detailed chronology of the immediate actions taken by NRC, the licensees, and Combustion Engineering (CE) following these observations. The chronology in Appendix A encompasses the time period from the initial indications of control rod guide tube wear in the CE nuclear steam supply system (NSSS) facilities to the first NRC approval of the sleeving modification as a corrective action to prevent further wear. The more general chronology described in this report provides an overview of the major actions taken from the initial observations to the current time period.

1.1.2 Short-Term Corrective Action for Continued Operation

On December 20, 1977, NRC notified all operating CE NSSS facilities to (1) insert 1/7 of all CEAs not fully inserted at least 10 steps each day, and (2) provide a request for amendment to allow CEA insertion beyond the present "full out" position. The basis for this short-term corrective action was predicated on the judgment that repositioning the control rods would minimize the potential for further local wear of the guide tubes and improve the assurance that control rods would be capable of scram.

The licensees of the affected plants designed by CE submitted their requests for amendments in early January 1978. The amendments involved a change of the Technical Specifications to permit the control rods to be inserted 3 inches further into the core. NRC approvals (Refs. 1-5) of the amendments submitted for Calvert Cliffs Units 1 and 2, Fort Calhoun, Maine Yankee, and St. Lucie Unit 1 were issued on January 6, 1978, authorizing continued plant operation with the 3-inch insertion of the control rods.

Subsequent to the above mentioned approvals, NRC set 20-day letters (Refs. 6-10) to the licensees and, in particular, to Baltimore Gas and Electric Company (BG&E), Omaha Public Power District (OPPD), Maine Yankee Atomic Power Company (MY), and Florida Power and Light Company (FP&L). The letters requested justification that excessive guide tube wear did not exist or, if unable to assure that such wear did not exist, justification that continued operation of the facility would not create undue risk. NNECO was sent a similar letter requiring additional justification for return to power.

The licensees' responses to the 20-day letters provided the licensees' analyses and evaluations related to the guide tube wear and the 3-inch CEA insertion requirement. The responses supported continued operation at Calvert Cliffs Unit 2, Maine Yankee, and St. Lucie Unit 1 and for the return to power with sleeving modifications incorporated at Calvert Cliffs and Millstone Unit 2. Inspection results (Ref. 11) from the Fort Calhoun facility operated by OPPD indicated no significant wear of the guide tubes.

Based on the results of guide tube inspections performed by OPPD and the differences in the Fort Calhoun plant design, as discussed in Section 1.5, the staff concluded that the excessive guide tube wear observed at other CE facilities did not exist at the Fort Calhoun plant (Ref. 12). Therefore, OPPD was granted approval to operate with the CEAs fully withdrawn. Section 1.4 provides a discussion in greater depth of the operating experience and inspection results of the CE facilities.

1.1.3 Cause and Location of Control Rod Guide Tube Wear

In the interim and ensuing time periods, NRC held meetings with the affected licensees and CE to discuss the cause of the guide tube wear, inspection results, and CE's proposed modifications to remedy the guide tube wear for subsequent cycles of operation. The cause of the wear was attributed to flow-induced vibration of the CEAs' Inconel-clad control rods that were rubbing against the Zircaloy guide tubes. Inspection results revealed that the maximum guide tube wear occurred in the upper elevation of the fuel assemblies corresponding to the location of the control rod tips in their full-out positions (see Figure 1). In general, the wear at this elevation was oriented to one side of the guide tubes with no apparent preferential orientation within the assemblies (see Figure 2).

1.1.4 Corrective Action to Prevent Control Rod Guide Tube Wear

To prevent guide tube wear, the licensees and CE proposed a sleeving modification. The modification involved the installation of chrome-plated stainless steel sleeves in the upper 22 inches of the guide tubes. The sleeving that was installed in new (reload) and unworn fuel assemblies provided protection of the Zircaloy guide tube from the wearing action of the CEA control rod tips. The sleeves that were installed in certain worn fuel assemblies served to restore lost structural margin resulting from the previous loss (wear) in the guide tube cross-sectional area. Section 1.9 describes the sleeving modification and installation procedures. Because the sleeving modification was not expected to eliminate the source of the wear (vibration of the CEAs), CE also proposed a limited number of additional demonstration assemblies that modified the flow characteristics of the coolant. Even with the placement of a limited number of demonstration assemblies in certain reactors, the complex nature of the flow-induced vibration of the CEAs required additional testing and evaluation. This approach to a final design solution was to provide a better understanding of the specific mechanism(s) exciting the CEA vibrational response. These long-range final design solutions and CE's investigations into the approach are discussed in Section 1.7.

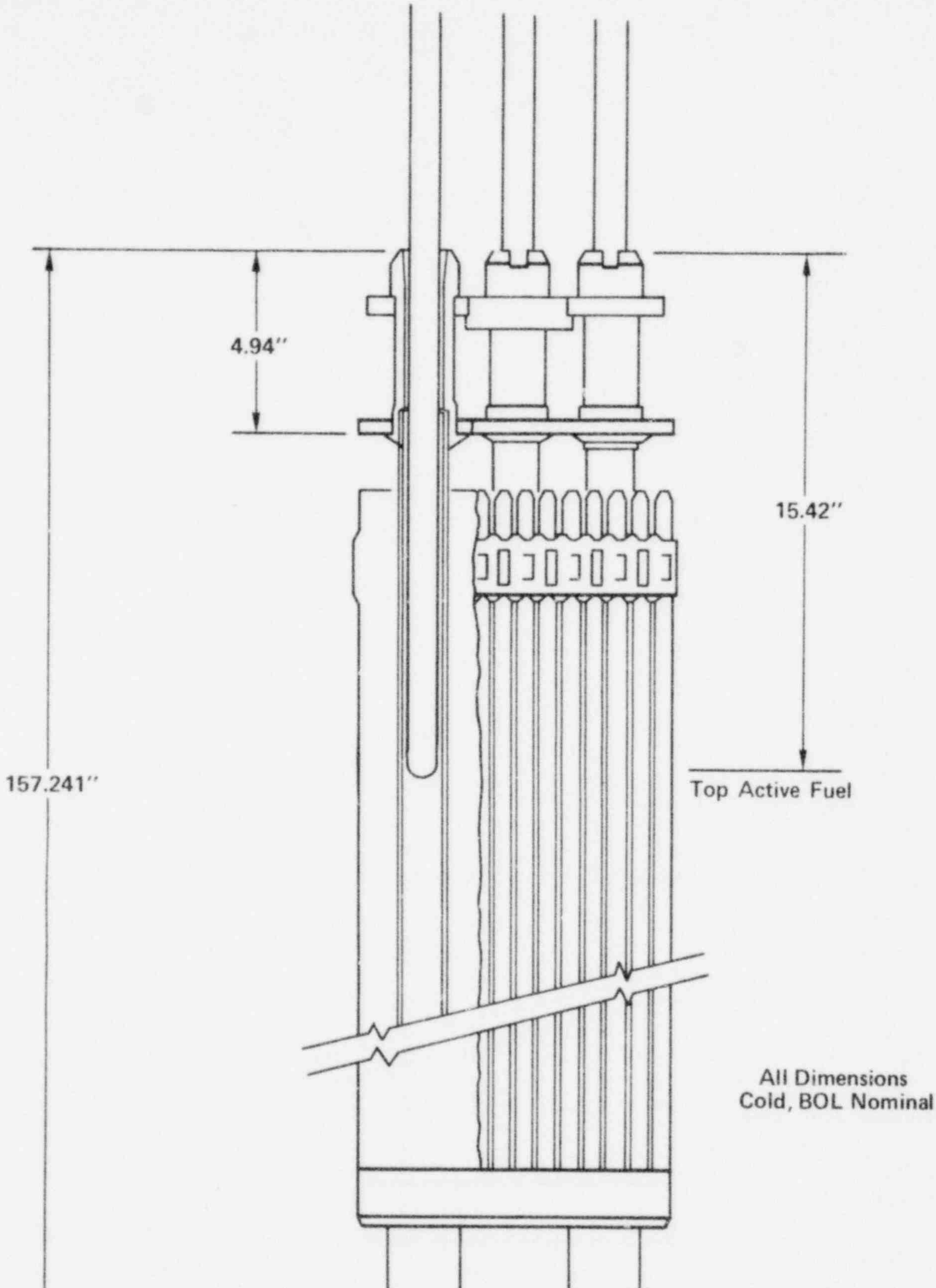

Figure 1. Millstone Unit 2 fuel assembly at ''parked'' position.

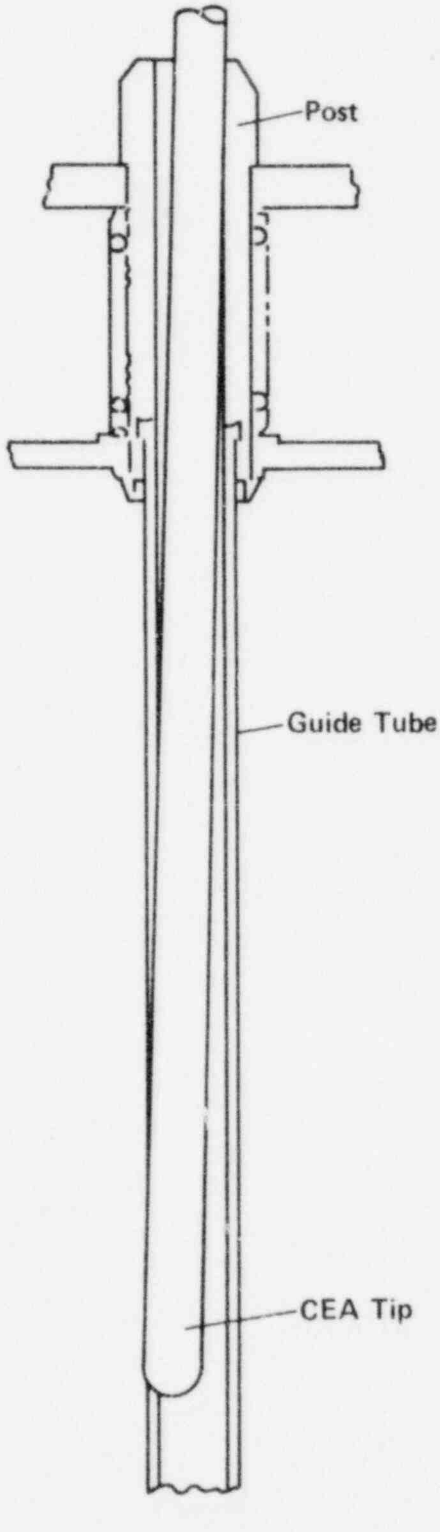

Figure 2. Representation of CEA/guide tube interference.

1.1.5 NRC Approvals and Reports for First Cycle Corrective Action

Table 1 lists in chronological order the NRC approvals for use of the sleeved guide tubes and demonstration assemblies in CE-designed operating reactors.

Table 1. NRC Approvals For First Cycle Corrective Action

Plant	Cycle	Amendment Number	Date of NRC Approval	Reference
Calvert Cliffs Unit 1	3	32	03/31/78	13
Millstone Unit 2	2	38	04/19/78	14
St. Lucie Unit 1	2	27	05/26/78	15
Maine Yankee	4	40	08/18/78	16
Calvert Cliffs Unit 2	2	18	10/21/78	17

On April 7, 1978, the NRC staff presented to the ACRS a critique of the guide tube wear phenomenon and the proposed remedies. The staff also identified the guide tube wear phenomena as a reportable abnormal occurrence in their April/June 1978 Report to Congress.

The above NRC approvals, for use of the sleeved guide tubes and the use of flow-modifying demonstration assemblies, were given for single cycles of operation in the affected plants. Continued use of these modifications for each plant is subject to acceptable inspection results at the end of each cycle of operation. The licensees are required to submit their planned surveillance program to the NRC prior to the refueling outage. Implementation of these programs requires NRC approval.

1.1.6 Results of First Cycle Corrective Action

The first opportunity to evaluate the performance of the sleeved guide tubes after reactor operation occurred during the Millstone Unit 2 refueling outage in the spring of 1979. Subsequent to the Millstone Unit 2 refueling, St. Lucie Unit 1 and Calvert Cliffs Unit 1 also provided additional evidence of the performance of the sleeved guide tubes. Based on the results of these inspections, the sleeving modification has performed well as a corrective measure to mitigate the guide tube wear.

Calvert Cliffs Unit 2 was scheduled for refueling in October-November 1979. This unit, in addition to the sleeving modification, has 16 demonstration assemblies with modifications designed to affect the coolant flow and perturb the vibrational characteristics of the control rods. The staff-approved surveillance program for Calvert Cliffs Unit 2 was witnessed by onsite staff observations of the fuel assembly inspections. "· ne Yankee is scheduled for refueling in January 1980. The Maine Yankee core, in addition to the sleeving modifications, has four demonstration assemblies. The four demonstration assemblies consist of two different flow-modifying designs (two of each). Submittal of Maine Yankee's surveillance program for staff approval is expected in December 1979.

1.1.7 NRC Approvals for Second Cycle Corrective Action

Table 2 lists the NRC approvals given to plants that allow second cycle operation with sleeved guide tubes and use of new fuel with sleeved guide tubes.

Table 2. NRC Approvals For Second Cycle Corrective Action

Plant	Cycle	Amendment Number	Date of NRC Approval	Reference
Millstone Unit 2	3	52	05/12/79	18
St. Lucie Unit 1	3	32	05/27/79	19
Calvert Cliffs Unit 1	4	39	06/14/79	20

The NRC staff has maintained and will continue to maintain close liaison with representatives of the licensees and CE on this issue and any related problems. All the licensee-proposed inspection programs will be reviewed prior to taking action at any facility. The staff has required that all inspection programs continue to be submitted for review well in advance of refueling shutdowns.

1.2 Design and Functions of Guide Tubes and Related Components

1.2.1 Combustion Engineering Fuel Assembly

The basic CE fuel assembly design discussed in this report (see Figure 3) consists of a 14x14 array. The various types (batches) of fuel assemblies mentioned herein (A, B, C, etc.) contain variations in the number of burnable poison rods within this 14x14 array design.

The structural frame of the CE fuel assembly consists of the guide tubes, spacer grids, and end fittings. Four outer guide tubes are mechanically attached to the stainless steel end fittings, and the spacer grids are welded to all five guide tubes. The Zircaloy-4 guide tubes are 1.115 inches in outside diameter and have a 0.040-inch wall thickness. The guide tubes serve functionally in a dual capacity as the principal structural members of the fuel assembly and as channels to guide the control rods during insertion.

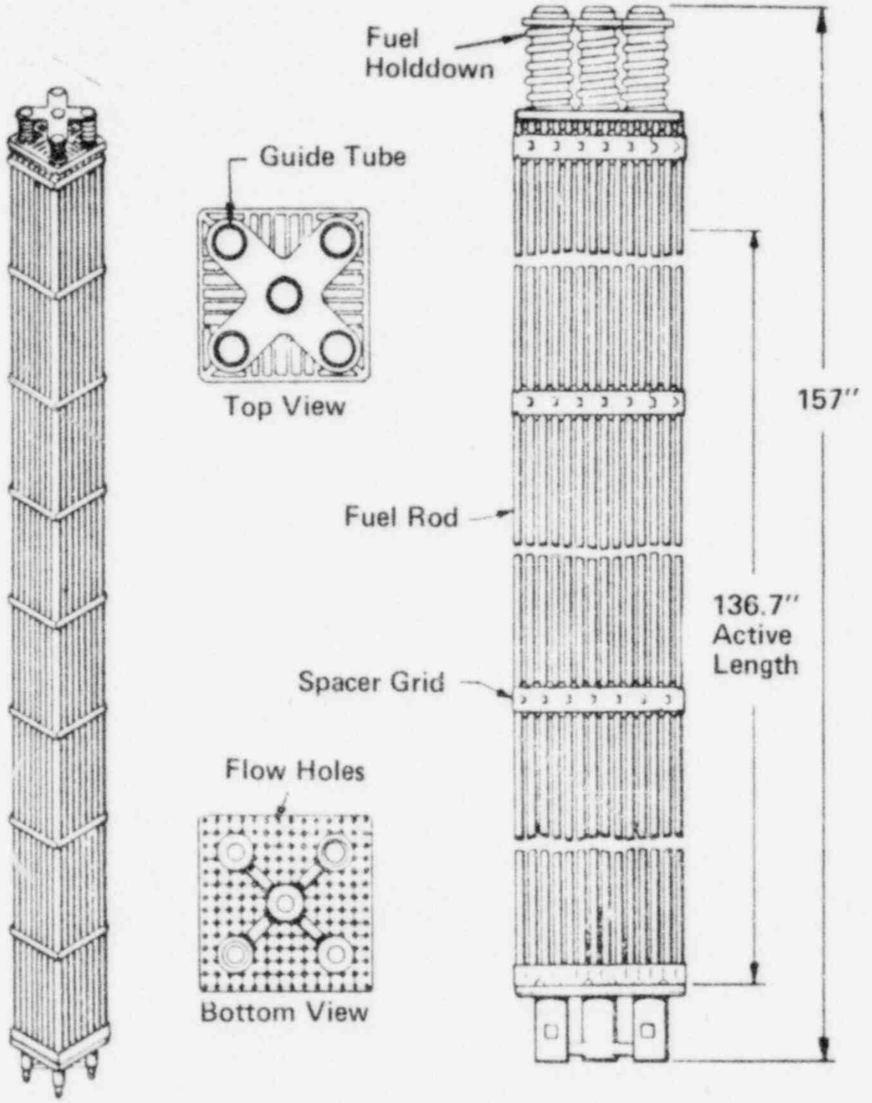

Figure 3. Combustion Engineering fuel assembly.

The guide tubes also function as coolant conduits for the CEAs. The coolant required to cool the control elements flows in the annulus between the control element and the guide tube, and then into the region outside the control element assembly shrouds. A similar but smaller leakage occurs around the flow restriction at the upper end of those guide tubes without control elements. In addition, the buffer system used to slow down the CEAs at the end of a scram stroke is accomplished by the guide tubes that have a reduced diameter in the lower section. When a CEA falls into the buffer region, the hydraulic pressure buildup in the lower guide tube supplies the force to slow down the CEA. The CEA velocity is thus decreased to yield an acceptable impulse/momentum exchange. The above design parameters are optimized to establish the best combination of buffer stroke and buffer annulus.

The fuel assemblies are designed to maintain their structural integrity under steady-state and transient operating conditions as well as under normal handling, shipping, and refueling loads. The design must take into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and CEA guide tubes, irradiation effects, and wear of all components. Mechanical tolerances and clearances have been established on the basis of the functional requirements of the components. All components must be highly resistant to the corrosive action of the reactor environment.

1.2.2 Combustion Engineering Control Element Assembly

The CEAs (Figure 4) consist of five Inconel tubes that are 0.948 inch in outside diameter and contain boron carbide and Ag-In-Cd. Four tubes are assembled in a square array around the central fifth tube. The tubes are joined by a spider at the upper end. The hub of the spider couples the CEA to the drive assembly.

The CEAs permit rapid changes in reactivity, as required to trip the reactor and to compensate for changes in moderator density and fuel temperatures associated with changes in power level. Therefore, reactivity control is achieved by operational maneuvering of single or double CEAs. The double CEA is made up of two single CEAs connected to separate grippers and carried by an extension shaft. The CEAs are used both for shutdown and for regulation. The CEAs designated for shutdown are divided into separately controlled groups; those designated for regulation are also divided into separate groups. During power operation, the shutdown groups are fully withdrawn while the position of the regulating groups is adjusted to meet reactivity and power distribution requirements. All CEAs, except the part-length CEAs (which have been removed), drop to a fully inserted position upon reactor trip.

The CEAs must be designed to maintain their structural integrity under all steady-state and transient operating conditions, and under handling, shipping, and refueling loads. Thermal distortion, mechanical tolerances, vibration, and wear must be accounted for in the design. Clearances and corresponding fuel assembly alignment are established so that possible stackup of mechanical tolerances and thermal distortion will not result in frictional forces that could prevent reliable insertion of the CEAs. Under normal design loadings plus design earthquake forces of normal operating loadings plus hypothetical earthquake forces, deflections are limited so that the CEAs can function and adequate core cooling is maintained. Under normal operating loadings plus hypothetical earthquake forces plus pipe rupture loadings, the deflection

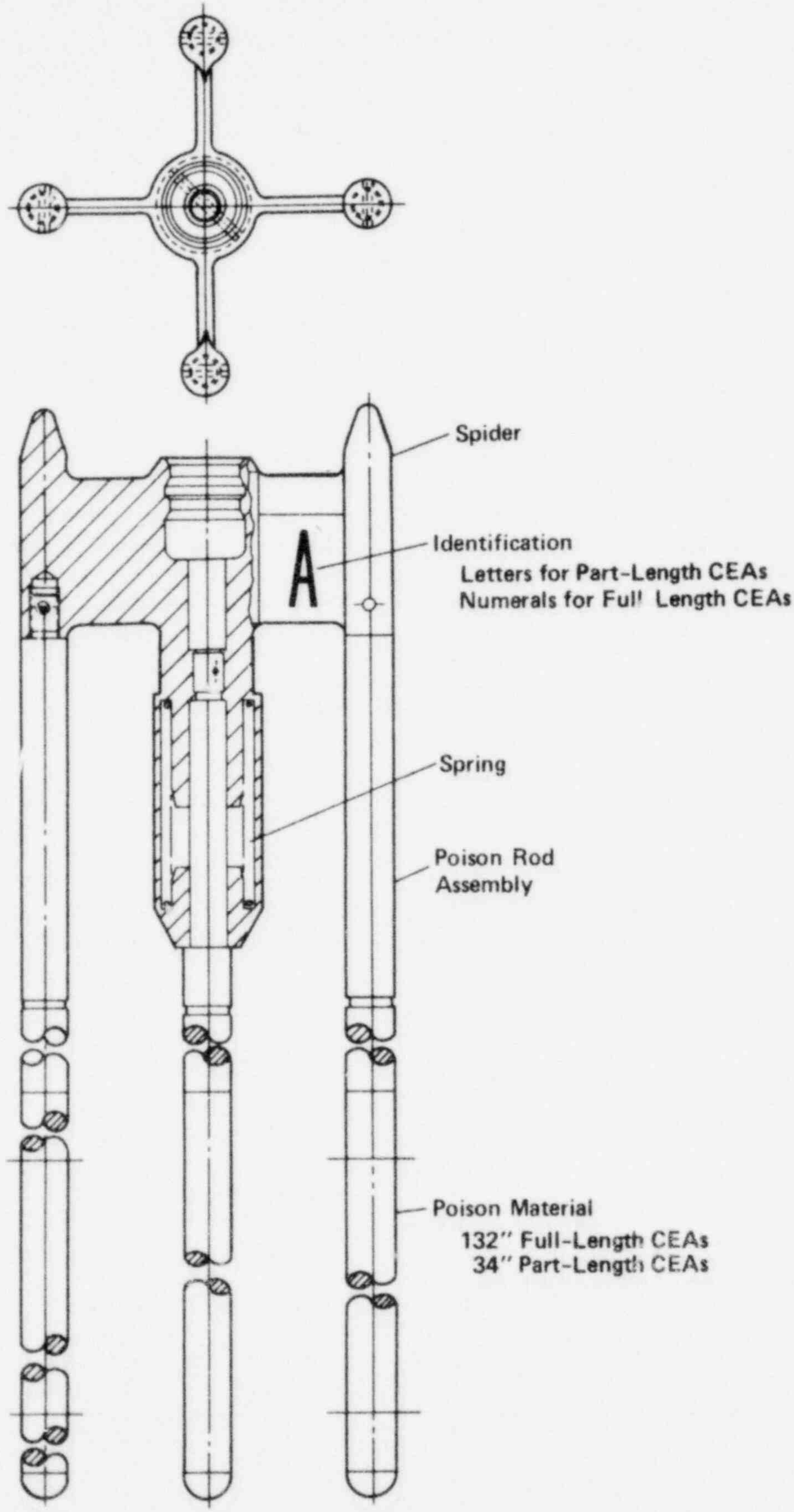

Figure 4. Combustion Engineering control element assembly.

design criteria depend on the size of the piping break. If the equivalent diameter of the pipe break is no larger than the largest line connected to the main reactor coolant lines, deflections are limited so that the core is held in place, the CEAs function normally, and adequate core cooling is maintained. Those deflections that could influence CEA movement are limited to less than the deflection required to prevent CEA function. For pipe breaks larger than those previously described, the criteria require the fuel to be held in place in a manner that permits core cooling and that maintains adequate coolant flow passages. For these major pipe break sizes, CEA insertability is not required to achieve shutdown since the rapid voiding during the ensuring blowdown and the subsequent refill with the borated safety injection water ensures adequate shutdown margin for the reactor. For the larger break sizes, critical components are restrained from buckling by further limiting the stress levels to two-thirds of the stress level calculated to produce buckling.

1.2.3 Combustion Engineering Control Element Drive Mechanism

The control element drive mechanism (CEDM) for the CE reactor designs subject to the control rod guide tube wear (see Section 1.5) is of the magnetic jack type of drive. Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its stroke. The control element drive mechanisms (CEDMs) must be capable of actuating the CEAs under steady-state and transient operating conditions and during hypothetical earthquake loadings. For pipe rupture accident loads, the CEDMs are designed to support and maintain the position of the CEAs in the core and to be capable of operation when these loads have diminished. The criteria for insertability of CEAs are dependent upon the size of the reactor coolant pipe rupture (see Section 1.7). The speed at which the CEAs are inserted or withdrawn from the core must also be consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CEDM power circuit breakers open to allow the CEAs and the connecting CEDM components to drop by gravity into the core. The reactivity is reduced during such a CEA drop at a rate sufficient to prevent exceeding fuel damage limits. Severe distortion, or mechanical interference resulting from holes in the guide tubes, would impede the gravity drop feature discussed above.

1.2.4 Combustion Engineering Upper Guide Structure Assembly

The upper guide structure assembly for the Maine Yankee core (Figure 5) shows the typical packing of the single and dual CEAs described in Section 1.2.2.

The upper support structure consists of control element assembly shrouds, a fuel assembly alignment plate, and an expansion compensating ring. The upper guide structure assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, prevents fuel assemblies from being lifted out of position during a severe accident transient, and protects the CEAs from the effect of coolant crossflow in the upper plenum (Figure 6). The control element assembly shrouds extend from the fuel assembly alignment plate to above the support plate. The single-type shrouds and dual-type shrouds in configuration consist of two single-type shrouds connected by a section shaped to accommodate the dual control element assemblies.

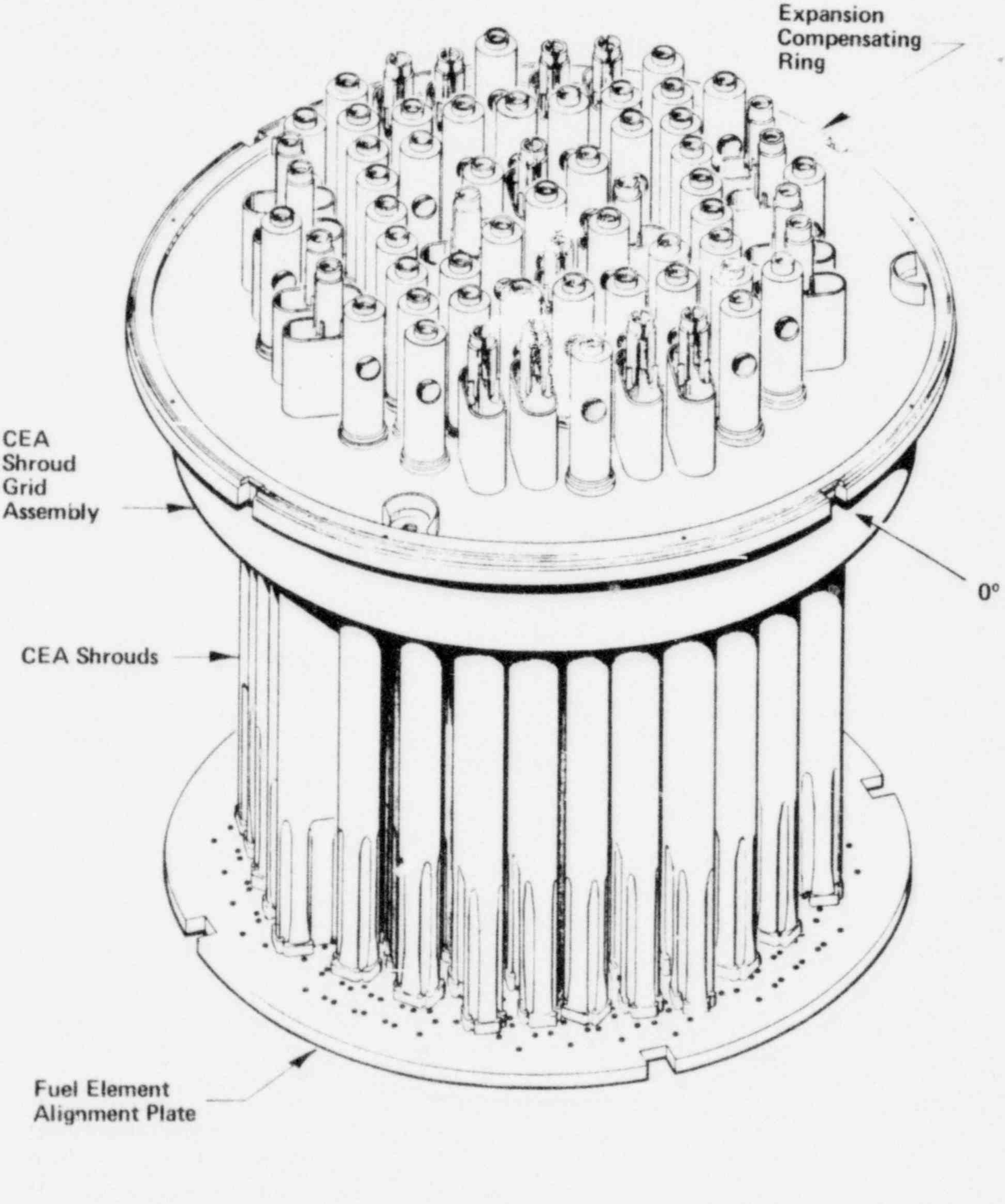

Figure 5. Maine Yankee upper guide structure assembly.

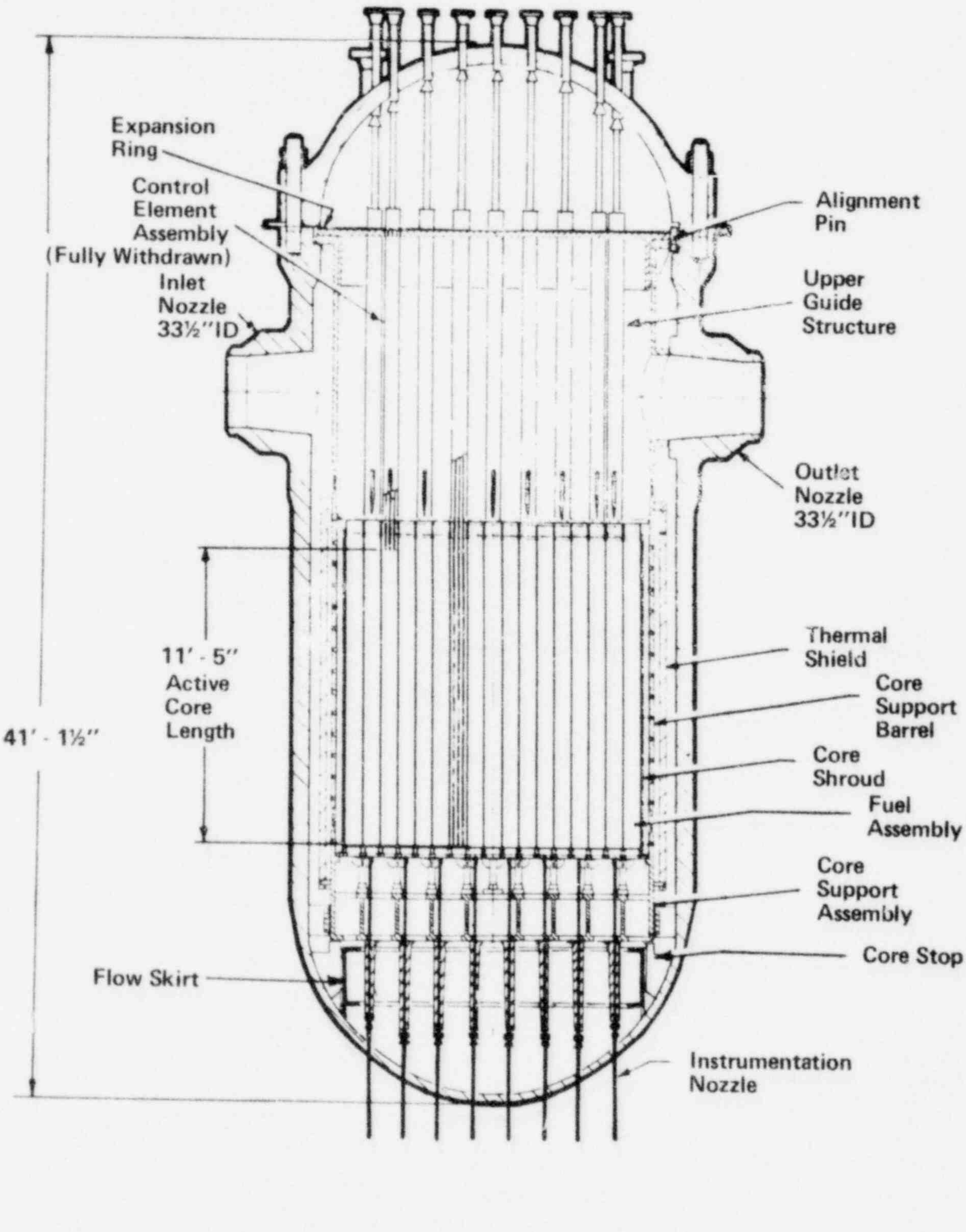

Figure 6. Maine Yankee (typical) reactor vertical arrangement.

It is the absence of lateral support from the CEA shroud, as well as the turbulent coolant flow as the coolant exits the fuel assembly into the shroud region, that sets up the vibration of the CEA control rods in the fully withdrawn position. This basic design difference between the CE design and other vendor NSSS designs is further discussed in Sections 2.2 and 3.1.

1.3 Safety Considerations

Based on the design and functions of the components described in Section 1.2, the staff has determined that structural integrity of the guide tubes is required under both normal and accident conditions. This is necessary for the core to maintain its coolable geometry and for the control rods to scram as required by the safety analyses.

The design and functional performance of those components described in Section 1.2 are directly affected by the integrity and design of the control rod guide tubes. Therefore, this report will address modifications designed to mitigate the control rod guide tube wear observed in the affected CE reactor facilities and the impact of the modifications on the performance of those functions.

1.4 Operating Experience in Combustion Engineering NSSS Facilities

1.4.1 Guide Tube Wear Hours Under CEAs

Table 3 indicates the amount of time the various fuel types operated in core locations containing CEAs in all the subject CE reactors. The wear-hour data are presented in terms of effective full-power hours (EFPH). Actual time under power conditions, including power levels less than 100 percent, could be up to 25 percent longer.

1.4.2 Guide Tube Wear Measurements

To characterize and determine the extent of guide tube wear in the CE NSSS facilities, measurements on discharged fuel assemblies were made at Millstone Unit 2, St. Lucie Unit 1, Calvert Cliffs Unit 1, Maine Yankee, and Fort Calhoun using eddy current testing (ECT) techniques. Two ECT methods were developed by CE to quantify average cross-sectional wear and azimuthal wear, respectively. The ECT inspections were supplemented with visual observations using a borescope and/or a periscope. Table 4 lists the guide tube tests performed at each of the facilities.

Table 3. Wear Hours and Numbers of Assemblies Under CEAs

Plant	Fuel Type (Batch)	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Approximate Time (EFPH)
Millstone Unit 2	A	61	-	-	-	11,700
	C	20	-	-	-	11,700
St. Lucie Unit 1	A	61	-	-	-	9,200
	C	20	-	-	-	9,200
Calvert Cliffs Unit 2	A	57	-	-	-	13,200
	C	28	-	-	-	13,200
Calvert Cliffs Unit 1	A	33	-	-	-	13,200
	A	24	-	-	-	13,200
	B	-	44	-	-	6,600
	B	-	1	-	-	6,600
	C	20	-	-	-	13,200
	C	8	8	-	-	19,800
	D	-	20	-	-	6,600
Maine Yankee	A	5	0	-	-	3,500
	A	0	57	-		13,500
	B	50	-	-	-	3,500
	B	-	-	21	-	6,600
	B	4	-	4	-	10,100
	B	-	-	8	-	6,600
	C	-	12	-	-	13,500
	C	-	16	16	-	20,100
	C	-	-	16	-	6,600
	C	26	-	-	-	3,500
	G	-	-	12	-	6,600
	H	-	-	8	-	6,600
Fort Calhoun	A	1	-	-	-	7,400
	A	-	4	-	-	8,900
	B	36	-	-	-	7,400
	B	-	1	1	1	22,700
	B	4	4	-	-	16,300
	B	-	12	-	-	8,900
	C	4	4	-	-	16,300
	C	4	4	4	-	23,200
	C	-	-	4	-	6,900
	C	-	8	-	-	8,900
	C	-	4	4	-	15,800
	D	-	4	-	-	8,900
	D	-	4	4	-	15,800
	D	-	-	8	-	6,900
	D	-	-	12	12	13,800
	D	-	-	-	4	6,900
	E	-	-	12	12	13,800
	E	-	-	-	11	6,900
	F	-	-	-	9	6,900

Table 4. Eddy Current Testing At Licensed Operating Combustion Engineering Reactors

Plant	Average Wear Probe	Azimuthal Wear Probe
Millstone Unit 2	Yes	Yes
Maine Yankee	Yes	Yes
Calvert Cliffs Unit 1	Yes	Yes
Fort Calhoun	Yes	No
St. Lucie Unit 1	Yes	Yes
Calvert Cliffs Unit 2	Yes	Yes

1.4.3 Results From Guide Tube Wear Measurements

The results obtained from the eddy current tests and comparisons of the eddy current data with direct visual observations indicated the following:

(1) The average wear probe tests confirm that the maximum wear locations correspond to the elevation in the guide tubes where the tips of the CEAs were located during plant operation.

(2) The average wear probe provides an adequate screening technique to identify the most worn guide tubes.

(3) The azimuthal wear probe provides a conservative representation of the most worn area in an axial location within the guide tube. This conclusion was based on three comparisons: (a) azimuthal wear probe data with measured laboratory standards, (b) azimuthal probe data with the borescopic examinations of the same guide tubes, and (c) azimuthal wear probe data with the results of hot cell destructive examinations of worn guide tubes. In each instance where the azimuthal probe indicated a hole in the guide tube, the visual examinations confirmed that indication. In addition, the circumferential extent of wear indicated by the azimuthal probe is greater than that indicated by the borescopic visual examinations.

1.5 Combustion Engineering Conclusions on Guide Tube Wear

Based on a review of the azimuthal probe data on the most severely affected guide tubes and depending on the particular reactor under examination, a significant number of guide tubes that contained CEAs could have through-wall penetrations.

The Fort Calhoun plant operated by OPPD was an exception to the above. In this case, even though the Fort Calhoun plant had more cycles of operation, with some fuel assemblies operating for three fuel cycles under CEAs, the ECT measurements indicated no significant guide tube wear. The lack of significant guide tube wear has been attributed to a lower coolant flow rate, as discussed in Section 1.8, and basic design differences between the Fort Calhoun plant and the other CE NSSS facilities (see Table 5).

Table 5. Fort Calhoun Design Differences Affecting Guide Tube Wear

Parameters	Fort Calhoun	Other Plants
Active core length	Shorter core	--
CEA drive mechanism type	Rack and pinion	Magnetic jack
Assembly holddown mechanism	None	Springs
CEA guide tube annulus flow rate	Lower flow rate	--

1.6 NRC Conclusions on Guide Tube Wear

Both the favorable ECT results indicating no significant guide tube wear for up to three cycles of operation and the design differences between Fort Calhoun and other CE-designed facilities formed the basis upon which OPPD requested relief from the additional control rod insertion limits (see Section 1.5). Review of the submitted information led the NRC staff to conclude (Ref. 12) that the excessive guide tube wear observed at other CE NSSS facilities was unlikely at the Fort Calhoun facility. Therefore, on August 7, 1978, NRC rescinded (Ref. 12) the additional 3-inch control rod insertion requirement under which OPPD had operated in cycle 4 since January 7, 1978.

Based on results of the ECT wear measurement taken at Millstone Unit 2, St. Lucie Unit 1, Calvert Cliffs Units 1 and 2, and Maine Yankee, and additional out-of-reactor tests described in Section 1.8, CE and the licensees of these plants requested NRC approval to modify the worn and unworn fuel assemblies by installing stainless steel sleeves in the guide tubes. This modification, with the exception of a few assemblies located in low wear positions and a limited number of flow-modifying demonstration assemblies, was made to all fuel assemblies placed under CEAs for the next cycle of operation.

The sleeving modification is described in Section 1.7.2. As such, the sleeving modification was proposed as a corrective action directed at mitigating or eliminating guide tube wear caused by CEA vibration. Section 1.7.3 describes CE's investigation into design modifications and flow-modifying demonstration assemblies directed toward an alternate corrective action.

The operating experience from the licensed CE facilities using the sleeving modification will provide the data base upon which each subsequent cycle of operation will be evaluated. This continuing program and the confirmatory inspections to monitor the performance of the corrective actions are described in Section 1.9.

The staff has concluded that the above course of action provides an acceptable systematic approach to resolving the control rod guide tube wear in operating CE NSSS facilities.

1.7 Corrective Action

1.7.1 Short-Term Corrective Action

In early January 1978, CE licensees submitted their request for a change in the Technical Specifications that would permit the insertion limits of the control rods to be extended 3 inches farther into the reactor core.

One of the factors leading to the requested 3-inch CEA insertion was a concern that severe wear of the guide tube inside diameter could interfere with the motion of the CEA during scram. This concern, although prudent, was not supported by results from the Millstone Unit 2 reactor, which showed no significant scram time changes between the initial startup testing (late fall of 1975) and a subsequent complete series of scram tests conducted in January 1977, or by results from scram testing at other CE plants in which guide tubes were in a worn condition. However, the additional 3-inch insertion would provide an added measure of assurance that such interference would not occur.

Another factor considered was that the additional 3-inch insertion would halt the wear process at the full-out position and redistribute the accumulative wear over a larger area. This would result in less severe local wear of the guide tubes for those plants that were in various stages of their current operating cycles.

A third consideration given to the deeper insertion of the control rods into the guide tubes was the added resistance to lateral displacements. By conferring additional stiffness to the guide tubes by virtue of the inserted control rods, the effect of the guide tube wear (loss of cross-sectional area) would be reduced under lateral (seismic) loading conditions.

The above considerations given to the advantages of the additional 3-inch insertion toward minimizing the severity of the guide tube wear are somewhat obvious. However, to further justify continued operation with the 3-inch insertion limits, the licensees performed fuel assembly stress analyses using loadings for normal and accident conditions and the limiting amount of guide tube wear observed in CE cores, with and without sleeves. The sleeving modification is discussed in Section 1.7.2. The resulting stresses were below allowable values. The various mechanical loads applied to the fuel assemblies included fuel assembly holddown loads, fuel assembly handling loads, CEA scram deceleration loads, and seismic loads. The capability of the worn guide tubes to sustain these loads was determined and demonstrated by the out-of-reactor tests (see Section 1.8). The tests showed that the lateral deflection of the guide tubes and the associated mechanical friction during scram were insufficient to prevent CEA insertion and that a coolable geometry was maintained by limiting permanent deformation of the fuel assembly.

The licensees' analyses of the mechanical integrity of the core for a postulated LOCA showed that the fuel remained in a coolable arr... Although these analyses did not include a treatment of asymmetric blowdown loads, the addition of sleeves would not change the overall core response to these loads. However, a review of the response of the core to this loading condition was deferred pending resolution of the generic Category A-2 Task Action Plan, "Asymmetric Blowdown Loads on PWR Reactor Vessels." The targeted completion date of this program that includes a revised LOCA analysis for the largest credible break size is early 1980. The continued operation in the short-term and interim period of time was considered to t justified by the staff (Refs. 13-17) in view of the low probability of a lar e pipe break.

The NRC staff used a sho -term, interim criterion to determine if an acceptable level of safety exists f the reactor vessel supports of operating PWRs under conditions of a postulate. pipe break. Using this criterion, the staff concluded that the probability of a pipe break severe enough to result in substantial transient loads on the vessel support system or other structures is acceptably small. The interim criterion was based on a simplified probabilistic model that incorporated elastic fracture mechanics techniques to estimate the probability of a pipe break. Critical flaw size and subcritical flaw growth rates were determined assuming the presence of a surface flaw located in a circumferential weld of a thick-walled pipe. Determination of the critical flaw size was based on an estimated fracture toughness value at a minimum temperature of 200 degrees Fahrenheit and a uniform tensile stress equal to the consideration of various operating conditions producing elastically calculated stresses ranging in value from one to three times the material minimum yield strength.

Using the calculated critical flaw size, the subcritical growth rate, and an estimated probability distribution of an undetected flaw in pipe welds, the upper bound probability of pipe break was estimated by the staff to be acceptably low (Refs. 13-17), thus justifying short-term operation of nuclear power plants.

In addition, other conservative factors exist that not only tend to mitigate the resulting loads of the postulated accident but also further reduce the low probability of the occurrence of this event. The factors are that (1) the break of primary concern must be very large, (2) it must occur at a specific location, (3) the break must occur essentially instantaneously, and (4) these welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI. Therefore, the staff concluded that operations could continue during the interim period while the matter was being resolved.

A seismic analysis was completed for the effects of a postulated safe shutdown earthquake using the limiting reactor vessel flange acceleration time history. The licensees determined that the response to this time history enveloped the response at all CE facilities. The staff concluded that this input conservatively defined the seismic excitation of the cores. The seismic analysis accounted for the interaction effects of adjacent fuel assemblies and the core shroud through the use of appropriate gap and impact elements. Therefore, the staff determined that the licensee's seismic analysis methods were acceptable.

Based on the preceding considerations, the NRC staff approved the requested amendment changes authorizing continued operations for St. Lucie Unit 1, Calvert Cliffs Unit 2, Maine Yankee, and Fort Calhoun with the control rods inserted 3 inches further into the core.

In the case of Fort Calhoun, the 3-inch insertion requirement was subsequently rescinded for the reasons described in Section 1.5. For the Millstone Unit 2 and Calvert Cliffs Unit 1 plants, the return to operation was authorized based on the use of sleeves in the guide tubes for those assemblies placed under CEAs (see Section 1.1).

1.7.2 Corrective Action With Sleeved Guide Tubes

As a corrective action to provide protection against wear of the Zircaloy guide tubes by the vibrating CEA control rods, CE and the licensees proposed the use of sleeve inserts into the guide tubes. The sleeves (Figure 7) are made of slightly cold-worked 304 stainless steel. They are chromium-plated on the inside diameter and on the upper part of the outside diameter to improve wear resistance.

The upper ends of the sleeves are conically shaped to fit the contour of the upper end fitting posts. Because the conical section is not connected to the post, free movement under heatup, cooldown, and differential irradiation growth exists between the guide tube and sleeve.

The sleeves extend from the top of the guide tube to several inches below the observed wear area. The sleeves are securely fastened in place by mechanically "bulging" both the sleeve and the guide tube at the lower end of the sleeve (see Figure 8). The "bulge" results in diametral expansion of the guide tubes of a few hundredths of an inch on new (unirradiated) guide tubes, and slightly less on used (both worn and unworn) irradiated tubes. In addition to the "bulge," several inches of the lower portion of the sleeves are expanded diametrically toward the guide tubes so that the annular gap between the guide tube and the sleeve is reduced to zero at operating temperatures.

A series of slots and holes is provided in the sleeves to permit water flow in the annulus between sleeve and guide tube, thereby minimizing the possibility of "steaming" caused by poor heat transfer between sleeve and guide tube.

To justify the use of the sleeves as an acceptable corrective measure, the licensees provided stress analyses of normal operating and accident loading conditions for the limiting conditions of wear in both sleeved and unsleeved assemblies. The analyses included fuel assembly holddown loads, fuel assembly handling loads, CEA scram deceleration loads, seismic loads, and loss-of-coolant accident (LOCA) loads.

The licensees' LOCA analyses concluded that the fuel remains in a coolable array. However, a review of the response of the core under this loading condition has been deferred by the NRC pending resolution of the generic NRC Category A-2 Task Action Plan, "Asymmetric Blowdown Loads on PWR Reactor Vessels."

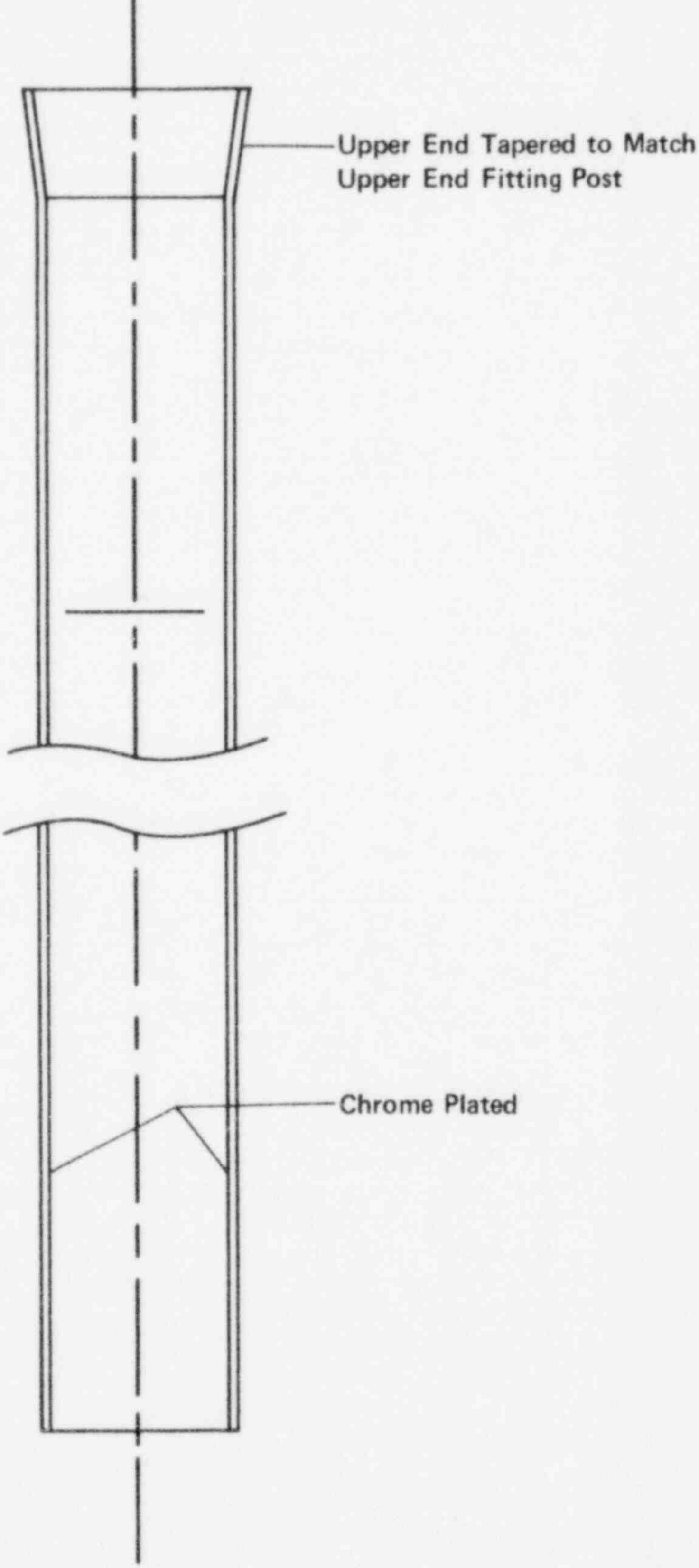

Figure 7. Combustion Engineering guide tube sleeve.

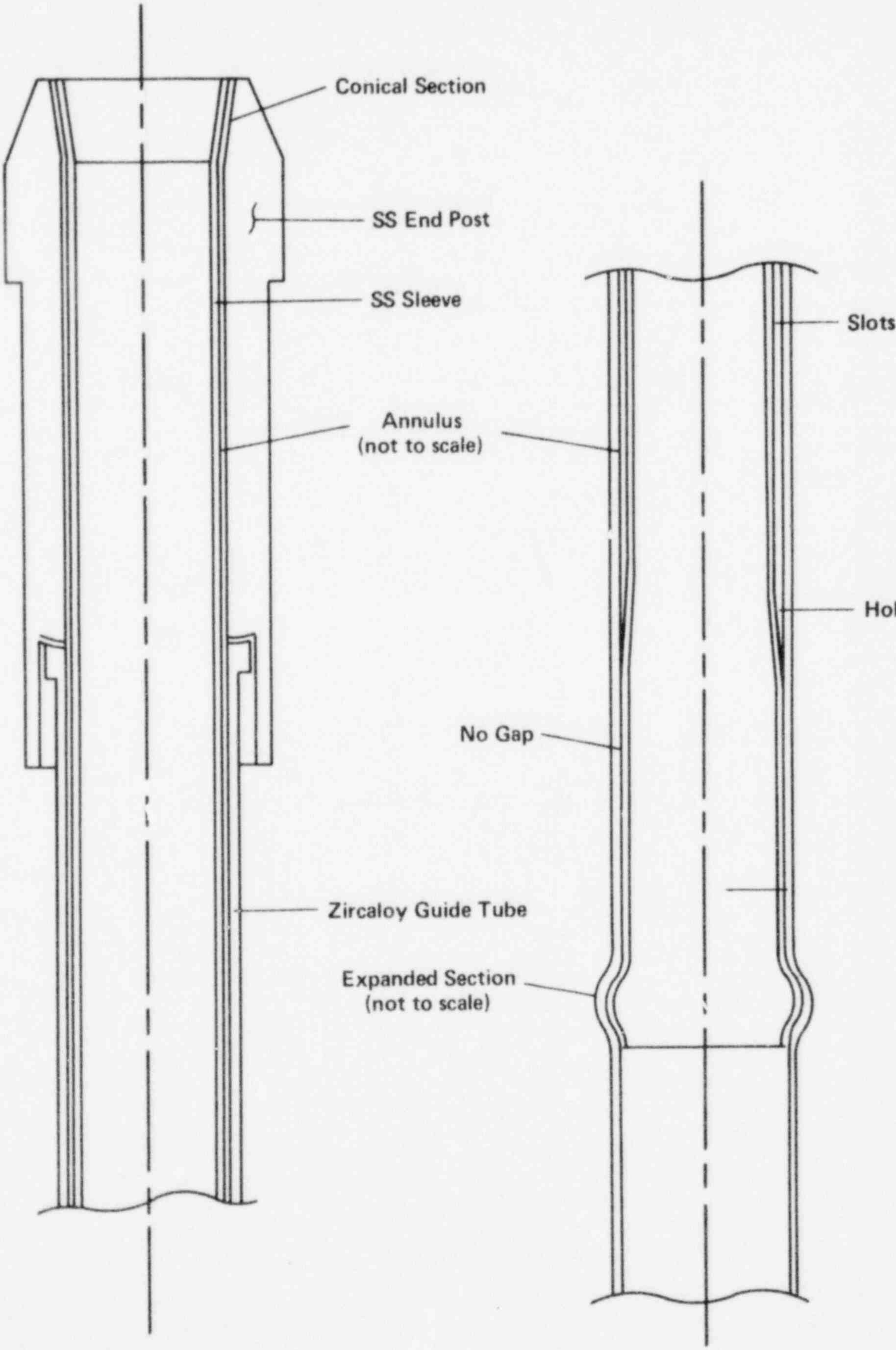

Figure 8. Installed Combustion Engineering guide tube sleeve.

The licensees' analyses showed that the state of stress during expected and postulated loading conditions in all guide tubes, whether sleeved or unsleeved, remained below the unirradiated yield strength of the Zircaloy-4 material. In addition, the stainless steel sleeve stress intensity was calculated for the corresponding portion of the load that it carries, and the stress was shown to be less than the material yield strength.

Interaction between the sleeve and guide tube was shown to create substantial secondary stresses in addition to the previously discussed primary stresses. Differential thermal expansion, differential irradiation-induced growth, and creep were considered. The staff concluded (Refs. 13-17) that the licensees' calculated stress intensities are low enough to assure an adequate margin of safety.

As additional demonstration to support the use of the sleeving modification, CE performed a number of tests on sleeved guide tubes to verify the mechanical strength of the assembly, the effect of sleeves on scram time, wear performance, the corrosion resistance in the annulus between the sleeves and tube. The tests are described in Section 1.8.

Based on the results of the licensees' analyses and CE's tests, the staff concluded that the licensees have demonstrated scrammability and coolability as required by the General Design Criteria. Therefore, the staff agrees that the sleeves provide an acceptable interim correction modification to prevent guide tube wear.

1.7.3 Alternate Corrective Action With Flow-Modifying Demonstration Assemblies

In addition to the installation of sleeves in the CEA guide tubes as protection against wear, CE has considered and tested several design modifications aimed at eliminating guide tube wear by preventing CEA vibration. Two different flow-modifying demonstration assembly designs have been adopted for further evaluation based on favorable results of CE's out-of-reactor tests described in Section 1.8.

Sixteen of the Type 1 demonstration assemblies have been approved for use in the Calvert Cliffs Unit 2 core (Ref. 17). The use of two of these assemblies has also been approved for use in the Maine Yankee core (Ref. 16). Only two of the Type 2 demonstration assemblies have been approved for use in the Maine Yankee core (Ref. 16). The limited number of demonstration assemblies will provide in-reactor experience to support either modification for use in subsequent 14x14 fuel designs.

Several factors were considered in determining the core locations of the demonstration fuel assemblies. Locations of relatively high CEA-induced wear were included so that the new designs would be put to a meaningful test but not by using only the highest wear locations. Based on previous ECT measurements, the locations selected included a broad spectrum of wear, but not the very worst. Fuel management considerations at Calvert Cliffs Unit 2 dictated putting four demonstration assemblies in non-CEA locations.

Dual CEA locations were also included. Because the demonstration assemblies are hydraulically dissimilar to sleeved assemblies, they were not mixed under dual CEAs. Because there were no experimental data showing how a dual CEA would perform under conditions of differential flow, the modified assemblies were installed as pairs under dual CEAs.

1.8 Out-of-Reactor Test Programs Performed by CE

1.8.1 Hot Cell Examination

To assist in the characterization of the guide tube wear, the upper portion of the five guide tubes of a worn Millstone Unit 2 assembly were cut from the assembly and shipped on December 28, 1977, to the Battelle-Columbus hot cell facility with the upper grid and upper end fitting (UEF) attached. The guide tubes were subjected to visual inspections and destructive examinations, including extensive metallography.

1.8.1.1 Visual Inspection

Visual examinations of the UEF guide tube assembly revealed wear openings in some of the guide tubes. The guide tube wear patterns were characterized by visual observations and azimuthal wall thickness measurements.

1.8.1.2 Metallography

Transverse sections were cut from the wear opening regions of the guide tubes to characterize the hydrogen distribution and wall thickness variations. The transverse sections of these metallographic specimens were used to confirm the metallographically based hydrogen estimates.

In general, the surfaces of interest were exposed by grinding, polishing, and etching to reveal the hydride precipitates. Cross sections of the specimens containing the maximum wear opening and wear regions at different elevations were examined. The respective azimuthal wall thickness data and hydrogen content were estimated by comparison with metallographic standards. The ring of guide tube material adjacent to a metallographic specimen was also sectioned and independently analyzed to confirm the hydrogen content estimates based on a comparison with the metallographic standards. A comparison of the analytical data with the metallographic hydrogen estimates for the adjacent material indicated good agreement, which supported continued use of the estimation method.

The hydrogen estimates and wall loss data at these elevations were summarized to provide the distribution of hydrogen picked up in the wear process. The data showed that the hydrogen pickup by both the worn and unworn sides of the guide tube is peaked at and just above the maximum wear opening. The decrease in hydrogen content with distance away from the wear opening was more pronounced for the worn side than for the unworn side.

1.8.1.3 Scanning Electron Microscopy

A specimen from one of the guide tubes at the maximum wear elevation was sectioned and subjected to a scanning electron microscopy (SEM) examination to characterize the worn and unworn surfaces. Circumferential striations were observed in the worn region similar to the SEM examination of specimens from the out-of-reactor test.

1.8.1.4 NRC Conclusions From Hot Cell Examinations

The hot cell examination results indicated that the guide tube wear phenomenon observed in-reactor was similar to the wear observed after evaluation of out-of-reactor flow test results.

The examined Millstone Unit 2 assembly was chosen during the early stages of the guide tube site inspection. The magnitude of wear characterized was typical for two cases: (1) wear openings on one side, and (2) wear around the entire circumference without a wear opening. However, the guide tubes did not represent a worst case for either type of wear pattern.

Nevertheless, the information obtained or inferred by the metallographic examinations showed that Zircaloy-4 will pick up relatively small amounts of hydrogen by virtue of the oxidation reaction in PWR water, but that exposure for no more than three fuel cycles should result in acceptable concentrations. The presence of radiation should not be a factor in the amount of hydrogen absorbed provided that the alloy maintains a protective oxide film. When fretting occurs, however, the oxide will fragment and act as an abrasive. This can keep some fresh metal surface exposed to the coolant and cause the hydrogen pickup to increase. Once absorbed, the hydrogen will not escape because it forms a stable solution of zirconium hydride precipitates.

The tests reported by CE involving wear of Zircaloy-4 tubes by control rods in a loop (not in-reactor) resulted in acceptable hydride concentrations after about one-half the wall thickness was abraded from the 40-mil tube. Metallographic examination of one cross section of one tube from the examined Millstone assembly showed large hydride concentrations at the feather edge of the hole worn through the tube by the control rod. Because neither the location nor the control rod guide tube can be said to be the worst examples of wear or hydrogen pickup, it would not be surprising to find hydride concentrations that are higher than usually observed in badly worn guide tubes.

Based on these facts, the staff concluded that CE PWR control rod guide tubes can suffer from three localized detrimental effects: (1) a notch in the form of a hole through the tube wall (stress concentration of at least three), (2) radiation effects in mechanical properties (increased strength and decreased ductility), and (3) relatively high hydride concentration. The following generalities give perspective to the effect of hydrides on irradiated Zircaloy-4 notch strength. For hydride concentrations of less than 100 ppm, no significant effect on the static mechanical properties would be expected over three fuel cycles. For hydride concentrations that are 100 to 200 ppm, the static mechanical properties at operating temperature would be essentially unchanged for three fuel cycles. However, the material will be notch-sensitive at room temperature (the notched strength will be less than the ultimate tensile strength). For hydride concentrations that are 1000 ppm, the mechanical properties will be degraded at all temperatures (up to about 1000 degrees Fahrenheit) relative to notch strength. Moreover, even at reactor operating temperatures, local (concentrated) stresses of 50 ksi could induce fracture at relatively sharp notches ($K_t > 3$) and the crack might, under constant load conditions, propagate through Zircaloy with hydride concentrations of about 150 ppm.

Because relatively high hydride concentrations have been inferred from metallographic examination of a tube from the Millstone assembly, the staff concluded that the ability of the guide tube material to perform its function is questionable when subject to formation of a wear hole. Because severe wear is known to have occurred in control rod guide tubes at some CE PWRs, the problem was assumed to be widespread. As previously mentioned, for those fuel bundles that were deemed questionable due to worn guide tubes, the structural integrity and thus the acceptability for use was restored by sleeving the subject guide tubes. At the present time (see Sections 2.1 and 3.1), the amount of guide tube wear reported by Westinghouse and Babcock & Wilcox (B&W) is much less severe, thus suggesting that the problem in other PWRs is of less immediate concern.

1.8.2 CEA Insertion Test

The objective of the CE CEA insertion test was to establish the magnitude of the mechnical friction resistance to CEA insertion expected to occur in a fuel assembly that had experienced severe guide tube wear and that had then been subjected to seismic excitation representative of the safe shutdown earthquake (SSE).

The CEA insertion test was performed on a standard 14x14 grid cage without guide tubes to simulate the mechanical effects of severe wear. (The complete fuel assembly less fuel rods is referred to as a grid cage.) The simulated wear areas were made in the form of rectangular cutouts in the sides of the guide tubes rather than attempting to duplicate actual circumferential and axial variations of wall thickness. The final selection of cutout angles was made to ensure that the stresses produced in the test section would bear an equal or more severe relationship to the test condition yield stress, much as the stress produced in the actual cross section would have to the yield stress in the irradiated worn tubes. In addition, with the strains produced in the grid cage components being dependent only upon the deflected shape of the assembly, the resultant stresses would be considerably higher in the room temperature test because the room temperature elastic modulus for Zircaloy (15.25×10^6 psi) is approximately 42 percent higher than the operating temperature value of 10.75×10^6 psi.

To determine the effect of the simulated guide tube wear on CEA insertion force, CE determined analytically the instant that the moment at the wear elevation reached a maximum during the SSE excitation. CE then statically deflected the grid cage to the shape corresponding to the time of peak moment because it would then represent the limiting effect of the wear.

These CEA insertion tests demonstrated acceptable CEA insertion performance even in the presence of severe wear in the guide tube. The drag force determined after the grid cage was allowed to spring back indicated negligible effect, being no greater than the forces commonly encountered for CEA motion in new assemblies. It is significant to note that the deflected shape used in this test exists only momentarily during postulated SSE excitation and would not persist following such excitation.

There was no evidence of permanent deformation of the guide tubes at and near the elevations of the cutouts. However, the staff noted that the guide tube material in the region of the cutouts did not simulate the high hydride content evidenced by the metallographic examinations (see Section 1.8.1.4).

1.8.3 Single-Element Flow Visualization Test

A flow visualization test program with a single CEA finger was conducted in a vertical, cold-water mockup of a CEA guide tube and upper shroud. The mockup allowed direct observation of the behavior of the CEA finger. Probes to measure lateral displacement of the CEA tip at various flows were installed, as were accelerometers to measure the frequency and amplitude of the CEA finger motion at midspan in the shroud region. Tests were conducted at mass flows from zero to more than double the reference flow in the guide tubes with varying depths of rod insertion into the guide tube, various CEA clearances and diameters, and varying configurations at the CEA tips and at the exit from the guide tube post. The results of these tests showed that the single CEA finger was sensitive to flow-induced vibration.

1.8.4 Flow Test of Unsleeved 14x14 Fuel Assembly

CE flow tested a 14x14 fuel assembly (Figure 9) in their TF-2 loop over a range of typical reactor temperature, pressure, and coolant flow rate conditions. The primary objective was to observe the control element assembly (CEA) scram capability. Repeated movement of the CEA included a significant number of full-height and partial-height operations.

After the test, the inside diameter of the guide tubes was examined using a borescope. CEA guide tube contact patterns were observed and evaluated to show no significant wear. Because of the wear observations at Millstone Unit 2, the top section of a CEA guide tube was visually examined and dimensionally characterized.

Sample sections were subjected to microscopic, metallographic, and chemical analysis. In summary, the observations indicated that metal loss was occurring due to mechanical removal of material by the control rod as it moved in a horizontal plane and abraded the surface of the guide tube. The maximum hydrogen pickup occurred in the worn region of the tubing and was substantially below levels where significant changes in tubing properties occurred.

1.8.5 Flow Tests of Sleeved 14x14 Fuel Assembly

A standard 14x14 bundle was equipped with sleeves inserted in the upper portion of the guide tubes. The sleeving was conducted underwater to simulate conditions under which the sleeving operation was conducted in the reactor spent fuel pool. The sleeves were then reinspected. No cracking or damage of the chrome was observed in the expanded regions.

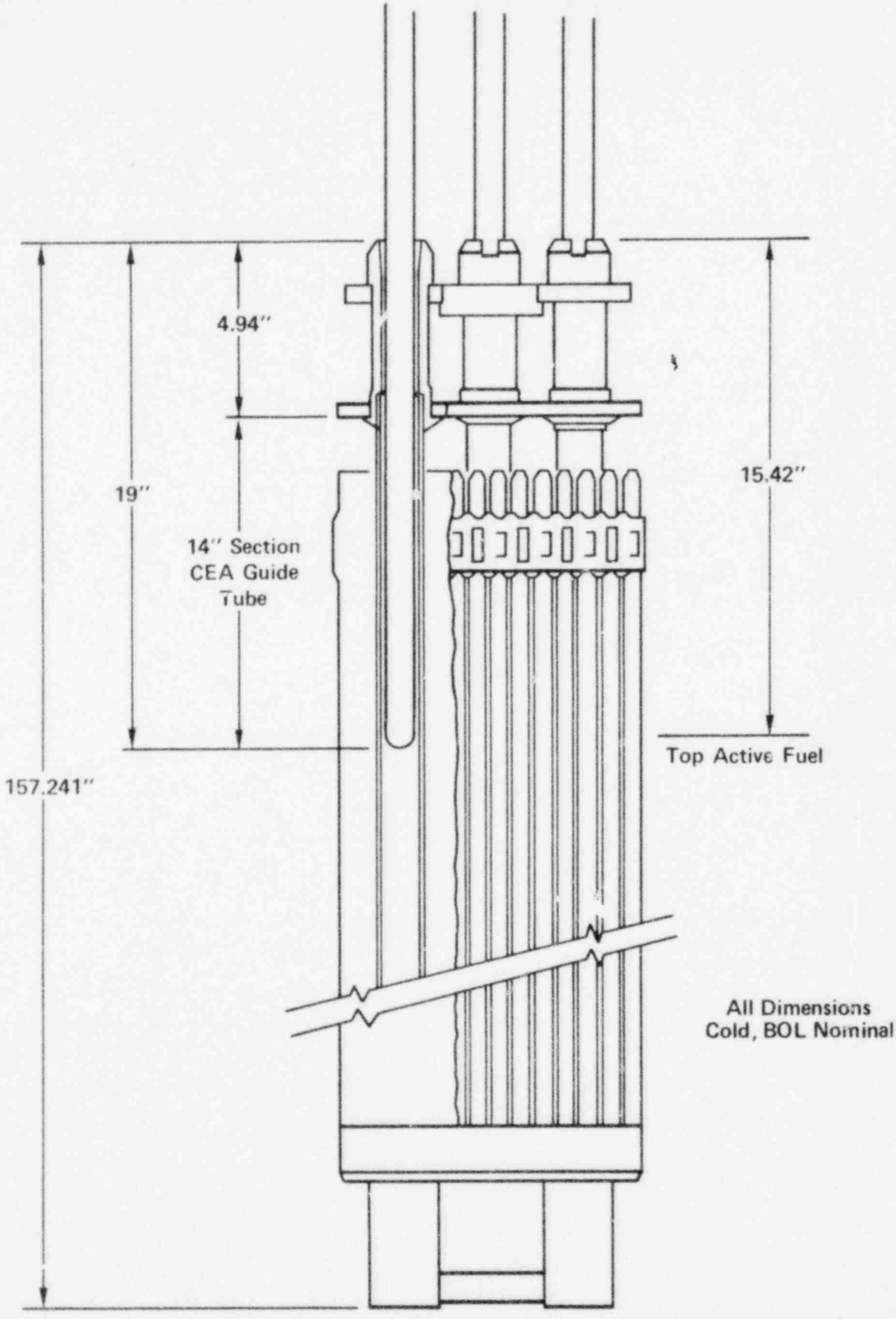

Figure 9. Combustion Engineering flow test bundle.

The bundle was then tested in the TF-2 loop. The test conditions were similar to the flow test conditions described in Section 1.8.4. The CEA was moved during the test and returned essentially to the same position relative to the top of the post. Following the loop test, the bundle was examined to evaluate the general condition of the sleeved guide tubes and to determine the extent of wear between the Inconel-625 clad control rods and both the chrome-plated and unplated stainless steel sleeves.

Examinations of the fuel assembly and the CEA fingers following the loop test included a visual examination of the sleeves and CEA fingers prior to disassembly, followed by a destructive examination of the guide tube and sleeves. The program included clamshelling of the sleeved guide tubes and a magnified visual inspection of all tube surfaces. This was followed by a metallographic examination of sections of sleeving from various locations.

Transverse sections from both the crimped and expanded regions of guide tubes were metallographically examined. The hydrogen levels were low, which was typical of the as-received conditions.

Based on the preceding results and a comparison of Zircaloy wear patterns obtained on guide tubes from the earlier 14x14 loop test and from the examined Millstone Unit 2 fuel assembly, CE concluded that results of the TF-2 loop test could be directly related to the expected in-reactor performance.

1.8.6 Flow Tests on Demonstration Assemblies

A number of out-of-reactor flow tests were performed to evaluate the adequacy of the flow modifying demonstration assemblies to reduce, or eliminate, the CEA vibration. These tests are described below.

1.8.6.1 Single-Element Flow Visualization Tests

CE's single-element test loop was used to investigate flow-induced vibration caused by the flow in the fuel assembly guide tube. Testing in this loop indicated that the use of flow-modifying designs should substantially reduce CEA vibratory motion resulting from flow in the guide tube.

1.8.6.2 TF-15 Testing

CE's TF-15 test loop allowed visual observations of flow-induced vibrations caused by flow through the fuel bundle, upper guide structure, CEA shroud and scuppers, and the fuel assembly guide tubes in various combinations. Standard instrumentation was used to monitor performance characteristics. Tests conducted in the TF-15 test loop indicated a substantial decrease in CEA vibration with the flow-modifying designs relative to results with the standard design.

1.8.6.3 TF-2 Testing

Full-scale flow tests in CE's TF-2 test loop involved a 14x14 fuel assembly containing a five-rod CEA, a control element drive mechanism, and prototypical internals components. The tests were conducted at flow conditions that bound maximum flow rates for any rodded fuel assembly. The purpose of this test was to verify the acceptability of the reduced guide tube flow rate with regard to CEA vibratory motion and to establish scram characteristics. Hot and cold short-duration tests were performed with a standard 14x14 fuel bundle geometry. Testing showed that CEA finger vibration was random at lower amplitudes. The natural frequency component observed with standard 14x14 fuel design was barely evident at cold conditions and was absent at hot conditions. Hot scram testing indicated the 90 percent insertion time increased by less than 1 second. In conclusion, the test indicated that guide tube flow resulting from the flow-modifying designs significantly decreased CEA vibration.

1.9 Continuing Programs

1.9.1 Discussion

As discussed in Section 1.1.4, the licensees' current corrective action is the sleeving procedure, which will be used until other alternate corrective actions are identified and demonstrated. Thus, the sleeving modification will be used in the primary reload fuel design during this interim period. To provide assurance that the interim corrective action performs as expected, the staff has requested that the licensees commit to continue the guide tube surveillance programs.

The continuing end-of-cycle (EOC) program consists of (1) sleeving (as a minimum) those fuel assemblies placed under CEA locations, (2) nondestructive testing of the guide tube sleeves to determine the extent of wear, and (3) ECT and pull tests to confirm that the sleeves are securely positioned in the guide tubes. A brief description of these segments of the continuing programs is discussed in the following sections.

1.9.2 Sleeving Procedure

The sleeving method is covered by a written procedure* that includes qualification of the tooling before each operation and replacement of those parts of the tooling subject to wear or deterioration to avoid any deleterious effects on the process.

Prior to implementing the sleeving procedures, a sleeve-expansion qualification program was performed as follows: (1) selection and irradiation testing of an elastomer that is highly resistant to irradiation damage to expand the sleeve, (2) determination of the proper elastomer geometry and Durometer hardness, and (3) a check of the variation in pressure required to expand sleeves in unirradiated and irradiated guide tubes using the same tool design as used in the field. All sleeve-expanding tools are designed to prevent inadvertent overexpansion of the sleeves.

After the sleeving procedure is completed, the following checks are made to ensure that the process performed correctly:

(1) A pull test is performed on each sleeve.

(2) A visual inspection is performed to ensure that the sleeve is properly seated and that no debris is left in the area.

*The detailed proprietary procedures are not provided since they are not necessary for purposes of this report.

(3) Two separate gauging operations, using a single-finger gauge and a five-finger gauge, are performed to ensure that there will be no interference with CEA operation.

Initially, CE performed visual examinations to make sure that the "bulging" process did not produce cracks in the irradiated guide tubes. Several assemblies were examined by periscope, including one of the most highly irradiated assemblies. No evidence of cracking was found.

The sleeving procedures initially performed at Calvert Cliffs Unit 1 were followed by a pull test to verify that the process was performed correctly and that the sleeves were secured in place. Subsequent to the sleeving operation at Calvert Cliffs Unit 1, which was the first CE unit to be sleeved, the pull test was performed after the crimping step but prior to the expansion step of the sleeving procedure. This procedural change corrected what was considered to be one of the factors that may have provided the additional resistance from the expanded sleeves, tending to mask the presence of inadequate crimps found at the end of the first cycle of operation with certain sleeves in the Calvert Cliffs Unit 1 plant. This type of wear is discussed in further detail in Section 1.9.5.

For all sleeves installed in 1979 and in Calvert Cliffs Unit 2, one other significant change was introduced; namely, a new crimp geometry. The "new crimp" is shorter and results in a more uniform and better definition of the crimp "bulge" that secures the sleeve within the guide tube.

CE has performed a number of tests on sleeved guide tubes to verify the mechanical strength of the assembly, effect of sleeves on scram time, wear performance, and corrosion resistance in the annulus between the sleeve and tube. The test facilities are described in Section 1.8. CE also determined that the force necessary to remove a sleeve from the guide tube was greater than two orders of magnitude larger than the drag force exerted on the sleeve by the control rods. To simulate relaxation that would occur in service, CE included a number of thermal cycles that ranged between room temperature to 625 degrees Fahrenheit in the testing program. Approximately one-half the force required prior to thermal cycling was needed to remove the sleeves after the thermal cycling.

Based on the information submitted by the licensees and CE, the staff concluded that the sleeving procedures and the sleeved guide tubes would perform their design function of reducing guide tube stresses to acceptably low values and that the mechanical design of the sleeved assembly was satisfactory for at least one fuel cycle. Any long-term effects of relaxation of the mechanical "bulge" joint, including the possibility of radiation-enhanced relaxation, would be reevaluated on selected assemblies at each refueling outage.

Therefore, the licensees have been required to prepare a CEA guide tube evaluation program plan and to submit it to the NRC at least 90 days prior to the scheduled refueling outage for each plant using the sleeving modification with the above sleeving procedures.

1.9.3 Nondestructive Tests

Because of the large number of guide tubes to be inspected, it was necessary for CE to develop a nondestructive test (NDT) technique to rapidly inspect and determine the axial location of the wear indications and general condition of the guide tubes. The NDT techniques developed by CE were based on their experience with eddy current testing of steam generator tubing. An averaging eddy current probe was built to test the full circumference of the guide tube as a function of position while the probe moved axially in the guide tube. The probe was calibrated with various machined standards. The information was used to determine the axial position of maximum wear and to screen the tubes to determine which guide tubes should be inspected more thoroughly.

To more accurately define the circumferential wear profile, an azimuthal type of eddy current probe was developed. The probe was used to search an incremental sector of the guide tube. The voltage output of the azimuthal probe was correlated with the amount of tube wall loss in the incremental sector.

To inspect a guide tube with the azimuthal probe, the probe is inserted to the axial location of interest and moved axially in and then out for a given length. With an indexing device, the coil is then rotated and the coil is inserted and withdrawn the same axial distance. The process is repeated for the full 360° circumference of the tube. From calibration curves obtained from the same standards used for the ECT coil and the strip chart recordings, the tube wall thickness profile for any 360° cross-sectional area of the tube can be determined.

A third type of eddy current probe was also developed by CE to rapidly measure the crimp bulge in the sleeved assemblies. This probe was similar to the ECT probe used in the screening process. The signal voltages were calibrated with manufactured standards to accurately assess the crimp depths. The minimum acceptable crimp depth was determined by comparing the measured ECT crimp signals with pull test results indicative of sufficient "bulge" interference to preclude axial movement under a given axial load.

In addition to developing nondestructive ECT techniques to measure guide tube wear and sleeve crimp adequacy, CE developed an ECT technique to measure CEA control rod wear indications. The device consists of a five-coil fixture. The test coils are individually balanced encircling coils. Each of the five control rods are measured independently at different times to minimize signal interference. The signal amplitudes are then compared with calibration standards to assess the extent of control rod wear. A more general description of ECT techniques is discussed in Appendix B.

1.9.4 Pull Tests

The pull tests referred to in Section 1.9.2 were part of the CE sleeving procedure until the Calvert Cliffs Unit 2 sleeving in 1979. The use of the pull test was to provide assurance that the crimping operation had provided adequate resistance to axial movement of the sleeve by virtue of the sleeving operations.

In addition, pull tests are performed after the fuel has experienced one or more cycles of exposure. Two types of pull tests are involved. One is identical to the pull test described in Section 1.9.2. Its purpose is two-fold; first, it is used to determine whether sufficient relaxation (or growth) has occurred between the sleeve and the guide tube to result in axial movement of the sleeve by the drag force of the control rods, and, second, the go/no-go movement of the sleeve under the pull test force is used to calibrate the ECT crimp signal for more rapid determination of the crimp adequacy.

The other type of pull test used to a lesser extent is a pull-to-remove test. A limited number of such tests have been used to determine the force required to completely remove the sleeve from the guide tube. Test results have shown that the pull-to-remove force is much greater than the drag force of the control rods.

Based on the evidence gathered during 1979 (that is, crimp sizes and pull test results), the sleeve pull test as part of the sleeving procedure has been deleted. In its place an eddy current test is required to measure the crimp size and therefore immediately determine its adequacy to resist axial motion.

1.9.5 Results of First Cycle Operating Experience With Corrective Modification

During the first refueling outages following the installation of the sleeves at Millstone Unit 2, St. Lucie Unit 1, and Calvert Cliffs Unit 1, CE conducted the guide tube sleeve inspections. TV scans of the top of the core were performed at all three plants. The TV scans showed no problems with proper seating of the sleeves in the guide tubes.

Eddy current wear tests were performed on a number of sleeved guide tubes located under CEAs in high wear positions. The ECT measurements from all three plants indicated that no detectable sleeve wear had occurred during the reactors' operation. Because of the lack of wear indications in the screening ECT examinations, a limited number of guide tubes were examined by the more sophisticated azimuthal ECT technique at the Millstone Unit 2 facility. Since the Millstone Unit 2 plant was the first of the three plants to refuel, it was considered the lead plant in regard to the extent of the inspections to be performed. Table 6 shows the type of examinations performed on guide tube sleeves and CEAs at each of these facilities. All the CEAs examined were used in sleeved fuel assemblies for the preceding cycle. Results of the CEA examinations confirmed that the CEA control rod tips experienced no appreciable wear as a result of contact with the harder surface afforded by the stainless steel sleeve inserts.

Table 6. Wear Examinations After One Cycle With Sleeved Guide Tubes

Plant	Type of Guide Tube Examinations		Type of CEA Examinations	
	Average Wear Probe	Azimuthal Wear Probe	ECT Wear	Periscope Visual
Millstone Unit 2	Yes	Yes	Yes	Yes
Calvert Cliffs Unit 1	Yes	No	Yes	Yes
St. Lucie Unit 1	Yes	No	Yes	No

In conclusion, the sleeving modification as a corrective action has eliminated wear of the Zircaloy guide tube and exhibited acceptable wear characteristics for both the sleeve insert and the CEA control rod tips for those plants examined.

The crimp at the base of the guide tube sleeve is designed to prevent axial movement of the sleeve in the cold condition. Differential thermal expansion between the stainless steel sleeve and the Zircaloy guide tube provides additional resistance to axial movement at operating temperatures. Conversely, the maximum drag force exerted on the sleeves by the control rods occurs at reactor startup conditions (lower temperature). Thus, the maximum axial force tending to lift the sleeve will occur when the benefit due to the differential thermal expansion is a minimum.

To assure the adequacy of the crimp in securing the sleeve under cold conditions, additional inspection tests are performed on the sleeved fuel assemblies. From experiments performed, CE has determined the drag force exerted by the control rods on sleeved fuel assemblies. The pull test performed on the sleeved guide tubes to assure adequate resistance to axial movement is approximately 10 times the magnitude of the measured drag force. To provide a more rapid means of determining the crimp adequacy, CE has performed a comparative evaluation of the ECT crimp probe signal versus the pull force exerted on the sleeves. The ECT probe calibration for determining crimp adequacy is discussed in Section 1.9.3.

Table 7 shows the ECT examinations and confirmatory pull tests performed on the guide tube sleeves at each of the three CE reactor facilities.

Table 7. Crimp Verification Tests

Plant	Sleeved Guide Tubes Examined ECT Crimp Probe	Sleeve Pull Tests
Millstone Unit 2	Yes	Yes
Calvert Cliffs Unit 1	Yes	Yes
St. Lucie Unit 1	Yes	No

During the refueling outage at Calvert Cliffs Unit 1, which followed the Millstone Unit 2 and St. Lucie Unit 1 outages, a significant number of guide tube sleeves did not exhibit the expected resistance to axial movement. However, as stated previously, the TV scan of the top of the core detected no improper positioning of the sleeves, and no sleeve wear was indicated.

As a result of the anomalous crimp indications at Calvert Cliffs Unit 1, the licensees' ECT crimp and pull test program was enlarged to assess the crimp acceptability in a variety of different category fuel assemblies. It was determined that there were basically four categories of sleeved fuel assemblies in the CE reactors using the sleeving modification. These four categories are given in Table 8. Table 8 also identifies the fuel batch loadings that were sleeved and eddy current tested for crimp signals in each category for each plant.

Table 8. Summary of ECT Crimp Test Per Fuel Category Per Plant

Category	Description	Plant	Fuel Batch	Sleeves ECT Tested
a	Irradiated unworn sleeved 1978 (old style crimp)	Millstone Unit 2	B, C	Yes
		St. Lucie Unit 1	B, C	Yes
		Calvert Cliffs Unit 1	A, C, D	Yes
b	Nonirradiated unworn sleeved 1978 (old style crimp)	Millstone Unit 2	D	Yes
		St. Lucie Unit 1	D	Yes
		Calvert Cliffs Unit 1	E	Yes
c	Irradiated unworn sleeved 1979 (new style crimp)	Millstone Unit 2	D	Yes
		St. Lucie Unit 1	B, D	Yes
		Calvert Cliffs Unit 1	D	Yes
d	Nonirradiated unworn sleeved 1979 (new style crimp)	Calvert Cliffs Unit 2 First plant		No data

After evaluating the expanded Calvert Cliffs Unit 1 data and comparisons with data from Millstone Unit 2 and St. Lucie Unit 1, the licensees and CE concluded that the observations of unacceptable crimps were unique to the Calvert Cliffs Unit 1 plant. In particular, the unacceptable crimps at Calvert Cliffs Unit 1 were isolated to the Category "a" type of sleeved assemblies. Examination of sleeving procedures used for the three plants also revealed that the pull verification tests at Calvert Cliffs Unit 1 were performed after both the crimp and expansion steps of the sleeving operations (see Section 1.9.2), whereas the pull verification tests at the Millstone Unit 2 and St. Lucie Unit 1 plants followed the crimp step and preceded the expansion step in the sleeving operation. Another factor considered by the staff was that the increased yield strength of the irradiated guide tube material resulted in greater resistance to deformation (crimping and expansion) for the Calvert Cliffs Unit 1 plant because of a higher average core burnup for this plant.

To correct the presence of unacceptable crimps in the Category "a" type of fuel assemblies returned to CEA locations for the next cycle in Calvert Cliffs Unit 1, the new style crimp (see Section 1.9.2) was superimposed (recrimped) over the existing old style crimp. After the recrimping operation, eddy current tests were performed on each of these assemblies to assure acceptable crimp sizes.

Reexamination of the fuel loadings for Millstone Unit 2 and St. Lucie Unit 1 showed that only eight of the Category "a" assemblies were placed under CEA locations for the second cycle of operation with sleeved guide tubes. All eight of these Category "a" fuel assemblies were in the Millstone Unit 2 plant, with no Category "a" fuel assemblies located under CEAs in the St. Lucie Unit 1 plant.

Therefore, as an additional measure to preclude the possibility of sleeve movement during subsequent shutdowns, CE considered it prudent to recommend a restriction on the movement of the control rods at system temperatures below 400 degrees Fahrenheit. The basis for this recommendation is related to the differential thermal expansion between the sleeves and the guide tubes as previously discussed.

1.10 NRC Conclusions on First Cycle Operations With Corrective Actions

The staff's review of the operating experience after one cycle of operation with the sleeved guide tubes and the staff's approval (Refs. 18, 19) for second-cycle operation with sleeved guide tubes at Millstone Unit 2 and St. Lucie Unit 1 were based on the favorable inspection results at these facilities. These results confirmed earlier tests and calculations to show that sleeving is an acceptable repair to alleviate CEA guide tube wear. The apparent anomalies at Calvert Cliffs Unit 1 that revealed a large number of sleeved fuel assemblies outside the ECT and pull tests acceptance criteria and the explanation of these results are discussed above.

Therefore, based on the remedial action (recrimp of the Category "a" fuel assemblies) taken by the Calvert Cliffs Unit 1 licensee and the above discussion, the staff approved (Ref. 20) Calvert Cliffs Unit 1 cycle 4 (second cycle with sleeved guide tubes) operations. Likewise, our approvals (Refs. 18, 19) for second-cycle operation with sleeved guide tubes in the Millstone Unit 2 and St. Lucie Unit 1 plants remained unchanged.

As shown in Table 8, the inspection results from Calvert Cliffs Unit 2 were not available at the time this report was being prepared. However, the staff has reviewed and approved the end-of-cycle inspection program for Calvert Cliffs Unit 2. In addition, staff members visited the Calvert Cliffs Unit 2 facility to observe part of the end-of-cycle inspections in progress.

1.11 References

1. Letter from D. K. Davis, NRC, to A. E. Lundvall, BG&E, Subject: Amendment 28 to Technical Specification Approving Three-Inch Control Rod Insertion, Docket No. 50-317, January 6, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

2. Letter from D. K. Davis, NRC, to A. E. Lundvall, BG&E, Subject: Amendment 13 to Technical Specification Approving Three-Inch Control Rod Insertion, Docket No. 50-318, January 6, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

3. Letter from D. K. Davis, NRC, to R. E. Uhrig, FP&L, Subject: Amendment 20 to Technical Specification Approving Three-Inch Control Rod Insertion, Docket No. 50-335, January 6, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

4. Letter from D. K. Davis, NRC, to R. H. Groce, MYAPC, Subject: Amendment 33 to Technical Specification Approving Three-Inch Control Rod Insertion, Docket No. 50-309, January 6, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

5. Letter from D. K. Davis, NRC, to T. E. Short, OPPD, Subject: Amendment 34 to Technical Specification Approving Three-Inch Control Rod Insertion, Docket No. 50-285, January 6, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

6. Letter from V. Stello, NRC, to A. E. Lundvall, BG&E, Subject: Justification For Continued Operation, Docket No. 50-317, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

7. Letter from V. Stello, NRC, to A. E. Lundvall, BG&E, Subject: Justification For Continued Operation, Docket No. 50-318, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

8. Letter from V. Stello, NRC, to R. E. Uhrig, FP&L, Subject: Justification For Continued Operation, Docket No. 50-325, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

9. Letter from V. Stello, NRC, to R. H. Groce, MYAC, Subject: Justification For Continued Operation, Docket No. 50-309, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

10. Letter from V. Stello, NRC, to T. E. Short, OPPD, Subject: Justification For Continued Operation, Docket No. 50-285, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

11. Letter from T. E. Short, OPPD, to G. Lear, NRC, Subject: Guide Tube Wear, Docket No. 50-285, February 10, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

12. Letter from R. Reid, NRC, to T. E. Short, OPPD, Subject: NRC Rescinds Requirement For Three-Inch Insertion of Control Rods, Docket No. 50-285, August 7, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

13. Letter from R. Reid, NRC, to A. E. Lundvall, BG&E, Subject: Amendment 32 to Technical Specifications, Docket No. 50-317, March 31, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

14. Letter from R. Reid, NRC, to W. G. Counsil, NNECO, Subject: Amendment 38 to Technical Specifications, Docket No. 50-336, April 19, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

15. Letter from R. Reid, NRC, to R. E. Uhrig, FP&L, Subject: Amendment 27 to Technical Specifications, Docket No. 50-335, May 26, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

16. Letter from R. Reid, NRC, to R. H. Groce, MYAC, Subject: Amendment 40 to Technical Specifications, Docket No. 50-309, August 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

17. Letter from R. Reid, NRC, to A. E. Lundvall, BG&E, Subject: Amendment 18 to Technical Specifications, Docket No. 50-318, October 21, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

18. Letter from R. Reid, NRC, to W. G. Counsil, NNECO, Subject: Amendment 52 to Technical Specifications, Docket No. 50-336, May 12, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

19. Letter from R. Reid, NRC, to R. E. Uhrig, FP&L, Subject: Amendment 32 to Technical Specifications, Docket No. 50-335, May 27, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

20. Letter from R. Reid, NRC, to A. E. Lundvall, BG&E, Subject: Amendment 39 to Technical Specifications, Docket No. 50-317, June 24, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

2. WESTINGHOUSE NSSS FACILITIES

2.1 Introduction to Guide Tube Wear in Westinghouse NSSS Facilities

As a result of the control rod guide tube (CRGT) wear observed at the Combustion Engineering (CE) NSSS facility (Millstone Unit 2) in December 1977 (see Section 1.1), the staff contacted Westinghouse on December 28, 1977, to discuss any experience Westinghouse may have had with wear of control rod guide tubes. Westinghouse indicated that wear had been found in guide tubes of low parasitic fuel assemblies at operating nuclear power plants. In general, the wear in most guide tubes was found at a location corresponding to the fully withdrawn or "parked" position of the control rods. The significant wear was local wall thinning confined to a 90° sector of the guide tube. The maximum wear measured had occurred after two cycles of operation.

Metallographic specimens from two assemblies had been examined in a hot cell. Additional hydriding was found in the worn areas. Westinghouse then estimated the hydride concentrations found in the wear locations. Westinghouse reassessed the mechanical integrity of fuel assemblies with worn guide tubes, assuming the worst wear conditions. They concluded that the assemblies were structurally able to fulfill their design functions for both normal and faulted conditions even after three full cycles of wear.

The staff's preliminary review of this information (Ref. 1) indicated that the Westinghouse fuel design may have less susceptibility to CRGT wear than the CE fuel design. To further evaluate the Westinghouse findings, the staff requested documentation of the Westinghouse test results and analysis (Ref. 2). This information was provided in References 3 through 5. Based on the review of this information, the staff requested a meeting (Ref. 6) to provide Westinghouse with the opportunity to clarify specific items needed for completion of the staff review. This request was simultaneously transmitted to five utilities with near-term operating licenses. The Westinghouse/NRC meeting was held at NRC headquarters, Bethesda, Maryland, on October 12, 1979.

Based on the information provided by Westinghouse, the staff agreed that no corrective action was warranted at this time. However, the staff has requested additional confirmatory information (see Sections 2.6, 2.7, and 2.8) on the CRGT wear performance for the Westinghouse 15x15 and 17x17 fuel designs.

2.2 Design and Functions of Guide Tubes and Related Components

2.2.1 Westinghouse Fuel Assemblies

The Westinghouse fuel assembly designs (Figure 10) consist of 14x14, 15x15, and 17x17 arrays. The Westinghouse HIPAR fuel assembly design has stainless steel control rod guide tubes with an inherently harder wear surface that is not considered to be susceptible to significant CRGT wear. Therefore, only the Westinghouse LOPAR fuel design with the Zircaloy CRGTs is discussed.

The structural frame of the Westinghouse fuel assemblies consists of guide tubes, spacer grids, and end fittings. The number of guide tubes within the fuel designs is 16, 20, and 24 for the 14x14, 15x15, and 17x17 arrays, respectively.

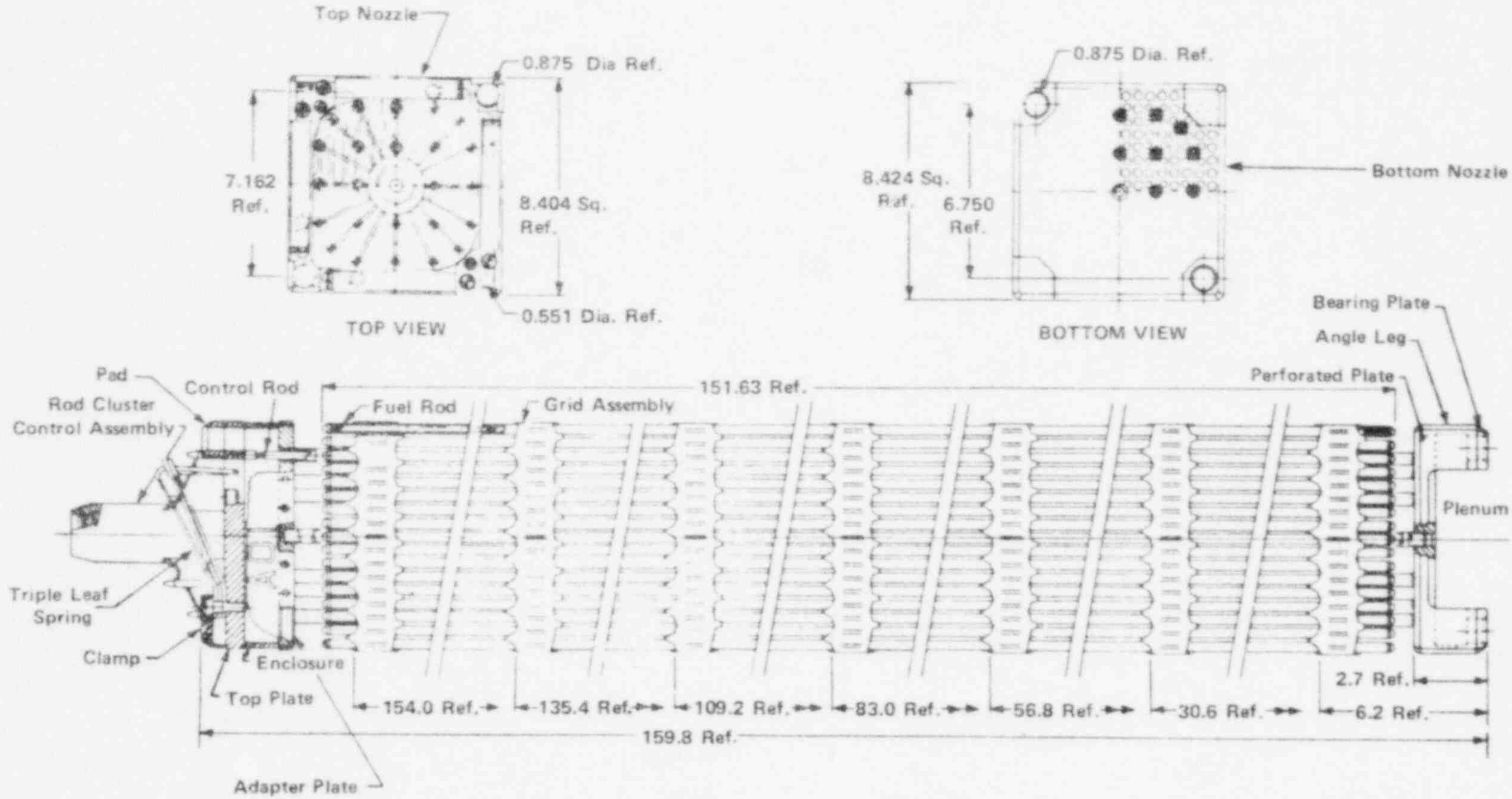

Figure 10. Westinghouse fuel assembly.

Above the dashpot region, the Zircaloy guide tube inside diameters range from 0.482 inch to 0.539 inch with wall thicknesses ranging from 0.016 inch to 0.017 inch depending on the array design. The CRGT portion, located between the top grid and the top nozzle (upper end fitting) includes an outer stainless steel sleeve. This sleeve is above the position of the control rod tips when the RCCA is in the fully withdrawn position. Therefore, it offers no wear protection to the Zircaloy CRGT from the control rod tips. The design functions of the control rod guide tubes in the Westinghouse designs are the same as those described in Section 1.2.1 for the CE design.

2.2.2 Westinghouse Rod Cluster Control Assemblies

The rod cluster control assemblies (RCCAs) consist of 16, 20, and 24 stainless steel tubes with outside diameters ranging from 0.381 inch to 0.431 inch (Figure 11) depending on the fuel assembly array discussed in Section 2.2.1. The RCCA includes a hub and spider arrangement with neutron absorber material inside the stainless steel tubes that insert into the Zircaloy control rod guide tubes (see Figure 12). The RCCAs all operate on separate drive lines with no dual control elements as are in the CE facilities.

The design functions of Westinghouse's RCCAs are essentially the same as CE's CEAs described in Section 1.2.2. However, two notable design differences between the Westinghouse RCCAs and CE CEAs are of interest: (1) the greater flexibility (less stiffness) afforded by the Westinghouse smaller diameter control rods, and (2) the use of stainless steel cladding material for the Westinghouse control rods in contrast to Inconel cladding used by CE. The stainless steel cladding has less potential to wear the Zircaloy CRGT by repeated or sliding contact than the Inconel cladding.

2.2.3 Westinghouse Control Rod Drive Mechanism

General design features (Figure 13) and functions are similar to CE control element drive mechanisms (see Section 1.2.3).

2.2.4 Westinghouse Guide Tube Assembly

The guide tube assembly (GTA) is part of the upper core structure. The GTA provides the housing and guide path for the RCCAs located above the core. Since each RCCA operates on a separate drive line, all guide tube assembly designs within each Westinghouse plant are similar. Therefore, there are no dual guide tube assemblies.

Figure 14 shows a typical guide tube assembly. In the bottom portion of the GTA, conduits for the control rods of the RCCA are provided by sheaths and split tubes. Figure 15 and cross-section B-B of Figure 16 show the plan views of the GTA for the 14x14 and 17x17 fuel design plants. A similar design is used for the 15x15 plants. The figures show that the sheaths and split tubes of the GTA provide continuous lateral support for the control rods over a length of approximately 40 inches above the fuel assembly.

Above the sheath/split tube region of the GTA, plates such as those shown in cross-section A-A of Figure 16 provide intermittent lateral support for the control rods in the parked position. Westinghouse suggested that the lateral

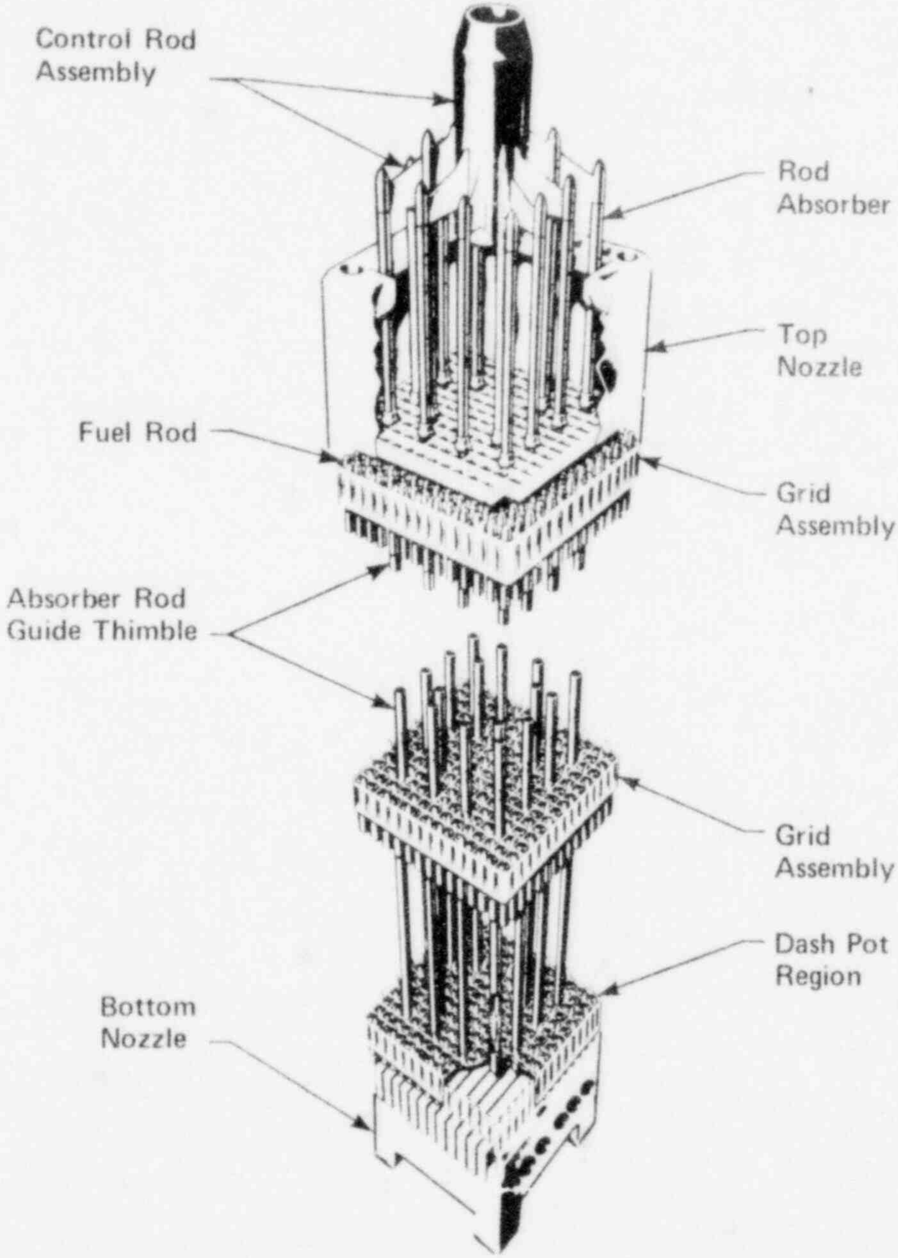

Figure 11. Cutaway of typical rod cluster control assembly inserted in fuel assembly.

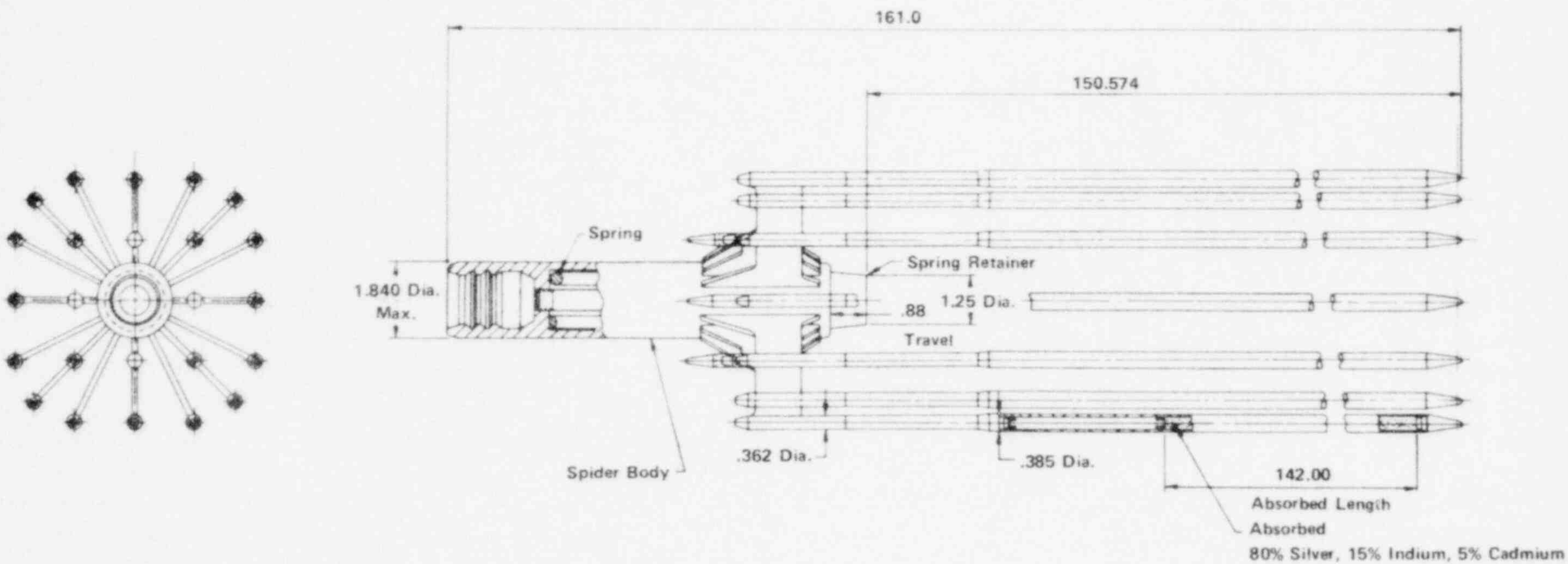

Figure 12. Westinghouse full-length rod cluster control assembly.

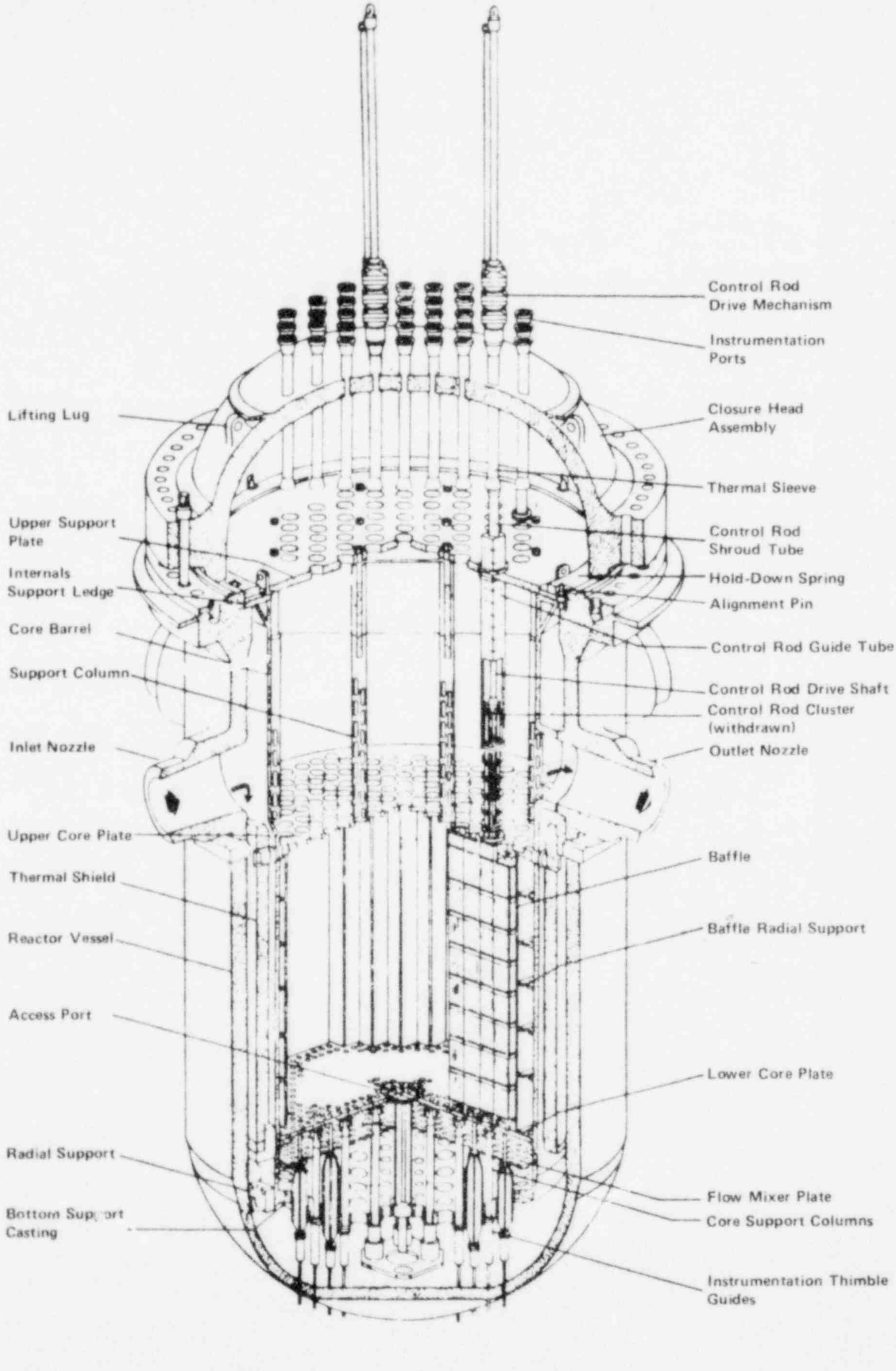

Figure 13. Westinghouse typical reactor vertical arrangement

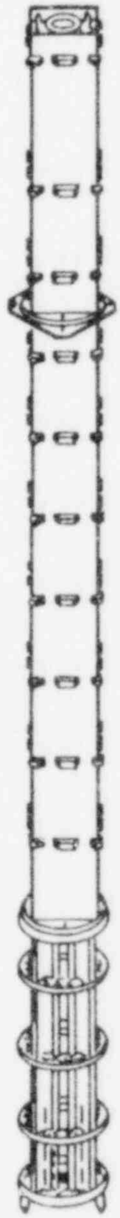

Figure 14. Westinghouse guide tube assembly.

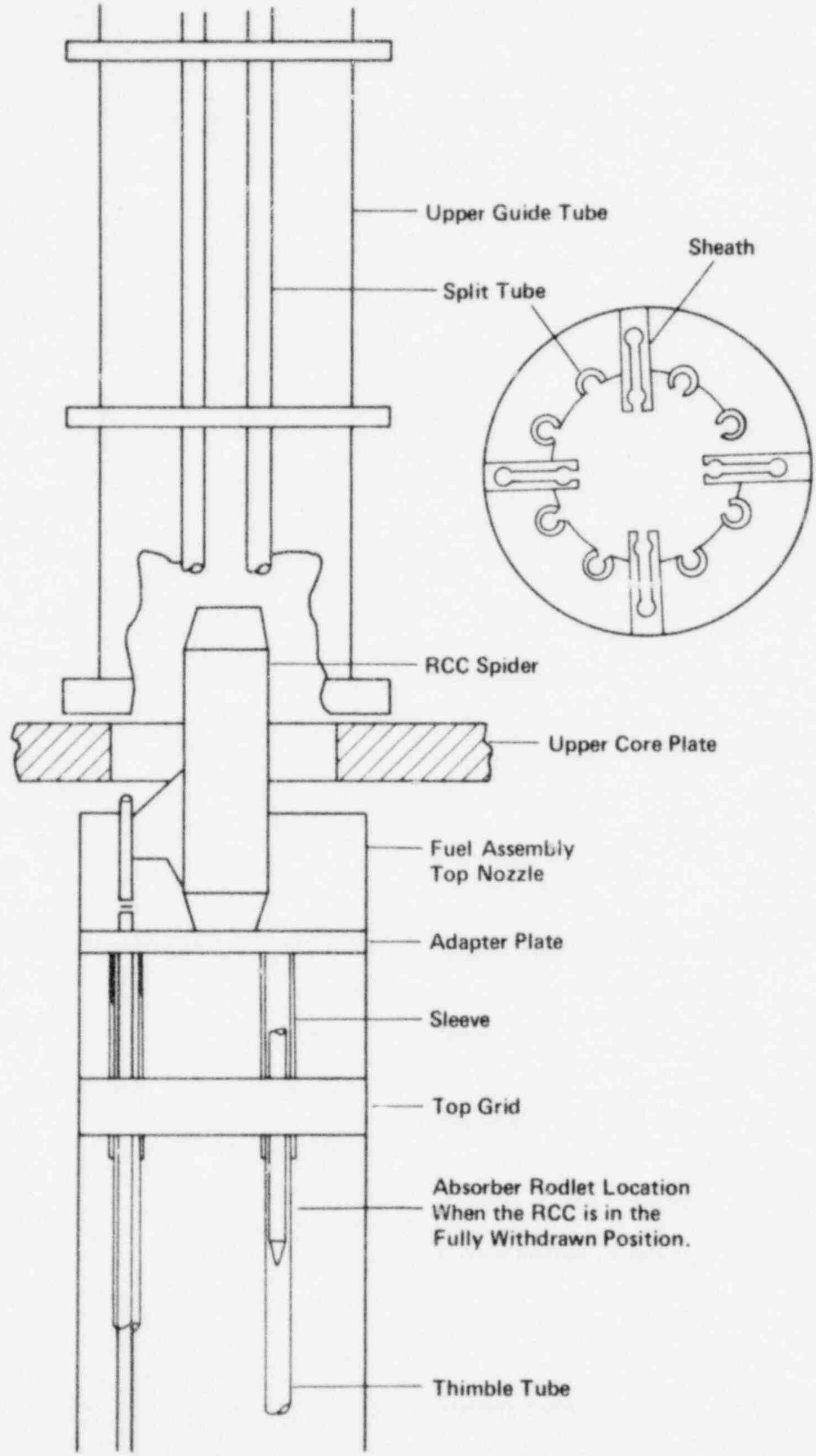

Figure 15. RCCA positions in fuel/guide tube assemblies.

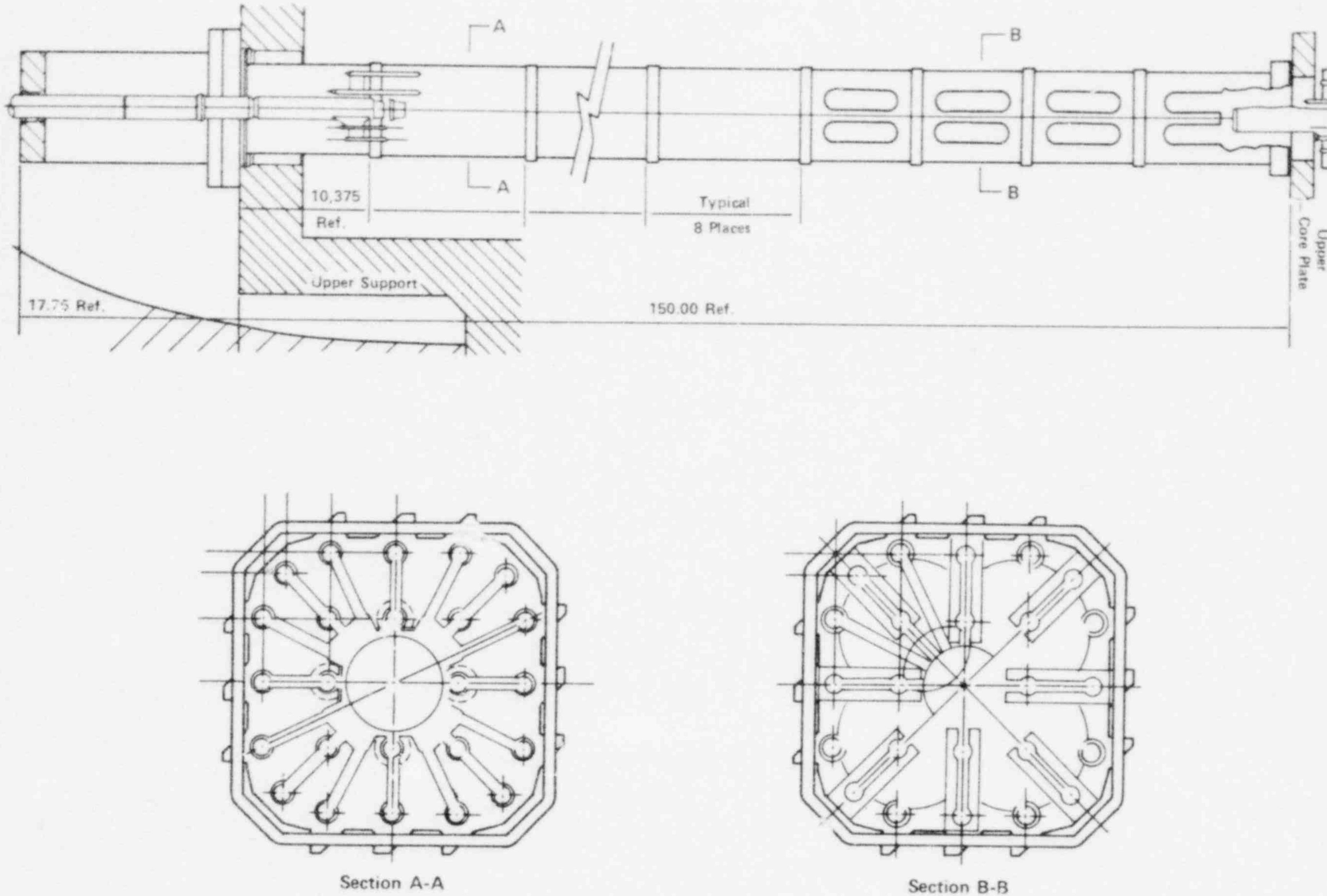

Figure 16. Westinghouse guide tube assembly cross sections.

support afforded to the control rods in the parked position reduces the propensity for CRGT wear in Westinghouse NSSS facilities.

2.3 Safety Considerations

The safety considerations for Westinghouse NSSS facilities are the same as for the CE facilities described in Section 1.3.

2.4 Operating Experience in Westinghouse NSSS Facilities

2.4.1 Discussion

In addition to the hot cell examinations described in Section 2.1, Westinghouse performed fuel surveillance programs which included guide tube wear determinations at two operating facilities. The operating facilities were of Westinghouse's two-loop design with 14x14 fuel assemblies.

Two types of eddy current tests (ECTs) were made to evaluate the guide tube wear. One type of ECT measurement was used to screen the guide tubes based on average wear indications. The technique is similar to that used by CE (see Section 1.9.3 and Appendix B). The second type of ECT measurement provides local azimuthal wear profiles at discreet axial positions and is also similar to the azimuthal probe measurements performed by CE at the CE-designed facilities.

As discussed in Appendix B, the ECT techniques developed and used by CE and Westinghouse to measure the guide tube wear in their respective plants differed somewhat in their specific design, application, and calibration standards. However, the principles and general applications of the ECT techniques used by CE and Westinghouse were compatible and oriented toward the same goal.

2.4.2 Guide Tube Wear Measurements

Table 9 shows the accumulated time of the Westinghouse 14x14 fuel operated in the two reactors prior to the examinations. The wear hour data are presented in terms of effective full-power hours (EFPH). Table 9 also shows the types of examinations performed.

Examination of the results from average wear ECT probe measurements on the CRGTs indicated a position of high localized wear in the axial direction. The axial location of the most severe wear corresponded to the position of the control rod tips when the RCCAs were in their fully withdrawn (parked) position.

The guide tubes exhibiting the highest wear indications from the average wear probe ECT examinations were also tested by the azimuthal wear probe. The results provided information from which the circumferential wear profile of the CRGTs at the position of maximum wear was determined. The azimuthal wear characteristics were used to evaluate the maximum wear in all CRGTs for which average wear indications were available at the axial position of the control rod tips with the RCCA in the "parked" position. The resultant data was searched to identify possible sources of nonrandomness in the wear results.

Table 9. Guide Tube Wear Measurements

Unit	Cycle No.	Core Location with RCCA*	EFPH	CRGT Sampled	Examination Performed
Unit 1	1	Yes	13002	Yes	Metallography and mechanical test
	1	No	12866	No	Dismantling and visual
	1 2	Yes No	13403 8208	Yes	Eddy current
	1 2	Yes Yes	13780 7452	Yes	Eddy current
	1 2	Yes Yes	14217 7404	Yes	Eddy current
	1 2	Yes Yes	14244 7401	Yes	Eddy current
	1 2	Yes No	13549 7907	Yes	Eddy current
	1 2	Yes Yes	13762 7533	Yes	Eddy current
	1 2	Yes No	13404 8215	Yes	Eddy current
Unit 2	1	Yes	13160	Yes	Eddy current
	1	Yes	13360	Yes	Eddy current
	1	Yes	13168	Yes	Eddy current
	1	Yes	13003	Yes	Eddy current
	1	No	15392	Yes	Eddy current

*RCCA control rods in parked position.

2.4.3 Results of Guide Tube Wear in Westinghouse NSSS Facilities

Based on a reduction of the preceding data, Westinghouse deduced the following:

(1) The worst wear is localized at the axial elevation corresponding to the position of the control rod tips when the RCCA is in the fully withdrawn (parked) position.

(2) The wear depth is time-dependent.

(3) The core location of the fuel assembly did not affect the wear observed in two plants.

(4) The upper guide tube geometry that guides the RCCA above the reactor core has a significant effect on the wear results.

(5) No through-wall wear was observed in the guide tubes of the Westinghouse 14x14 fuel design for up to two cycles of operation with the RCCAs in the parked position.

(6) No through-wall wear is projected for the guide tubes of the 14x14 fuel design for wear times greater than 250 weeks of continuous operation under a parked RCCA. In addition, the probability of a fuel assembly undergoing 250 weeks of operation under an RCCA is less than 0.005.

2.4.4 Westinghouse Guide Tube Wear Modeling

To bridge the gap for determining the potential for similar guide tube wear in other LOPAR Westinghouse fuel designs (15x15, 17x17), Westinghouse developed a mechanistic guide tube wear model. The model incorporated the suspected major phenomenological variables that would affect the guide tube wear. Some of the variables modeled included the turbulence and cross-flow velocities of the coolant at the fuel assembly exit regions, rod vibration, unsupported rod lengths, contact forces, rod stiffness, geometric effects, and stainless steel to Zircaloy wear characteristics. The developed wear model was shown to conservatively bound the ECT measurement results described in Section 2.4.2.

Using the guide tube wear model, Westinghouse calculated the maximum guide tube wear at the 95th percentile as a function of time for both the 14x14 and 17x17 fuel designs. The results of these analyses indicated that only those control rods guided by the split-tube configuration (see Figure 16) of the 17x17 fuel design could develop a through-the-wall hole in the guide tubes after nearly 250 weeks of operation under a parked RCCA. Conversely, the 14x14 fuel design and the control rods guided by the sheath configuration in the 17x17 fuel design could experience more than 250 weeks of operation under a parked RCCA before developing a wear hole in the guide tubes.

2.4.5 Stress Analysis of Guide Tube Wear

Based on the bounding wear projections of the wear model, Westinghouse calculated stress levels and stress intensities for the 14x14 and 17x17 fuel design as a function of guide tube wear. The analyses included conservative nonoperational

6 g handling loads, conditions 1 and 2 operational loads, and conditions 3 and 4 (seismic plus LOCA) accident loads. The stress analysis included the geometric effects of the wear scar and guide tube embrittlement resulting from the determined hydrogen uptake in the region of the wear scar (for through-the-wall wear, these determinations may be nonconservative).

2.5 Westinghouse Conclusions

Based on the preceding results, Westinghouse concluded that the integrity of the guide tube is maintained during normal operation, accident conditions, and nonoperational loading conditions for at least 250 weeks (> 3 cycles) of fuel assembly operation.

2.6 NRC Conclusions

The staff has concluded that the bounding analysis technique provides sufficient conservatisms when compared to the data base for the 14x14 fuel design with no wear hole formations. Likewise, the staff agrees that the mechanistic wear model developed by Westinghouse appears to include the major variables believed to be the source of control rod vibration. However, because of the complex nature of such forcing functions in the reactor environment and the uncertainties associated with the contact forces and the surface-to-surface wear rates, the extrapolation to other fuel designs (15x15, 17x17) needs further confirmation. These findings by the staff were conveyed to Westinghouse during the Control Rod Guide Tube Wear Meeting held in Bethesda, Maryland, on October 12, 1979.

2.7 Corrective Action

Because of basic differences between the CE-designed reactors and the Westinghouse-designed reactors, together with the examination results from the operating units and the wear predictions and stress analysis of worn guide tubes, Westinghouse has concluded that the mechanical integrity of the Westinghouse fuel assemblies is not impaired by local control rod guide tube wear. Therefore, no corrective action or design modifications to mitigate or eliminate CRGT wear has been proposed.

Based on the information provided by Westinghouse, the staff agrees that the need for any corrective action is not warranted at this time. However, additional confirmatory information supporting the absence of significant wear (no wear hole formation) in the 15x15 and 17x17 fuel designs has been requested by the staff (see Section 2.6).

2.8 Continuing Programs

In response to the staff's request (see Section 2.6), Westinghouse agreed to meet with the utilities involved to determine what onsite examinations should be implemented. When these plans have progressed, the specifics of the surveillance program will be communicated to the staff. The expected time frame for this information is January 1980.

2.9 References

1. Memorandum from L. C. Shao, NRC, to D. G. Eisenhut, NRC, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

2. Letter from D. G. Eisenhut, NRC, to T. M. Anderson, Westinghouse, June 14, 1978. Available in NRC Public Document Room Subject File No. RD 8-2 (Westinghouse) for inspection and copying for a fee.

3. Letter from T. M. Anderson, Westinghouse (NS-TMA-1936), to D. G. Eisenhut, NRC, September 12, 1978. This document is not publicly available because it contains proprietary or national security information.

4. Letter from T. M. Anderson, Westinghouse (NS-TMA-1992), to D. G. Eisenhut, NRC, June 27, 1979. This document is not publicly available because it contains proprietary or national security information.

5. Letter from T. M. Anderson, Westinghouse (NS-TMA-2102), to D. G. Eisenhut, NRC, June 27, 1979. This document is not publicly available because it contains proprietary or national security information.

6. Letter from B. K. Grimes, NRC, to T. M. Anderson, Westinghouse, September 7, 1979. Available in NRC Public Document Room Subject File No. RD 8-2 (Westinghouse) for inspection and copying for a fee.

3. BABCOCK & WILCOX NSSS FACILITIES

3.1 Introduction to Guide Tube Wear in Babcock & Wilcox NSSS Facilities

Because of the control rod guide tube (CRGT) wear observed at the Combustion Engineering (CE) NSSS facility (Millstone Unit 2) in December 1977 (see Section 1.1), the staff contacted Babcock & Wilcox (B&W) on December 28 and 29, 1977, to discuss guide tube wear problems that B&W reactors may have experienced. From these discussions, it was determined that the B&W examinations (see Section 3.4) aimed at determining the extent of CRGT wear were encouraging but not sufficient for the staff to reach a final conclusion.

However, the staff was able to conclude (Ref. 1) that the severity of guide tube wear at the B&W-designed plants should be less severe than the CRGT wear observed at the Millstone Unit 2 (CE) plant. This preliminary conclusion was based on three considerations: (1) the lack of indicated through-the-wall wear indications from the air tests (see Section 3.4), (2) the design differences between the CE plants and the B&W plants, and (3) the design similarities between the B&W plants and the Westinghouse plants. The above design features and their relationships to the severity of CRGT wear are discussed further in Sections 2.2.2 and 2.2.4. Of the above three considerations, the last two figured predominantly in the staff's preliminary conclusion.

To provide further assurance that significant wear was absent from the B&W facilities, the staff formulated specific questions (Ref. 2) addressed at resolution of this concern. B&W replied to Reference 2 via Reference 3. The B&W reply was determined insufficient (Ref. 4) for the staff to finalize their review.

Based on the staff's preliminary conclusions, no corrective action was warranted at the time. However, at B&W's request (Ref. 6), the staff transmitted their concerns (Refs. 6-11) to all licensees of B&W NSSS facilities for additional confirmation on the absence of CRGT wear in their plants (see Section 3.7).

3.2 Design and Functions of Guide Tubes and Related Components

3.2.1 Babcock & Wilcox Fuel Assembly

The B&W fuel assembly design (Figure 17) primarily used in licensed reactors is a 15x15 array. Therefore, only the B&W 15x15 fuel design is discussed herein.

The structural frame of the B&W fuel assembly consists of guide tubes, spacer grids, and end fittings. Each 15x15 fuel assembly has 16 Zircaloy guide tubes having an outside diameter of 0.530 inch with a wall thickness of 0.016 inch. The guide tubes are located as shown in Figure 17, with the center position in the assembly reserved for instrumentation. The guide tubes are secured to the upper and lower end fittings. The basic design functions of the guide tubes are to (1) provide continuous guidance for the control rods, (2) provide structural support for the fuel assembly, and (3) provide the path for the coolant flow to the control rods. Unlike the Westinghouse and CE guide tube designs, the buffer system for decelerating the control rods is part of the control rod drive mechanism (CRDM). Other than this buffer feature, the B&W fuel assembly

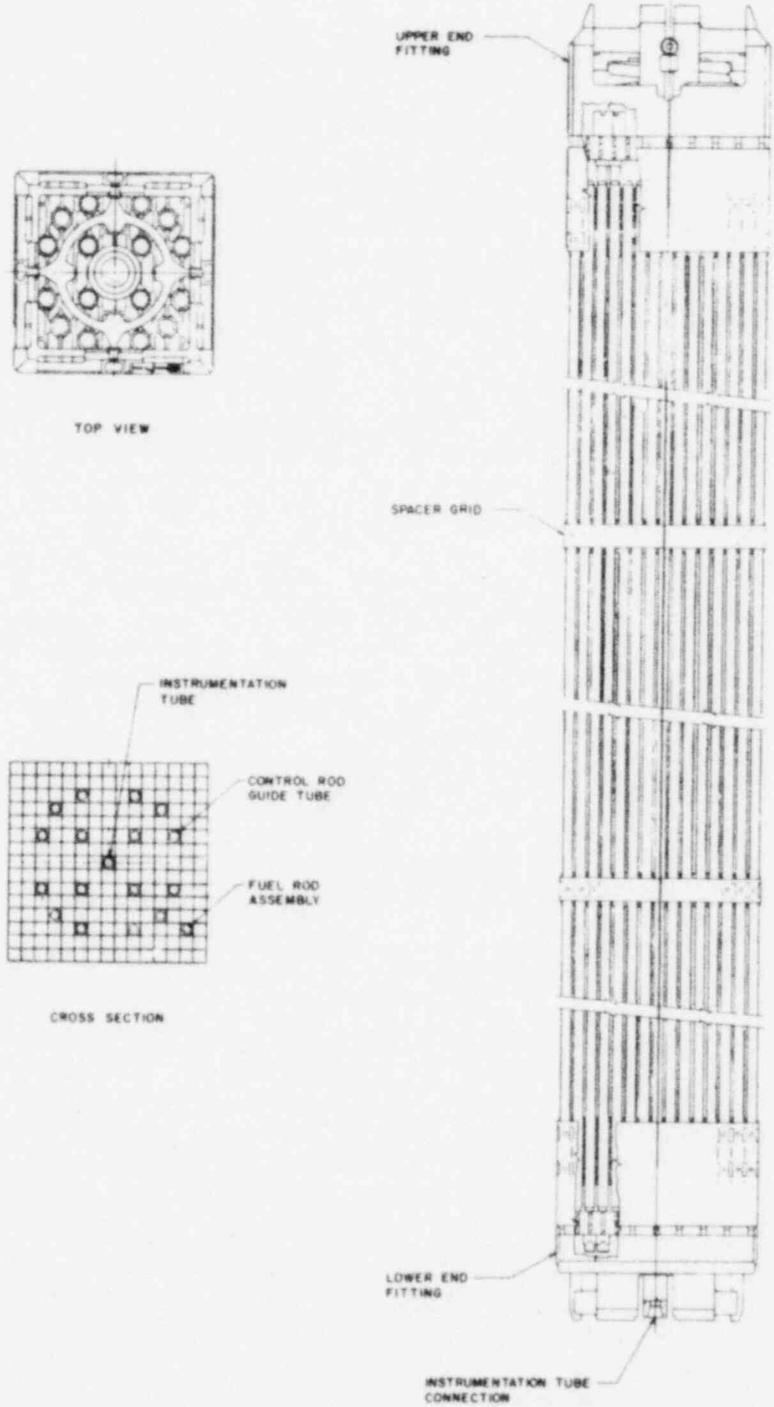

Figure 17. Babcock & Wilcox fuel assembly.

structural integrity must be capable of sustaining basically the same type of loadings as those in the CE and Westinghouse fuel assemblies (see Section 2.2.1).

3.2.2 Babcock & Wilcox Control Rod Assembly

The B&W control rod assembly (CRA) consists of 16 stainless steel tubes having a diameter of 0.440 inch (see Figure 18). The CRA includes a hub and spider arrangement with neutron adsorber material in the stainless steel tubes that insert into the Zircaloy control rod guide tubes.

The control rods are designed to withstand all operating loads, including those resulting from hydraulic force, thermal gradients, and reactor trip deceleration. Because of their length and potential for deviation from straightness over the entire length of the rod, some interference between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible and only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, control rod assemblies will not encounter significant frictional resistance to their motion in the guide tubes.

Two notable design differences between the B&W CRA and the CE CEAs are of interest. Conversely, the design differences become similarities when comparing the B&W CRA and the Westinghouse RCCA designs. The design features are: (1) the greater flexibility (less stiffness) afforded by the smaller diameter B&W and Westinghouse control rods, and (2) the use of stainless steel cladding material in the B&W and Westinghouse control rods rather than Inconel used by CE. Stainless steel has less of a CRGT wear potential for repeated or sliding contact between the control rods and the Zircaloy guide tubes.

3.2.3 Babcock & Wilcox Control Rod Drive Mechanism

The control rod drive mechanisms provide for controlled withdrawal or insertion of the control rod assemblies out of or into the core and are capable of rapid insertion or trip. Typically, the B&W drive mechanisms are sealed, reluctance motor-driven screw units. The speed at which the control rod is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the shim safety drive mechanism releases the CRA and supporting CRDM components permitting gravity to drive the CRA into the core. The reactivity is reduced during such a rod insertion at a rate sufficient to control the core under any operating transient or accident condition. The control rod is decelerated at the end of the rod trip insertion by a buffer assembly in the CRDM upper housing. The buffer assembly supports the control rod in the fully inserted position. Even though the B&W CRDM and buffer system differ significantly from the CE and Westinghouse designs, severe distortion of the fuel assembly guide tubes, or interference from hole formations in the guide tubes, could impede the gravity drop feature of this mechanism.

3.2.4 Babcock & Wilcox Column Weldment

The assembly providing the housing and guide path for the CRAs above the core is shown as the column weldment in Figure 19. Figure 20 shows a cross-sectional view of the column weldment. As can be observed from Figure 20, the control

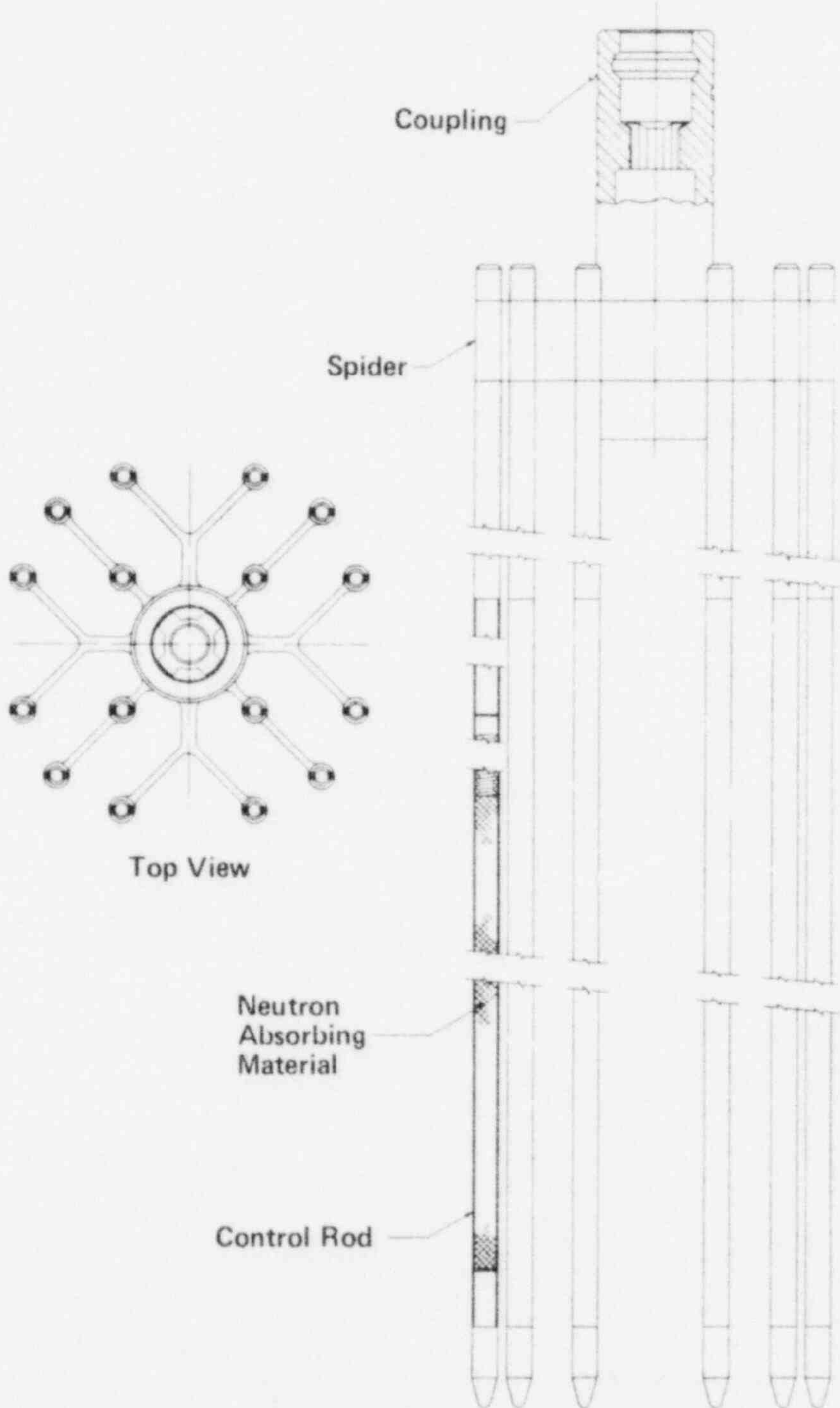

Figure 18. Babcock & Wilcox control rod assembly.

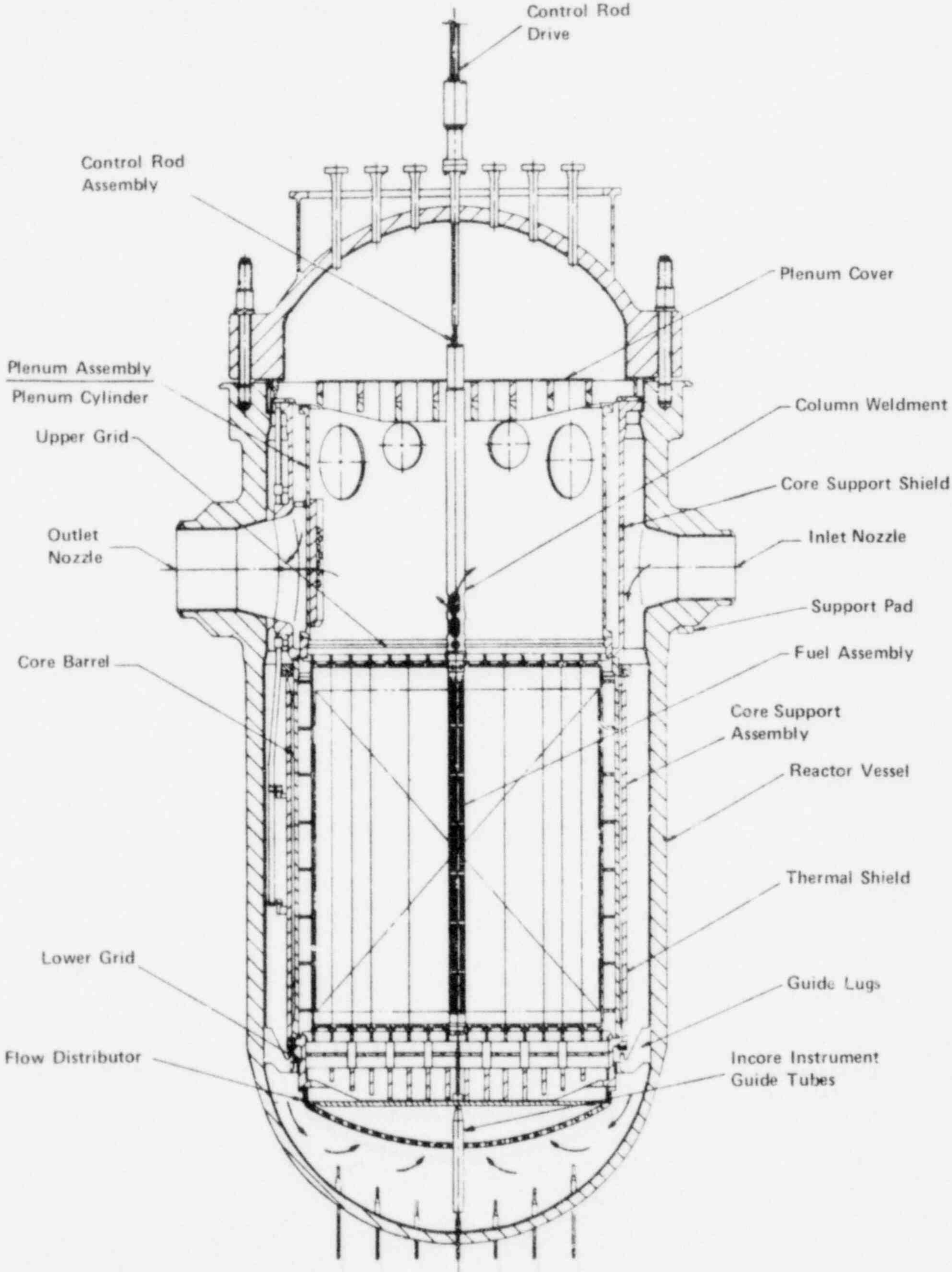

Figure 19. Typical Babcock & Wilcox reactor vertical arrangement.

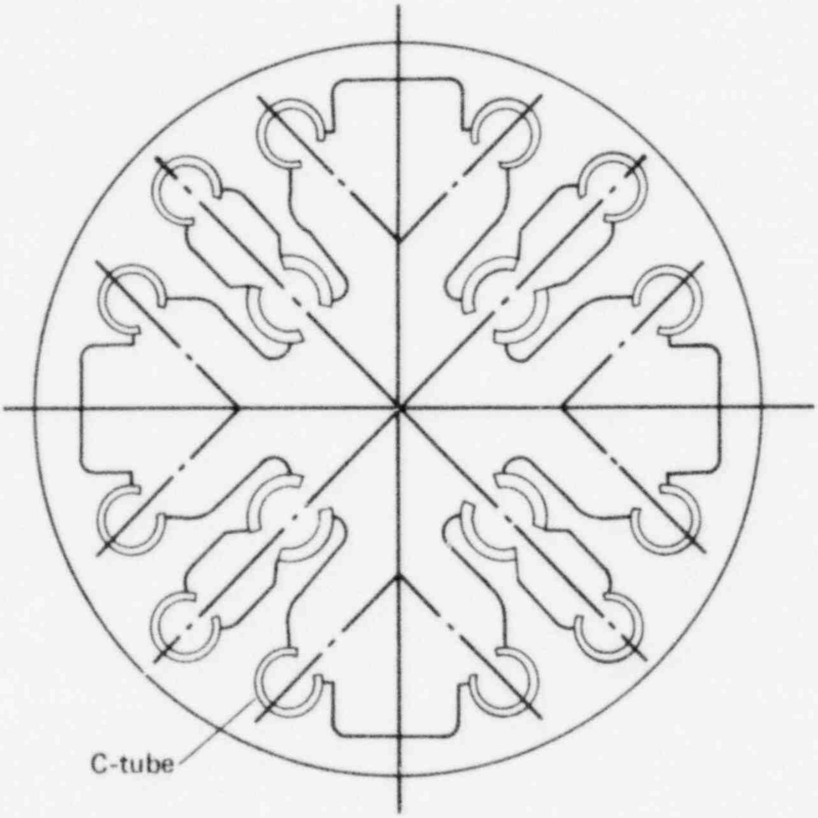

Figure 20. Cross-sectional view of column weldment.

rods are contained within C-tubes. The C-tubes are fixed by support plates spaced at approximately 11-inch intervals in the vicinity of the exiting CRGT coolant flow. It should be noted that the B&W C-tube design is similar to the Westinghouse split-tube design shown in Figure 16. Thus, the full-length B&W C-tube configuration provides continuous lateral support for the control rods, particularly in the cross-flow region that is believed to be a source of excitation for vibration of the control rods. This lateral support of the control rods, together with the two notable design features discussed in Section 3.2.2, were dominant factors considered in the staff's preliminary findings discussed in Section 3.1.

3.3 Safety Considerations

The safety considerations for B&W facilities are the same as for the CE facilities described in Section 1.3.

3.4 Operating Experience in Babcock & Wilcox NSSS Facilities

The surveillance experience on worn guide tubes in B&W-designed NSSS facilities consists of air testing of 16 guide tubes from an Oconee Unit 1 15x15 fuel assembly that had experienced one cycle of operation under a control rod assembly (Ref. 3). No escaping air bubbles were observed until the water in the guide tubes had been displaced to the level of the flow holes.

Out-of-reactor flow tests of a Mark C (17x17) prototype fuel assembly were run. Clamshell sectioning of two guide tubes from the 17x17 fuel assembly that had undergone the 1000-hour flow tests under a control rod assembly indicated no significant wear at the elevations at which the control rod tips were in the fully withdrawn position.

3.5 Babcock & Wilcox Conclusion

On the basis of the tests described in Section 3.4, B&W concluded that significant CRGT wear was absent in the B&W NSSS facilities.

3.6 NRC Conclusions

Because of basic design differences between the B&W-designed plants and the CE-designed plants, the control rod guide tube wear is of less immediate concern in the B&W NSSS facilities. However, additional evidence on the absence of significant wear is required for the staff to complete its review.

3.7 Corrective Action

Because there were no evident conclusive indications of wear similar to that found at the CE reactors, neither B&W nor the licensees of the B&W plants initiated quantitative inspection programs to specifically detect guide tube wear in the B&W-designed NSSS facilities.

The staff has requested (see Section 3.8) that the licensees inspect their fuel assemblies for indication of CRGT wear or provide additional information and tests to support the absence of CRGT wear (Refs. 6-11).

3.8 Continuing Programs

The staff has requested (Refs. 6-11) that the licensees commit to a fuel surveillance program to specifically determine the extent of CRGT wear in the B&W NSSS facilities. This program is to be reviewed by the staff prior to implementation. The staff review will be coordinated for all B&W plants with operating licenses and those B&W plants presently being considered for operating license approval. A meeting was scheduled at NRC headquarters in Bethesda, Maryland, on December 20, 1979. This meeting provided the licensees with an opportunity to present and discuss their fuel surveillance programs.

3.9 References

1. Memorandum from L. C. Shao, NRC, to D. G. Eisenhut, NRC, January 18, 1978. Available in NRC Public Document Room for inspection and copying for a fee.

2. Letter from B. K. Grimes, NRC, to J. H. Taylor, B&W, June 13, 1978. USNRC Accession No. 7910240826. Available in NRC Public Document Room for inspection and copying for a fee.

3. Letter from J. H. Taylor, B&W, to B. K. Grimes, NRC, January 12, 1979. USNRC Accession No. 7901190081. Available in NRC Public Document Room for inspection and copying for a fee.

4. Letter from B. K. Grimes, NRC, to J. H. Taylor, B&W, August 22, 1979. Available in NRC Public Document Room Subject File No. RD 8-2 (B&W) for inspection and copying for a fee.

5. Letter from J. H. Taylor, B&W, to B. K. Grimes, NRC, September 10, 1979. USNRC Accession No. 7909140383. Available in NRC Public Document Room for inspection and copying for a fee.

6. Letter from R. W. Reid, NRC, to R. C. Arnold, MED, Subject: Control Rod Guide Tube Wear in Facilities Designed by B&W, Docket No. 50-289, November 23, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

7. Letter from R. W. Reid, NRC, to J. J. Mattimore, SMUD, Subject: Control Rod Guide Tube Wear in Facilities Designed by B&W, Docket No. 50-312, November 23, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

8. Letter from R. W. Reid, NRC, to W. Cavanaugh, AP&L, Subject: Control Rod Guide Tube Wear in Facilities Designed by B&W, Docket No. 50-313, November 23, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

9. Letter from R. W. Reid, NRC, to L. E. Rod, TEC, Subject: Control Rod Guide Tube Wear in Facilities Designed by B&W, Docket No. 50-346, November 23, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

10. Letter from R. W. Reid, NRC, to W. P. Stewart, FPC, Subject: Control Rod Guide Tube Wear in Facilities Designed by B&W, Docket No. 50-302, November 23, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

11. Letter from R. W. Reid, NRC, to W. O. Parker, DPL, Subject: Control Rod Guide Tube Wear in Facilities Designed by B&W, Docket Nos. 50-269, 50-270, and 50-287, November 23, 1979. Available in NRC Public Document Room for inspection and copying for a fee.

4. COMBUSTION ENGINEERING NSSS FACILITIES WITH EXXON/WESTINGHOUSE FUEL

4.1 Discussion

Three Combustion Engineering (CE) facilities are scheduled to introduce Exxon Nuclear Company (ENC) or Westinghouse (W) fuel in their upcoming fuel cycle. The Fort Calhoun and Maine Yankee plants will use ENC fuel, and Millstone Unit 2 will use Westinghouse-designed fuel.

4.2 Operating Experience in Combustion Engineering NSSS Facilities With Exxon/Westinghouse Fuel

The next fuel cycle will be the first use of ENC and Westinghouse fuel in operating CE NSSS facilities. Therefore, no operating experience with these fuel designs exists.

4.3 NRC Consideration for Continued Operation

As discussed in Section 1.5, the Fort Calhoun plant has not experienced significant CRGT wear in previous cycles of operation. The absence of CRGT wear has been attributed to design and operational features unique to the Fort Calhoun facility. These features, being inherent to the Fort Calhoun plant, are essentially insensitive to fuel design. Therefore, the staff review of the upcoming Fort Calhoun refueling will primarily address the mechanical, thermal-hydraulic, and physics compatibility of the ENC fuel, and the methods of analysis.

However, since both the Maine Yankee and Millstone Unit 2 plants have been susceptible to significant CRGT wear, an internal staff memorandum was issued that discussed specific items to be considered during the staff review of the proposed reload fuel designs and core analyses (Ref. 1). These items will be resolved prior to NRC approval of the next cycle of operation.

4.4 Continuing Programs

The staff will require that the Maine Yankee and Millstone Unit 2 licensees commit to an end-of-cycle fuel surveillance program specifically addressed at monitoring guide tube wear (Ref. 1). The surveillance programs will require NRC approval prior to implementation.

4.5 References

1. Memorandum from P. Check, NRC, to R. Reid, NRC, September 13, 1979. Available in NRC Public Document Room for inspection and copy[illegible] for a fee.

5. WESTINGHOUSE NSSS FACILITIES WITH EXXON FUEL

5.1 Discussion

Five Westinghouse NSSS facilities now use Exxon Nuclear Company (ENC)-designed fuel in their current operating cycle. These facilities are H. B. Robinson Unit 2, D. C. Cook Unit 1, R. E. Ginna, Prairie Island Unit 2, and Kewaunee. In addition, ENC reload fuel will be introduced into the Prairie Island Unit 1 facility during 1980.

5.2 Operating Experience in Westinghouse NSSS Facilities With Exxon Fuel

At the present time, ENC has not reported guide tube wear inspection results on their fuel that has operated in Westinghouse NSSS facilities (see Section 5.5).

5.3 NRC Consideration for Continued Operation

The staff has considered evidence from all PWR NSSS facilities in justifying continued operation of the Westinghouse NSSS facilities with ENC fuel designs. The staff position, in addition to the design considerations discussed below, has allowed an increased operational data base of the irradiated ENC fuel. The increased data base should therefore enable a more quantitative assessment as to the extent of, or absence of, CRGT wear in the ENC fuel designs with appreciable exposure (cycles of operation).

The staff's design considerations noted that the CRGT wear observed in the Combustion Engineering (CE)-designed NSSS facilities was attributed to the CEA control rods vibrating and wearing against the softer Zircaloy control rod guide tubes. Thus, the presence of significant CRGT wear could be considered as an effect of CEA vibration. The cause of the wear (CEA vibration) could be more appropriately attributed to other design features inherent to the specific NSSS facility design.

It was noted in Section 2.2 that three design features in the Westinghouse NSSS facilities contributed to apparently less severe guide tube wear in these facilities. The features are: (1) more flexible control rods, (2) lateral support to the control rods in the "parked" position, and (3) stainless steel cladding on the control rods. All are independent of ENC fuel design.

5.4 Corrective Action

Based on the above considerations, the staff determined that no corrective action in the form of design modifications to mitigate CRGT wear is warranted at this time.

5.5 Continuing Programs

To provide assurance that significant CRGT wear is absent in the ENC fuel designs in the Westinghouse-designed NSSS facilities, the staff requested that ENC provide confirmatory evidence (Ref. 1). ENC's reply to the staff request (Ref. 2) cited the absence of obvious CRGT wear in the limited visual examinations of Westinghouse-designed fuel from the H. B. Robinson Unit 2 plant and the geometric similarities between the Westinghouse fuel design and the ENC fuel design. ENC's reply also stated that no surveillance of the inside of the ENC guide tubes had been performed. However, ENC had planned borescopic examinations on the H. B. Robinson Unit 2 fuel during the 1979 outage. If the examinations at H. B. Robinson Unit 2 revealed significant CRGT wear, the ENC fuel in R. E. Ginna and D. C. Cook Unit 1 would be inspected. The results of these inspections would be reported as definitive response to Reference 2.

The results of the planned ENC inspections have not been submitted for staff review. In accordance with telephone communications between the staff and Exxon personnel on November 21 and December 3, 1979, inspection results from the H. B. Robinson plant will include ECT measurements. The inspection results will be communicated to the staff by January 1980.

5.6 NRC Conclusion

Pending the results of the ENC guide tube inspection, the staff holds in abeyance evaluation of the conclusion that significant CRGT wear is absent from ENC fuel used in Westinghouse NSSS facilities.

5.7 References

1. Letter from B. K. Grimes, NRC, to W. Nechodom, ENC, July 20, 1978, USNRC Accession No. 7811060121. Available in NRC Public Document Room for inspection and copying for a fee.

2. Letter from G. Owsley, ENC, to B. K. Grimes, NRC, September 15, 1978, USNRC Accession No. 780810252. Available in NRC Public Document Room for inspection and copying for a fee.

APPENDIX A

CONTROL ROD GUIDE TUBE WEAR CHRONOLOGY OF ACTIONS

December 13, 1977	Cracks observed in control element assembly (CEA) guide tube during fuel inspection program at Millstone Unit 2.
December 15, 1977	Northeast Nuclear Energy Company (NNECO) and Combustion Engineering (CE) met with NRC engineers presenting preliminary findings. We request general meeting to evaluate the guide tube problem.
December 19, 1977	Representatives from Baltimore Gas and Electric (BG&E), Florida Power and Light (FP&L), Maine Yankee Atomic Power Company (MY), NNECO, Omaha Public Power District (OPPD), and CE met with NRC to present Millstone Unit 2 data on CEA guide tube wear.
December 20, 1977	All affected CE licensees with operating facilities were notified to: (1) insert 1/7 of all CEAs, not fully inserted, at least 10 steps each day, and (2) provide request for amendment to allow CEA insertion beyond the present "full out" position.
December 23, 1977	CE provides proprietary version of slides used at December 19, 1977 meeting with the staff.
December 27, 1977	Received request for Technical Specification (TS) changes from all operating CE licensees.
January 4, 1978	BG&E provides information on necessary inhibits to allow CEA insertion of 3 inches. Also accepts proposed TS telecopied by staff.
January 6, 1978	Amendments for Calvert Cliffs Units 1 and 2, Fort Calhoun, Maine Yankee, and St. Lucie were issued authorizing CEA insertion 3 inches.
January 12, 1978	Second meeting with facilities and CE was held. CE presented more ECT inspection data and described the possible temporary fix; sleeving of selected CEA guide tubes.
January 16, 1978	DOR requests IE to ensure the intent of January 6, 1978 amendment letter is implemented at each facility.

January 17, 1978	CE provides nonproprietary version of slides used at December 19, 1977 meeting with the staff.
January 18, 1978	Twenty-day letters sent to BG&E, FP&L, MY, and OPPD requiring, pursuant to 10 CFR 50.54(f), that "justification that excessive guide tube wear does not exist in your facility, or, if unable to assure that such wear does not exist, justification that continued operation of the facility would not create undue risk...." NNECO was sent a similar letter requiring "additional justification for return to operation...."
January 1978	Engineering Branch (EB) contracts with Idaho National Engineering Laboratory to provide confirmatory analyses of the guide tube problem.
February 3, 1978	DOR provides operating experience Memorandum No. 11 on guide tube wear to DSS.
February 6, 1978	The staff prenoticed the "resolution of the operational problems related to CEA guide tube wear prior to return to power operation" for Calvert Cliffs Unit 1 and Millstone Unit 2.
February 14, 1978	All responses to 20-day letters received. (Note: Some responses late due to adverse weather conditions.) BG&E and FP&L submitted CEN-79-P for the operating reactors, Calvert Cliffs Unit 2 and St. Lucie. NNECO references CEN-82-P for return to operation of Millstone Unit 2. (CEN-82-P references Calvert Cliffs Unit 1, but it was not submitted by BG&E.) OPPD documents "that there is no significant guide tube wear in the Fort Calhoun Station Unit 1 fuel assemblies." MY submittal states that "the amount of guide tube wear on MY fuel that is similar to the fuel currently residing in the reactor is significantly less than the wear experienced by fuel in earlier cycles."
February 15, 1978	BG&E met with the staff to inform us that they plan to sleeve the CEA guide tubes in 110 fuel assemblies.
February 17, 1978	BG&E submits CEN-83(B)-P on "Calvert Cliffs Unit No. 1 Reactor Operation With Modified CEA Guide Tubes."
February 17, 1978	CE recommends that BG&E place a hold on the handling of 14 selected fuel assemblies in the Calvert Cliffs Unit 1 reactor.

February 21, 1978	BG&E and CE met with staff to answer our concerns on sleeving. At conclusion, the staff informed BG&E that operation with sleeved CEA guide tube involves an unreviewed safety question.
February 24, 1978	NNECO and CE met with the staff to present their plan to sleeve approximately 35 guide tubes. The staff notified them that operation with a sleeved CEA guide tube involves an unreviewed safety question.
February 27 and February 28, 1978	FP&L and BG&E, respectively, notified the staff that "preliminary results of the CE analysis of the more severely worn assemblies identified to date indicate that the stress criteria established in some guide tubes during the limiting seismic excitation (SSE)." They also stated that test program "results continue to support the conclusion that guide tube wear will not prevent CEAs from inserting following an SEE."
March 8, 1978	NNECO provides additional information on CEA guide tube wear including Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P.
March 8, 1978	NNECO requires ar endment to authorize operation of Millstone Unit with sleeves installed in the CEA guide tubes. The letter also transmits CEN-80(N)-P, "Millstone Unit 2 Reactor Operation With Modified CEA Guide Tubes."
March 15, 1978	NNECO submits additional information on CEA guide tube wear including Amendment 2-P to CEN-79-P, CEN-80(N)-P, and CEN-83(B)-P and sleeving procedures used at Calvert Cliffs Unit 1 and Millstone Unit 2.
March 16, 1978	NNECO responds to staff questions and submits Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P.
March 16, 1978	BG&E response to staff questions, including Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P.
March 17, 1978	BG&E request for amendment to operate with five sleeved CEA guide tubes and to remove all part length CEA plus response to staff questions, including Amendment 2-P to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P, "Additional Information on Guide Tube Wear."

March 20, 1978	BG&E submits revised reload analyses for Cycle 3 operation.
March 20, 1978	BG&E responds to staff questions including results of visual examinations of sleeved guide tubes and worst case wear observed at Calvert Cliffs Unit 1.
March 29, 1978	NRC provides MY with questions on location of worst possible wear of the guide tubes, fracture mechanics, type analysis, and revised seismic analysis.
March 31, 1978	NRC issues Amendment 32 to BG&E authorizing Cycle 3 operation for Calvert Cliffs Unit 1 with sleeved CEA guide tubes.

APPENDIX B

NONDESTRUCTIVE TESTING TECHNIQUES

1. EDDY CURRENT TESTS

The eddy current test (ECT) techniques discussed herein vary with the special requirements of the specimen being tested. Different designers will also usually favor somewhat different ECT fixtures or components for basically similar test specimens. Likewise, the calibration standards used to evaluate and compare the test specimens with manufactured simulations vary from designer to designer and test specimen to test specimen. Therefore, even though different ECT techniques exist, the principles of the eddy current tests are the same.

The ECT technique uses a probe containing a wire coil supplied (driven) by an electronic oscillator that causes an alternating electromagnetic field to be radiated from the probe. The field will induce eddy currents in a conductive material brought near it. The eddy currents can reveal differences in physical properties because (1) induced eddy currents in the specimen oppose the driving field (impedance resistance), and (2) once a field balance is established over a known sound material (calibration standard), any changes in the induced field will affect the impedance of the driving field which can be monitored to evaluate changes in material, discontinuities, or probe to material spacing.

The ECT techniques and the calibration standards used by Combustion Engineering and Westinghouse, although different, were directed toward the same goal. As implied above, the ECT technique is sensitive to the electri:al conductivity of the material. The density variations resulting from hydr gen pickup and local cold working of the specimen tested may therefore affect the ECT results. These effects have been conservatively accounted for in the ECT results discussed in this report.

2. VISUAL EXAMINATION

Both closed-circuit TV and periscope have been used for remote visual inspections at the CE facilities. Specific features can be documented by videotape or 35 mm photographs. The visual examinations observed by the staff at Calvert Cliffs Unit 2 were of excellent clarity.

NRC Form 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG 0641

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Control Rod Guide Tube Wear In Operating Reactors - Operating Experience Report

5. DATE REPORT COMPLETED
MONTH December YEAR 1979

DATE REPORT ISSUED
MONTH April YEAR 1970

6. (Leave blank)

7. AUTHOR(S)
R. R. Riggs

8. (Leave blank)

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

13. TYPE OF REPORT
Technical Report

PERIOD COVERED (Inclusive dates)
December 1977 to December 1979

14. (Leave blank)

15. SUPPLEMENTARY NOTES

16. ABSTRACT (200 words or less)

Evidence of control rod guide tube wear has been observed in operating pressurized water reactors. The cause of this wear is identified as flow-induced vibration of the control rods. This report describes the measures being taken by both the industry and the NRC to deal with this matter. The staff also presents its technical positions and requirements to support continued operation of the plants as of December 1979 pending completion of this generic effort.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)
Unclassified

21. NO. OF PAGES

22. PRICE
S

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
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