Independent Review of

"Control Rod Insertion Following a Cold Leg LBLOCA, D.C. Cook, Units 1 and 2"

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Introduction

WCAP-15245 (proprietary) and WCAP-15246 (non-proprietary), dated May 28, 1999, present the results of Westinghouse analysis of control rod insertion following a cold leg large break loss of coolant accident (LBLOCA) at D. C. Cook, Units 1 and 2. SCIENTECH performed a third party independent review of these Westinghouse reports, which support rod cluster control assembly (RCCA) insertability following certain design basis (DB) and leak before break (LBB) large cold leg LOCA events combined with a seismic event. The intent of this third party review is to provide assurance that the conclusions of the report are correct and in conformance with industry safety practice, as RCCA insertion credit following a cold leg LBLOCA has not been previously pursued by Westinghouse. The intended use of the results of the Westinghouse analysis is to credit control rod insertion for purposes of determining whether there is a return to criticality following switch to hot leg injection.

Approach

SCIENTECH reviewed the technical approach, assumptions, analysis methods and results presented in WCAP-15245 to assure that there is no safety impact due to recriticality following a cold leg LBLOCA at the D. C. Cook plant. SCIENTECH did not perform a quality assurance verification of the work supporting the subject Westinghouse topical report.

The work scope included a review of the Westinghouse methodology compared to industry practice and pertinent Babcock & Wilcox / Framatome Technologies Inc. (B&W/FTI) LOCA analysis methodology, which relies upon the insertion of the control rods following the blowdown phase of a LBLOCA. B&W/FTI received NRC approval for their LOCA evaluation model for recirculating steam generator plants in the early 1990s.

Review Results

SCIENTECH performed the following review tasks:

1. Reviewed the approach used in the Westinghouse Topical Reports (WCAPs -15245 and -15246) and provided an independent assessment of the technical approach used. The emphasis was on technical adequacy of the methods used, and the regulatory (licensing/safety) aspects such as leak before break, risk informed implications, etc. SCIENTECH did not perform an independent verification of the Westinghouse calculations.
2. Investigated and reported on the Framatome methodology, i.e. how does this methodology compare to that used by Westinghouse.

Internal to the reactor vessel, control rod insertion time can be influenced by distortion of the guide tubes or of the fuel assemblies. Westinghouse performed analysis of upper internals guide tube loads due to combined LOCA and seismic forces. They also calculated fuel assembly displacements and grid loads due to combined LOCA and seismic loads. These analyses show that there are no loads which result in displacements sufficient to preclude control rod insertion.

Break Sizes and Locations Considered

Two LBB breaks and three DB breaks were considered:

LBB Breaks: 60 in^2 accumulator line break
98 in^2 pressurizer surge line break

DB Breaks: 144 in^2 reactor vessel inlet nozzle break
144 in^2 reactor vessel outlet nozzle break
594 in^2 reactor coolant pump outlet nozzle double-ended guillotine break

The break not considered is the double-ended guillotine hot leg break at the steam generator inlet nozzle. For return to criticality considerations, this break is irrelevant since the injection flow will go to the break and will not contribute to decreasing the boron concentration in the core. Thus, the spectrum of breaks considered includes all of those which could result in blowdown forces being generated sufficient to preclude control rod insertion following a cold leg LBLOCA. The 144 in^2 reactor vessel outlet nozzle break is actually a hot leg break, so there was no need to include this case to reach the conclusions drawn in the report.

It would seem that the double-ended guillotine break at the reactor coolant pump outlet nozzle will generate the maximum blowdown forces. Westinghouse, however, also included the smaller breaks at the reactor vessel inlet and outlet nozzles in their analysis. The maximum break area is smaller at these locations because the pipe motion is geometrically constrained to prevent the broken ends of the pipe from moving a sufficient distance to allow the full double-ended flow area to open. The reviewer thus concludes that Westinghouse has considered an appropriate spectrum of break sizes and locations, comparable to that of the approved FTI/B&W topical report.

Calculation of Blowdown Loads

Westinghouse used the MULTIFLEX 3.0 computer code to calculate the blowdown loads on the reactor vessel and the reactor vessel internals, including the guide tubes and core barrel. MULTIFLEX has previously been used by Westinghouse to calculate blowdown loads. The version of MULTIFLEX used for this analysis was an improved version that was developed specifically for the WOG Baffle-Barrel-Bolt Program (BBBP). According to Westinghouse, previous BBBP analyses performed using this version of the code were accepted by NRC. A conservative, and previously accepted, 1 ms break opening time was assumed. Therefore these calculations of LOCA blowdown loads are judged to be conservative. Also, the blowdown loadings have been determined using a methodology which has been previously accepted by NRC.

Guide Tube Evaluation

At this point, the guide tubes are handled differently from the fuel assemblies. For the guide tubes, Westinghouse has experimentally established allowable loads. When the combined loads

remain below these experimentally determined levels, control assembly insertion will not be impaired. Westinghouse determined the total LOCA loads by combining the inertial acceleration and acoustic loads calculated by MULTIFLEX with the hydraulic cross flow loads, i.e. drag loads, which were estimated based upon scale model tests and plant strain measurements, together with information from the MULTIFLEX and other hydraulic calculations. A dynamic load factor was applied to account for the transient nature of the drag loading. This total LOCA load was added using the square root sum of squares (SRSS) method to the peak safe shutdown earthquake (seismic) load to obtain the total load. The reviewer judges that the methodology used by Westinghouse to calculate the combined peak guide tube loading is conservative and consistent with industry practice.

Westinghouse compared the calculated combined peak loads to the allowable values. Due to the differences in fuel assemblies between Unit 1 (15 x 15) and Unit 2 (17 x 17), the allowable loads and the peak combined loads differ between the units. For both units, the calculated peak combined load showed considerable margin to the allowable. Unit 2 showed a greater than 100% margin for all 5 breaks. Unit 1 has a minimum margin of 24% for the double-ended guillotine break at the main coolant pump outlet nozzle. Therefore, the maximum guide tube deflection which occurs under the combined LOCA and seismic loading will not prevent the control rods from inserting.

Fuel Assembly Evaluation

Fuel assembly deflection and fuel assembly grid loading was then determined using a multi-step process. First, Westinghouse calculated the core barrel, baffle and plate displacements due to LOCA and seismic loadings using the WECAN computer code. Displacements of the fuel assemblies and grid impact forces are then calculated separately for LOCA and seismic loadings using the Westinghouse Commercial Nuclear Fuels Division (CNFD) methodology. Figure 8.2 of Reference 1 shows a schematic of a typical CNFD model. The model used for D.C. Cook has four individual fuel assembly array models. The effects of seismic and LOCA induced motions are then combined using the SRSS method to obtain maximum grid impact forces. Because of the different fuel designs in Unit 1 and Unit 2, separate calculations were performed for each unit. Westinghouse has experimentally established allowable grid loads at the 95 percent confidence level. When peak grid loads are below the allowable then RCCA control rod insertion will not be impaired. The methodology used is judged to be representative of the best current industry practices for determining fuel assembly response to LOCA and seismic loads.

Results of the comparison of calculated peak combined loads to allowable values shows that there is greater than 50% safety margin. The minimum margin occurred for the double-ended guillotine break at the main coolant pump outlet nozzle for the Unit 2 intermediate flow mixer (IFM) grid. All other cases had greater than 100% margin. These results demonstrate that for a design basis or LBB cold leg break, control rod insertion will not be impaired.

Comparison to Babcock & Wilcox / Framatome Technologies Methodology

The Babcock & Wilcox / Framatome Technologies (B&W/FTI) LBLOCA Methodology for recirculating steam generators, described in Reference 3, clearly takes credit for rod insertion following all LBLOCAs. Section 4.3.2.4 of the Topical Report states that "For the reflood phase, the void moderator coefficient will become ineffective and both safety rod injection and borated water injection are used to maintain the reactor subcritical." B&W/FTI received NRC approval for their LOCA evaluation model for recirculating steam generator plants (see Reference 4). Note 2 to Table 2.1.2-1 of this SER (Reference 4) notes that fission heat during reflood is calculated by assuming that the fission power to total power ratio is constant at the value determined at the end of blowdown. Therefore, a kinetics calculation is not performed so no calculated return to power is possible.

B&W/FTI's calculation of LOCA loads is given in Reference 5, which addresses only B&W designed plants with once through steam generators. The B&W designed plants also include "vent valves" which are check valves between the upper plenum and the downcomer. Their purpose is to allow steam to vent from above the core to a cold leg break without having to flow through the loop, thus alleviating steam binding. Incidentally, they may also reduce the forces on the core barrel following a cold leg break by relieving differential pressure loadings. However, this is not their primary purpose. Crediting these valves during a blowdown loading calculation is questionable since the valve opening time is of the same order as the loading transient. B&W's methodology credits scram insertion even for Westinghouse plants without vent valves. Therefore, they obviously do not play an important role of the B&W approach.

Westinghouse has demonstrated that cold log LOCA loadings are not sufficient to prevent control rod insertion, so the presence of vent valves is clearly not needed to conclude that control rod insertability is acceptable in this case.

The B&W methodology of Reference 5, i.e. calculation of blowdown loads with subsequent comparison to allowable loads is essentially the same approach currently being used by Westinghouse. The purpose of the B&W report was to document the calculation of asymmetric LOCA loadings on the components and equipment in the reactor vessel compartment in response to an NRC request. Loads on fuel assemblies and reactor internals were included. Calculated peak LOCA loads are compared to allowable loads for the guide tube assemblies and spacer grids (Section 10.6) and shown to have significant safety margins.

Comparison of the B&W Fuel Bundle with Westinghouse

The B&W 17x17 fuel assembly is shown in Figs. 1 and 2 while the Westinghouse 17x17 fuel bundle is presented in Figs. 3 and 4. As demonstrated in the figures, both bundle designs are identical. Both bundle designs are 12 feet in length, contain eight spacer grids, and utilize a 24 finger control rod cluster. As such, the mechanical behavior of both bundles is expected to be similar so that the Westinghouse load evaluation can be concluded to also reflect the expected behavior of the B&W bundle design.

Given that References 3 and 4 document an NRC approved LBLOCA methodology for recirculating steam generator, i.e. Westinghouse and CE designs, plants which credits rod insertion following all LBLOCAs, there is no relaxation of safety margin in crediting rod insertion following cold leg LBLOCAs as requested for D. C. Cook.

Conclusions

SCIENTECH has performed an independent third-party review of the Westinghouse Topical Reports, WCAP-15245 and WCAP-15246. Westinghouse used a version of the MULTIFLEX 3.0 computer code previously accepted by NRC for the WOG BBBP to calculate the blowdown loads on the reactor vessel and the reactor vessel internals. A conservative, and previously accepted, 1 ms break opening time was assumed and seismic loads were combined with the LOCA loads. For the guide tube acceptance criteria, Westinghouse used experimentally established allowable loads. A dynamic load factor was applied to account for the transient nature of the drag loading. The reviewer judges that the methodology used by Westinghouse to calculate the combined peak guide tube loading contains considerable conservatism.

Fuel assembly deflection and fuel assembly grid loading using the WECAN computer code and the CNFD methodology. This methodology used is judged to be representative of the best current industry practices for determining fuel assembly response to LOCA and seismic loads. Results of the Westinghouse analysis clearly and conservatively demonstrate that for a design basis or LBB cold leg break, control rod insertion will not be impaired.

It is also observed that there is a large margin of conservatism in the evaluation of whether recriticality can occur following the switch to hot leg injection following a LBLOCA, due to the following factors:

1. The event itself, i.e. LBLOCA, has a frequency of occurrence which is very small, especially for the design basis accident.

2. LBB should be credited for these LOCA events since it has been well established that the design basis guillotine break is unphysical. However, the Westinghouse analysis demonstrates that control rod insertion will not be impaired, even for the unphysical guillotine cold leg break.

3. The calculation of boron concentration is very conservative. On a best estimate basis, it is judged that the boron concentration will remain sufficiently high to preclude a return to criticality, even if the control rods do not insert. The Attachment addresses the conservatism in the boron concentration.

4. Westinghouse has demonstrated that the combined seismic and LOCA loading forces will not result in deformations which impair control rod insertion following a design basis or LBB cold leg break.

With these multiple layers of safety margin, it is apparent that the risk associated with return to criticality following a switch to hot leg injection is insignificant.

References

1. J. A. Barsic, et. al., "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2", WCAP-15245, May 28, 1999 (proprietary).

2. J. A. Barsic, et. al., "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2", WCAP-15245, May 28, 1999 (non-proprietary).

3. B&W Fuel Company, "RSG Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, Volume 1 – Large Break", BAW-10168-A (non proprietary)

4. Letter from A. C. Thadani, USNRC to J. H. Taylor, B&W Nuclear Technologies, "Acceptance for Referencing of Licensing Topical Report, BAW-10168P, Revision 1 'RSG LOCA – B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants", dated January 22, 1991.

5. Babcock & Wilcox Co., "Effects of Asymmetric LOCA Loadings - Phase II Analysis", BAW-1621, July 1980.

ATTACHMENT

Boron Concentration Analyses

D.C. Cook Nuclear Plants Units 1 and 2

BACKGROUND

Westinghouse has identified the possibility for re-criticality for DC Cook Units 1 and 2 following a large break LOCA for locations only in the cold leg. The potential re-criticality is a result of the melted ice in the containment following a cold leg break that dilutes the concentration of soluble boron in the containment sump. During the recirculation mode of ECC injection, the ECC pumps take suction from the containment sump. At some later time during post-LOCA long term cooling, the ECC high-pressure injection is switched from all cold side injection to simultaneous hot and cold leg injection. At this time, typically ten hours or more after the opening of the break, the injection of the diluted sump water into the core from the hot leg with the break in the cold leg creates the potential for re-criticality. Hot leg breaks are not a concern since the high concentrate boric acid that accumulates in the core is continuously expelled by the cold side injection through the break in a flushing action to supply sufficient borated water to the sump to preclude dilution with the melted ice. For cold leg breaks, because the cold side injection does not provide a flushing action to expel the high concentrate mixture from the core into the containment, the ice coupled with the accumulation of boric acid in the vessel allows the sump liquid boric acid content to become dilute.

It is important to note that the recriticality is a result of the assumption that the dilute sump water, with a concentration of 1500 ppm, does not mix with the high concentrate boron mixture in the core. This no mixing assumption is considered very conservative, as there are natural convection and diffusion processes that will promote the mixing of the dilute mixtures with the high concentrate boric acid mixtures in the core.

Prior to the switch to hot and cold side injection, borated water enters the core, which is cooled during the long term by pool nucleate boiling. With borated water entering the core, the steam in a bubbly flow mixture, rises to the surface of the two-phase region in the upper plenum where the steam disengages from the two-phase mixture. The boron remains behind in the two-phase mixture continually increasing the boron concentration. After many hours, the concentration in the vessel will increase until the solubility limit is reached causing precipitation of the boric acid.

Boron mixing experiments have demonstrated that as the boron builds up in the core, the high concentrations that develop in the core produce concentration gradients that promote diffusion of the born into the lower plenum from the core. The mixing volume in the vessel therefore includes the portion of the upper plenum containing two-phase, the core, and the lower plenum. The natural convection currents that exist in the core during this natural circulation, pool nucleate boiling process will promote the mixing of the highly borated water in the core with the dilute hot side injection mixture.

Experiments addressing boric acid mixing in the vessel further demonstrate that the diffusion and natural convection processes that govern the fluid behavior in the vessel during long term cooling provide the physical mechanisms to promote uniform mixing of the boric acid throughout the upper plenum, core and lower plenum regions. During the switch to simultaneous injection, the diluted hot side injection water from the sump is expected to mix with the high concentrate boric acid mixture in the core, sufficiently to preclude re-criticality following all large cold leg breaks.

A discussion of the analysis of the transient boron concentration in the vessel following a large break LOCA is discussed below.

DISCUSSION

A calculation of the boron concentration in the vessel for the DC Cook nuclear steam supply system was performed to assess performance of the ECCS to control boron acid content in the vessel and preclude re-criticality.

The attached figure presents the boron concentration versus time following a large break LOCA where reactor coolant system pressure remains near atmospheric pressure. The figure shows the boric acid content as a function of time with no flushing flow assumed and indicates that about 30 hours is available for the operators to initiate simultaneous injection. The solubility limit at 14.7 psia indicated in the figure is 27.53 wt%. Since the safety injection system provides sufficient flow to maintain the core covered with two-phase following reflood of the core, the rate at which the boron builds up is a function of the decay heat generation rate. Any additional injection into the cold side is merely spilled through the break. The equation solved to compute the boron concentration under these conditions is (please see the nomenclature section for identification of the symbols):

$$\frac{d}{dt}C = \frac{W_{bo}}{\rho V_{mix}} C_s$$

The mixing volume, V_{mix}, is taken to be the volume in the lower plenum, core, bypass, and upper plenum.

During the long term, the operators will need only a single high pressure pump to cool and control the boric acid build-up in the core. To show the effect of the net core or flushing flow, it was assumed that the operators switch to simultaneous injection at 12 hours into the event. The effect of several flushing flow rates is demonstrated in the attached figure where injection flows of 467, 233, and 10 gpm were assumed. These flushing flow rates maintain the long term boron concentrations at the values of 1.15, 1.73, and 12.9 wt% (i.e. 2001, 3017, and 22,680 ppm), respectively. Flushing flows associated with the 467 gpm injection rate therefore produce boron concentrations above the sump or source concentration at the time simultaneous injection is initiated. However, these analyses show that for a wide range of ECC injection rates, the concentration can be maintained well above the re-criticality condition of 1500ppm.

The equation that is solved to compute the boron concentration in the vessel following the switch to simultaneous injection is given below.

$$\frac{d}{dt}C = C_s\frac{W_{in}}{V_{mix}} - C\frac{W_{flush}}{V_{mix}}$$

The assumption for this analysis are summarized below:

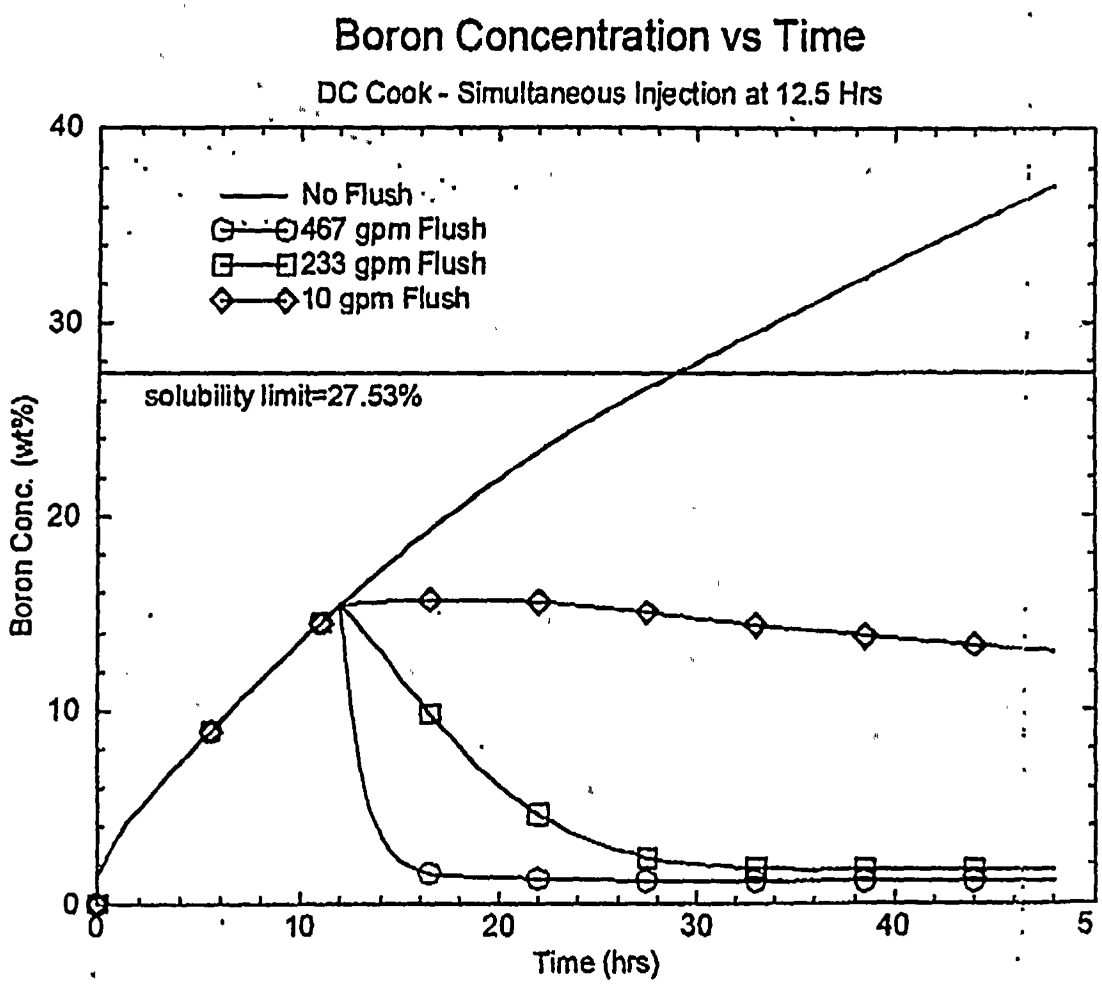
1) The mixing volume consists of the lower plenum, core, bypass, and upper plenum.
2) The RCS is at 14.7 psia (solubility limit = 27.53 wt%).
3) The operators do not initiate simultaneous injection until 12 hours after the break opening. At switch, ½ of the total injection flow is split equally between the hot and cold legs.
4) The operators can reduce the ECC flow by terminating pumps and/or throttling injection.
5) Core power is 1.02% of 3411Mwt.
6) Decay heat is based on the ANS 1971 standard with a multiplier of 1.0.
7) The concentration of the RWST is 2400 ppm.

CONCLUSION

An analysis of the boron concentration in the reactor vessel following a large break LOCA, where the RCS does not refill with ECC injection, demonstrates that the core can be maintained in a cooled condition while controlling boric acid precipitation and precluding re-criticality. The analyses contained herein demonstrate that the boric acid content in the reactor vessel can be maintained at concentrations that are well above re-criticality. Injection flow in the range 10 to 467 gpm can maintain core cooling (i.e. core remains covered with a two-phase mixture) while controlling the boric acid content well below the precipitation limit of 27.53 wt% and above the 1500 ppm value for re-criticality.

NOMENCLATURE

C	= boron concentration
C_s	= source boron concentration
W_{bo}	= core boil-off rate or injection rate prior to flush
W_{flush}	= core flushing flow rate
W_{in}	= injection flow rate in hot or cold side
V_{mix}	= vessel mixing volume
ρ	= liquid density
t	= time


Boron Concentration vs Time
DC Cook - Simultaneous Injection at 12.5 Hrs
No Flush
467 gpm Flush
233 gpm Flush
10 gpm Flush
solubility limit=27.53%
Boron Conc. (wt%)
Time (hrs)
0
10
20
30
40
5

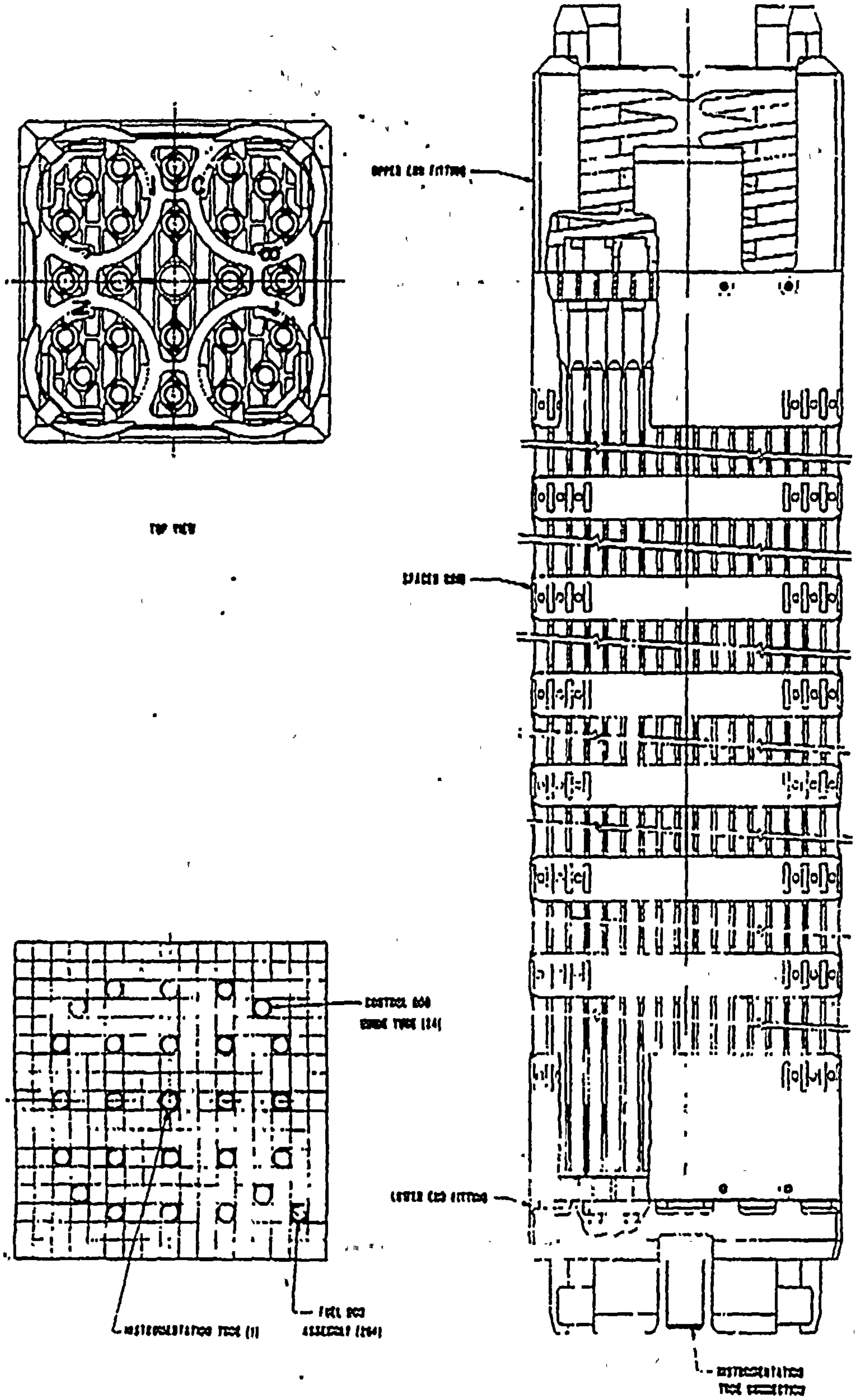


Figure 1 B & W 17 X 17 (Mark C) Fuel Assembly

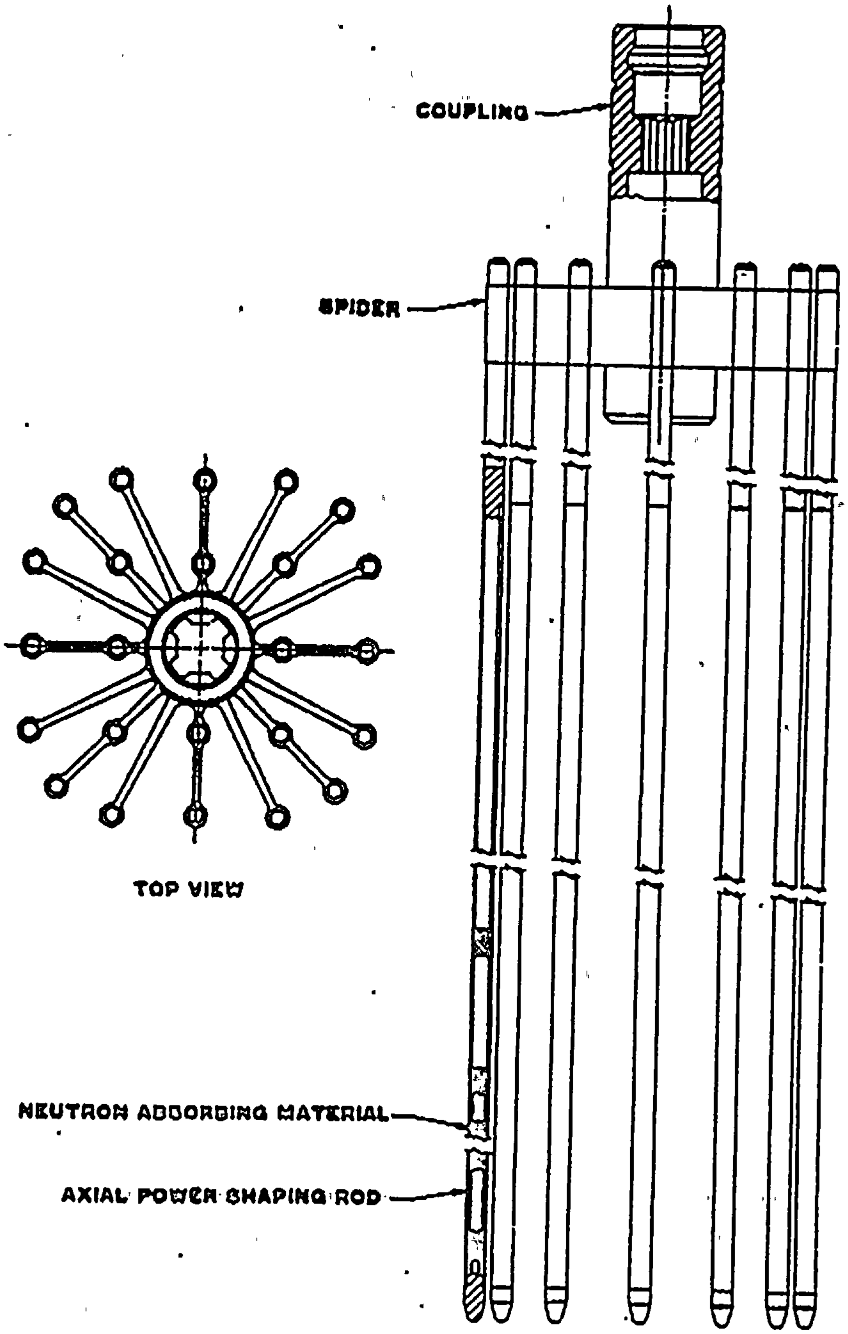


Figure 2 B & W 24 Element CRA

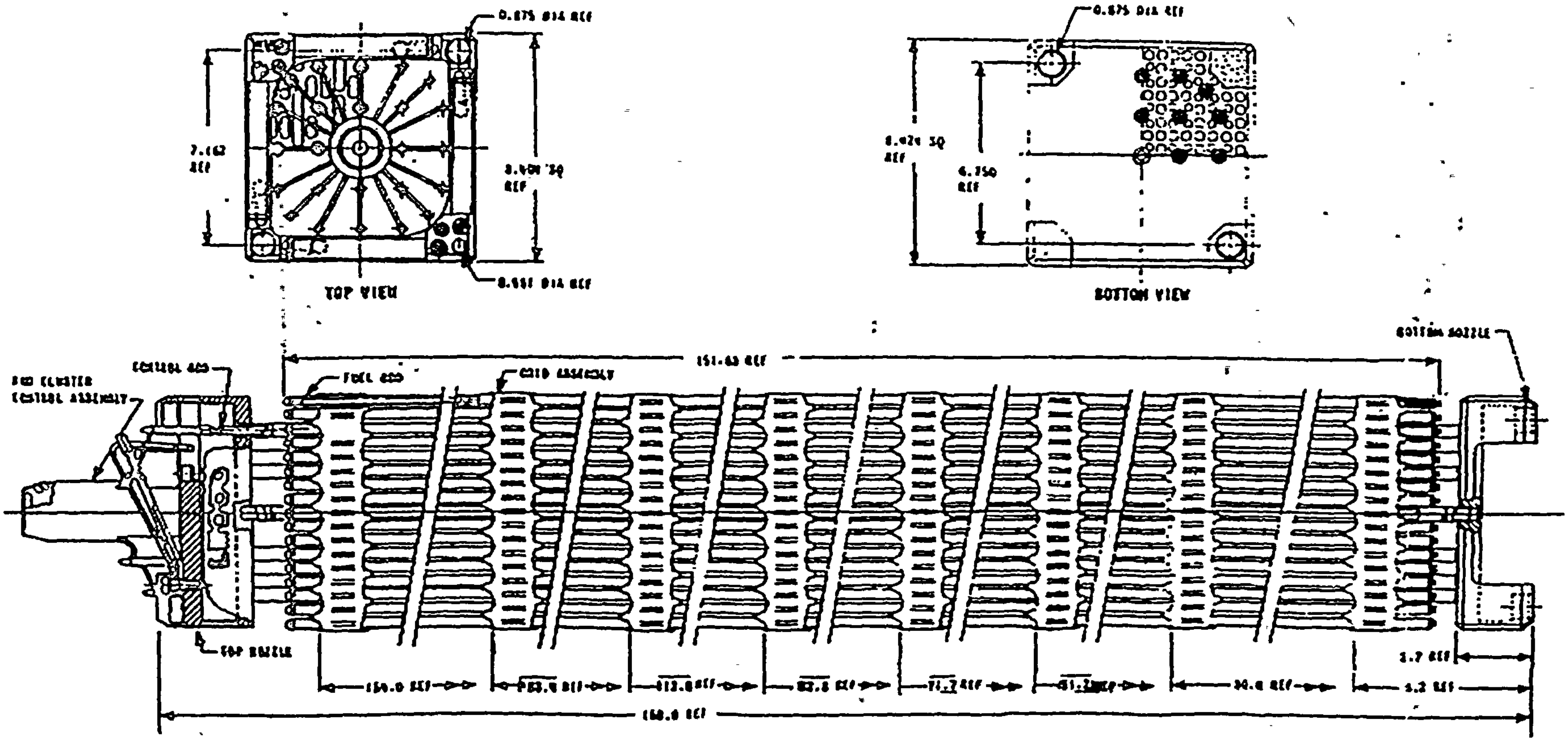


Figure 3 Westinghouse 17 X 17 Fuel Assembly Longitudinal View

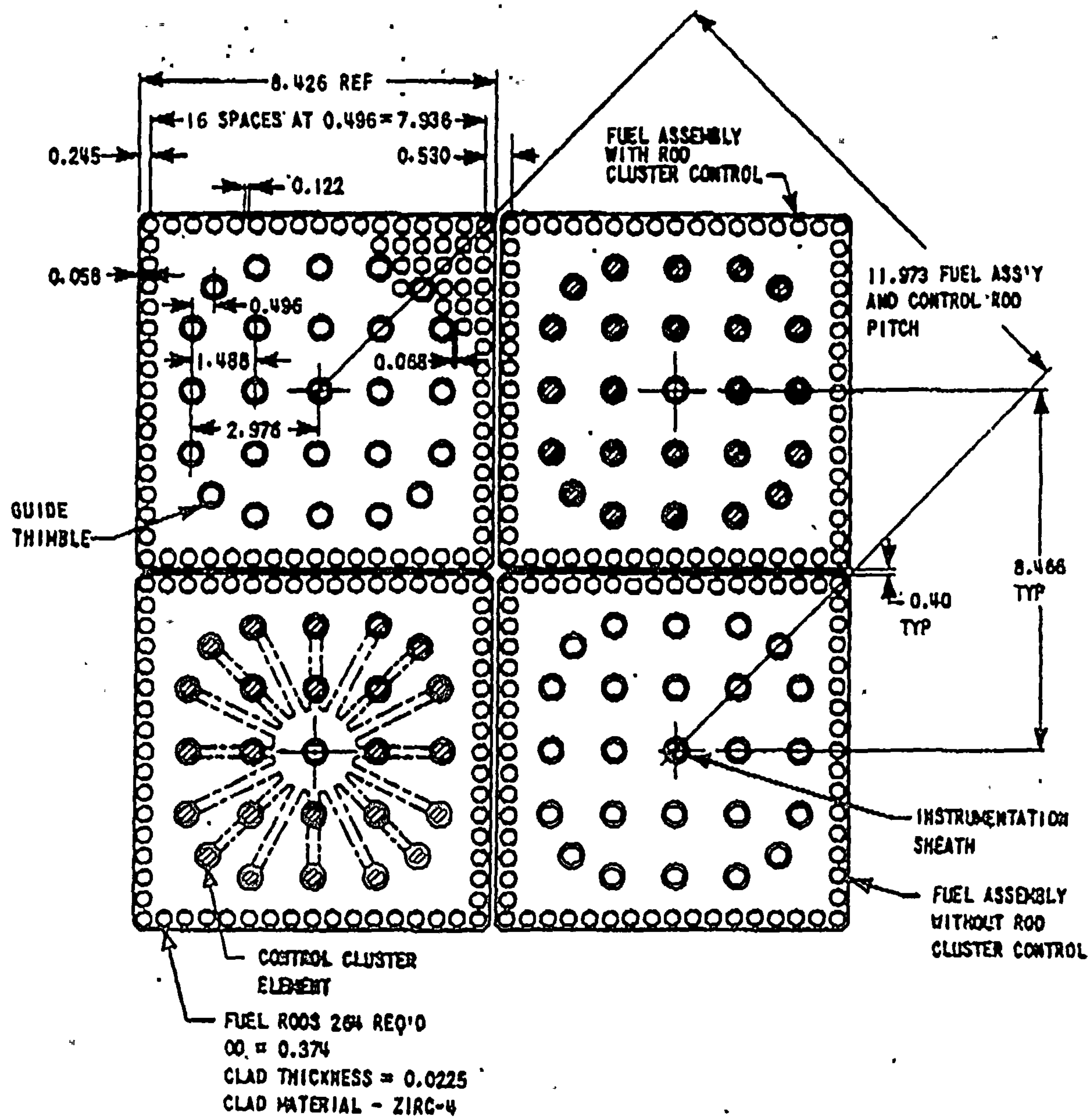


Figure 4 Westinghouse 17 X 17 Fuel Assembly, Cross-Section View

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