CRITERIA AND DESIGN OF PRESSURIZED WATER REACTOR COOLANT SYSTEM SUPPORT STRUCTURES – STATE OF THE ART

J. D. STEVENSON,

Departement of Civil Engineering,
University of Pittsburgh, Pittsburgh, Pennsylvania, U.S.A.

ABSTRACT

A survey of twenty-three U.S. reactor plant systems licensed for construction between 1 January 1967 and 30 June 1970 has been conducted to determine the current state of the art and to describe the resultant structural engineering practice used to design Reactor Coolant System Support Structures. Areas discussed explicitly are the structural design criteria, analytical assumptions, methods of analysis, resultant designs and quality control and fabrication procedures.

Introduction:

The design of nuclear power plant facilities has presented the structural designer with two new and relatively unique major design efforts. The first of these two efforts, that of containment design has received considerable attention in the past few years in the technical literature[1,2,3,4,5] and by the standards and code committees of the American Concrete Institute[6] and the American Society of Mechanical Engineers.[7]

The second of the two major structural design efforts, that of major nuclear component support structure design and analysis, has received scant attention in the technical literature and only relatively recent attention by the code committees of the American Society of Mechanical Engineers.[8]

This paper is an attempt to present a comprehensive review of the current "state of the art" in Reactor Coolant System, (RCS) component support design, criteria and analysis.

The discussion is restricted in detail to Pressurized Water Reactors (PWR) supports systems designed by U.S. Architect-Engineers and Manufacturers due to scope limitations imposed by the author. However, since the same design agents are generally involved in support design of Boiling Water Reactors (BWR) systems it should be expected that general conclusions presented in this paper are equally applicable to BWR Systems.

The material presented herein is based on a detailed study of the criteria, analytical assumptions design features and fabrication, erection and quality control procedures used in the design of the twenty-three individually designed PWR support systems which have received U.S. Atomic Energy Commission (USAEC) construction permits in the 3.5 year period between 1 January 1967 and 1 July 1970. This material is based on the data presented in the Preliminary and Final Safety Analyses Reports, (PSAR), (FSAR) filed with the USAEC plus interviews with a majority of the Architect-Engineers and Manufacturers who have acted as design agents.

There are in the United States three Nuclear Steam Supply manufacturers of PWR systems, Babcox & Wilcox, Combustion Engineering and Westinghouse Electric Corporation.

The PWR reactor coolant system is composed of four major components. The heart of the system is the nuclear reactor which operates as a thermal energy source using pressurized water as a coolant at approximately 2250 psi pressure and 600^{o}F temperature. The pressurized water is circulated through a primary closed loop from the reactor to a steam generator and returned by means of a reactor coolant pump. These pumps are designed to move water at approximately 90,000 gallons per minute. A pressurizer is employed to maintain proper pressure within the reactor coolant system. Steam from the secondary side of the steam generators flows through steam pipes to provide motive force for the steam turbine electric generators.

During the period surveyed one common plant size, approximately 800 MWe was available from all three manufacturers. During the period studied, Westinghouse and Combustion Engineering had in addition a nominal 500 MWe and Westinghouse a 1100 MWe system available. Since this period, Babcox & Wilcox and Combustion Engineering have also provided 1100 MWe systems. Typically, Babcox and Wilcox and Combustion Engineering provide a two-loop system with two pumps per loop for either the 800 MWe or 1100 MWe size. Westinghouse provides two loops for the 500 MWe, three loops for the 800 MWe and four loops for the 1100 MWe systems with one pump per loop. Both Combustion Engineering and Westinghouse employ vertical U-tube steam generators while Babcox & Wilcox employs a once-through vertical steam generator.

In Table 1 can be found the salient features of the three systems affecting component support design and in Figures 1, 2 and 3 can be found typical layouts of the three systems.

The responsibility for the design and analysis of the reactor coolant system is normally shared by the manufacturer and the electric utility or its Architect-Engineer, A/E design agent, The design and analysis scope split varies between two extremes. At one end of the spectrum the manufacturer performs all engineering design and analysis of the reactor coolant components and steel support structure with the utility or design agent input limited to definition of seismic motion of the supporting building concrete structure. At the other extreme the utility or its A/E performs all engineering design and analysis of support structures with the manufacturer input confined to the definition of the system design loads and the evaluation of resultant forces and moments on the individual components. In most cases no formal Design Specification for support structures similar in content to that required by ASME Section III paragraph N141 for nuclear vessels have been prepared. The design criteria used for most projects are typically those defined in the PSAR as finally agreed to by the AEC and supplemented by topical reports[9,10] submitted to the AEC by the various Manufacturer's and A/E's which define their standard criteria in more detail.

Analytical Assumptions:

Typically, support structures are designed for two kinds of loading, service loads and extreme loads. Service loads are defined as those loads which are expected during normal operation or anticipated transients. For these load cases the system is expected to be capable of continued operation or normal shutdown subsequent startup. Such loads are defined in the present American Society of Mechanical Engineers, ASME Section III code as Normal and Upset Load Conditions. Usually considered in this category are dead and, operating live loads such as fluid weight and pressure and steady-state temperature loads. Also included are transient temperature and pressure effects due to anticipated normal and abnormal operation and start-up and shutdown. The Operational Basis Earthquake, OBE load is also included in this category.

Extreme loads characterize those load conditions for which safe shutdown of the system is possible. In general there is no requirement for continued operation nor that the system will necessarily be brought back to operational status. Such loads are defined in the ASME Section III Code as Emergency and Faulted Load Conditions. Usually considered in this category are the applicable service loads combined with the Design Basis Earthquake, (DBE) loads or the complete rupture of a reactor coolant pipe and in some instances steam/feedwater pipe break loads which are defined as a Design Basis Accident, (DBA) loads or the postulated simultaneous occurrence of DBA and DBE loads.

The dynamic nature of the seismic and design basis accident loads are considered in design. The response spectrum technique together with modal analysis are normally used to determine seismic loads. Therefore, while seismic loads or stresses are determined by dynamic analysis, the resultant effect on the structure is normally determined by a "best estimate" square root sum of squares combination of modal maxima. As a result the time relationship of the seismic loads are lost and seismic effects assume a single value load or stress relationship when combined with other loads which is independent of time.

The dynamic nature of the DBA loads are treated in a variety of ways. The techniques range from a complete time history multi-degree of freedom elastic analysis where support loads are determined as a function of time to that of an equivalent static load determination with Dynamic Load Factors, (DLF) evaluated assuming elastic or implicit elasto-plastic response. Another technique employed uses an upper bound equivalent static load DLF equal to 2.0 but permits a 25 percent increase in yield strength of the material due to the rapid loading rate.

The total load for the combination of DBE and DBA effects is determined by direct summation. It is, however, the author's opinion that in consideration of the true dynamic nature of both the seismic and blowdown loads the direct combination of simultaneous maxima of DBE and DBA is a highly improbable event.

While orthogonality (or independence of seismic and blowdown loads cannot be demonstrated) analystically, the absolute sum of the two loads appears overly conservative.

An integrated structural system composed of the reactor coolant components, their steel support structures and the connecting pipe are normally analyzed for extreme as well as service loads. Typically, analysis is performed on a per loop basis but the entire RCS is analyzed when dictated by lack of symmetry requirements. In most cases the concrete structures which form the foundation of the component steel supports are considered rigid.

Seismic analysis is usually performed for two separate horizontal directions of earthquake with each considered simultaneously with a vertical component of load. Seismic analysis normally is performed using a modal analysis together with the ground response spectrum defined for the site in those cases where the building structure including the building foundation is included in the mathematical model. In other instances floor response spectrum developed for the level of the reactor coolant system supports in the building structure are used as input to the seismic analysis. In this way the detailed mathematical model of the RCS is uncoupled from that of the building support structure.

There is little uniformity in the techniques used for accident analysis. In some cases a complete time history dynamic-elastic analysis is performed. In other cases equivalent static loads are used assuming elastic or elastic-plastic response of the structural system. Consideration of the loads during the blowdown transient vary from application of time varying loads at points of discontinuity and changes in direction to the use of a single equivalent static load at the assumed point of break. Usually, equipment support design is performed in two phases. The supports are sized initially using equivalent static loads recommended by the manufacturer acting on the supports as isolated free bodies. Seismic loads usually account for less than 25 percent of the total load on the supports except for high seismic sites, hence in the preliminary design of supports seismic loads are often treated as a small percentage increase to equivalent static accident loads. Once the supports have been sized so that their stiffness characteristics can be defined, then an integrated analysis of the RCS and its supports can be performed to determine resultant forces in the equipment, piping and supports. If the analysis indicates criteria design limits of the various components are exceeded, the support designs are modified and re-analyzed until criteria limits are met. Behavior criteria limits for the reactor coolant system used by the three U.S. manufacturers are usually defined in terms of the vessels, pipes and supports. In general these criteria fall into the Normal, Upset, Emergency and Faulted Load categories defined in Section III of the ASME Code.

Except for those cases where the support design is entirely within the manufacturer's scope, the design of equipment supports is shared between the equipment manufacturer and the A/E. All manufacturers have at least one skirt mounted component. These skirts which are integrally attached to the component are the manufacturers design responsibility. Usually, the integrated loop analysis does not consider individual support members in detail since the support for even a single component may consist of more than 100 structural members. Supports are usually represented as a series of linear spring restraints attached to a lumped mass of stiffness node of the component or in the form of a 6 x 6 stiffness matrix defining the restraint of a mass or stiffness node of the component. In this way the detail analysis of the support structures may be separated from the integrated RCS loop analysis. This has a particular advantage since in most cases the loop analysis is performed by the manufacturer and much of the support design by the A/E.

Component Support Design Features:

Reactor Vessel -

The support of the reactor vessel is of two major types, (1) skirt mounted and (2) supported under the nozzles. The Babcox and Wilcox reactor vessels have been typically supported by a skirt attached to the lower vessel head.

The Combustion Engineering and Westinghouse vessels have typically been supported under the nozzles. The number of support points vary from 3 to 6. The actual supports under the nozzle vary from a continuous ring girders supported in some instances on a series of steel columns or by the concrete primary shield wall to individual pad supports under the nozzles. The close proximity of the reactor vessel nozzles to the primary shield wall usually requires forced cooling of the support structure. Both air and water cooling have been used.

In addition to the methods just described, neutron shield tanks have been used which have the effect of a skirt support under the nozzles. Lateral as well as vertical support are provided by the support systems just described.

Steam Generators -

Steam generators present the biggest challenge to the designer which develops the largest variety of support schemes. This is a result of the usual requirement which necessitates a break in the primary reactor coolant system does not cause a consequencial break in the secondary steam system and vice-versa. The major categories of support schemes used are listed as follows:

1) Skirt supports integrally attached to the lower steam generator channel head.
2) A steel truss composed of rolled wide flange structural shapes supporting both the lower and upper steam generator shell.
3) A steel frame composed of rolled and built up wide flange structural shapes.

4) Vertical support by means of hinged steel pipe columns and lateral support by means of rolled and built up wide flange girders.

The upper steam generator shell is also typically supported near the center gravity by a variety of tie bars, snubbers or bumper systems.

Reactor Coolant Pump -

The support systems of reactor coolant pumps include spring hangers, pinned columns, frames and skirts. Lateral restraint for the DBA load case is usually provided by a system of tie bars or bumper limit stops.

Pressurizer -

The pressurizer is normally supported on an integral skirt or by a bearing ring attached to lugs integrally attached to the pressurizer shell. The loads imposed on the pressurizer supports are typically much smaller than those on the rest of the RCS components since the pressurizer surge line which connects the pressurizer to the remainder of the RCS has a crossectional area 20 to 50 times less than the main RCS piping. In most instances the foundation of the pressurizer is the concrete building structure but structural steel frame and truss towers have also been used.

Reactor Coolant Pipe -

Reactor coolant pipe restraints are sometimes provided which serve two purposes 1) provide points of intermediate support of the reactor pipe and thereby reduce the loads on the equipment supports and 2) restrain or restrict the potential of reactor coolant pipe whip in the event of rupture.

Fabrication - Quality Control Procedures -

When structural steel elements are designed for service loads, factors of safety in the form of allowable working stresses or load factors are typically in the range of 1.5 to 2.0.

The magnitude of extreme loads and their very low probability of occurrence dictate for rational design that the safety factors must be reduced. In many cases these factors approach 1.0.

Therefore, it is essential that structures designed for extreme load must behave as designed. In particular the materials and fabrications making up the structures must perform as assumed in design. In general this requires a level of quality-control and assurance more stringent than that employed in conventional steel structural design. These additional factors not normally considered in conventional support designs are summarized as follows:

1. Close control of specified materials to include only American Society of Testing Material (ASTM) designated materials with well-defined minimum structural behavior requirements.

2. Traceability of such material by heat and in some cases user tests of such materials to insure mill tests requirements have been met.

3. The demonstrated ability of the material to absorb energy and remain ductile under tensile or impact load, Nil Ductility Transition Temperature (NDTT) properties.

4. The reduction or elimination of residual stresses by means of stress relief.

5. Testing of welds by other than visual inspection to include radiography, magnetic particle and dye penetrant methods where appropriate to the critical nature and type of weld.

6. Ultrasonic testing of plate material to disclose lamination where such laminations would be detrimental to the structural capacity or integrity of the member.

7. Derating or reduction of the structural capacity of the material to carry load in a particular direction when subjected locally to biaxial or triaxial stress states. (Use of stress intensity instead of uniaxial stress limits).

8. Derating or reduction of the structural capacity of the material due to non-isotropic properties in directions other than in the direction of rolling.

All of the concerns listed have to some extent been considered in Section III and VIII of the ASME Code in the design of pressure vessels. To varying degrees the criteria developed in the ASME Code for pressure vessels has been applied to support structures by the design agents. There is, however, no consistent pattern in the manner these additional requirements have been specified.

The erection and setting of reactor coolant system components and their supports is a significant task. Costs associated with field erection of supports and placements of components can run as much as 20 to 30 percent of the total cost of the support system. It is the author's opinion some of these relatively high erection costs are a result of lack of appreciation on the part of support designers of the magnitude of the fabrication tolerances involved in the fabrication of the large reactor coolant system components and the supporting concrete structures. Such tolerances are often measured in inches and fall outside the adjustment range typically made available by encasing anchor bolts in sleeves or by slotting of oversized bolt holes. Provisions for detailed positive adjustment of the support systems should be made an explicit part of the support design.

Conclusions:

One obvious hope is that the great variety of support systems presently in use will give way to reduced number of relatively more optimal designs. This goal, however, will be difficult to achieve until the basic criteria governing such design is standardized. The development of the new ASME Section III-NF is one major step in that direction. Historically in the U.S. the criteria used for the design of supports has been somewhat of a moving target. The design criteria is normally fixed at the time of issuance of a construction permit but then is subject to review approximately 3 years later at the time of USAEC issuance of an operating permit.

Any modifications to typical criteria or changes in the "state of the art" that have been insued in that 3-year period often become the subject of additional analysis or re-evaluation of structural design adequacy. While one cannot fault the technical advisability of such a procedure, the designer as a result while designing to one criteria must always anticipate the potential changes to that criteria which may occur over the next three-year period. In retrospect this tends to cast the designer into the role of always being either over or under conservative in his design depending on his ability to project future requirements. Either way he tends to incur the criticism of his client. The development of optimized designs while an admirable goal will be very difficult to achieve until criteria is allowed to become truly standardized and not part of a stocastic process.

A detailed survey of the present "state of the art" of PWR Reactor Coolant Support Design has illustrated an appreciation on the part of the designer of the critical nature of these structures and is reflected in the extreme loads considered in design as well as the extraordinary attention paid to improved fabrication and quality control requirements over that required to support conventional components.

TABLE 1 - PWR REACTOR COOLANT SYSTEM COMPONENT NOMINAL DESIGN PARAMETERS

Manufacturer	Nominal Size MWe	Reactor Coolant System Pressure	Reactor Coolant System Temp. °F	R.V. (Kips) Design Weight	Steam Generator (Kips) Dry Wt.	Steam Generator (Kips) Oper. Wt.	Steam Generator (Kips) Flooded Wt.
Babcox & Wilcox(1)	847	2500	650	1700	1145	1275	1480
Combustion(2) Engineering	800	2500	650	2000	1004	1218	1527
Westinghouse(3)	780	2485	650	1250	660	800	1095

	Coolant Pump (Kips)			Pressurizer (Kips)			Reactor Coolant Pipe Legs			
	Dry Wt.	Oper. Wt.	Flooded Wt.	Dry Wt.	Oper. Wt.	Flooded Wt.	Hot Dia./In.	Crossover Dia./In.	Cold Dia./In.	Surge Dia./In.
Babcox & Wilcox	--	190	--	305	340	400	36	28	28	10
Combustion Engineering	141	--	148	206	--	300	42	30	30	12
Westinghouse	184	188	--	175	204	256	29	31	27.5	14

(1) Oconee Unit #1 - Duke Power Company

(2) Calvert Cliffs - Baltimore Gas & Electric Company

(3) Surry - Virginia Electric Power Company

NOTE: Weights shown are approximate.

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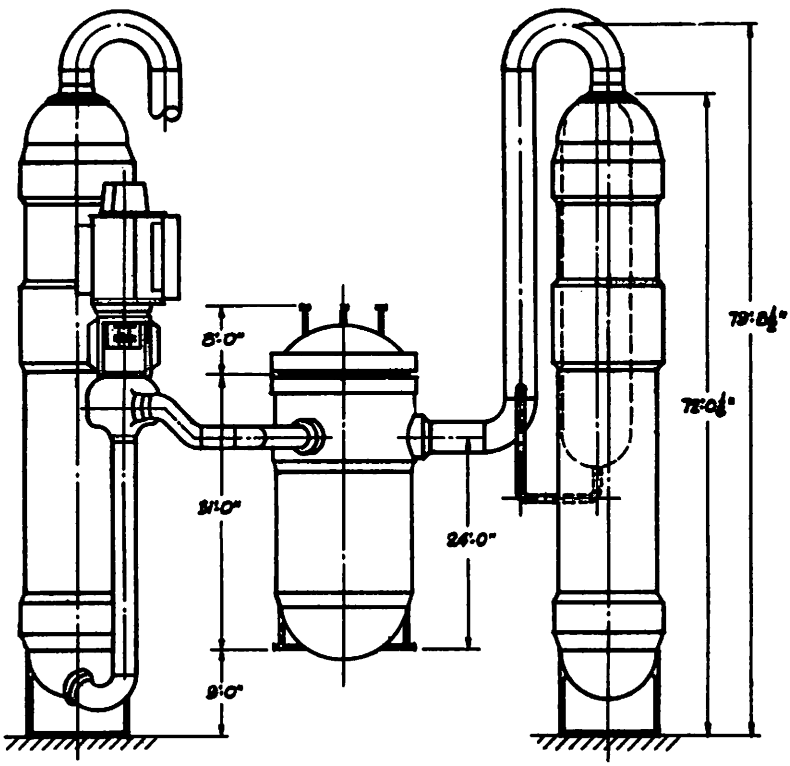

FIGURE 1. TYPICAL LAYOUT OF BABCOCK - WILCOX 2 LOOP REACTOR SYSTEM

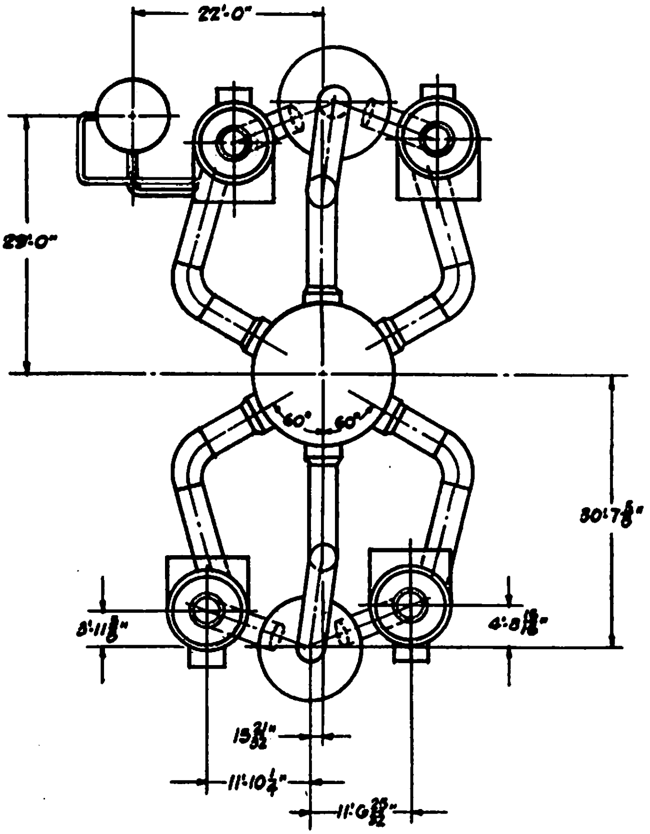

FIGURE 1, CONT. TYPICAL LAYOUT OF BABCOX - WILCOX 2 LOOP REACTOR SYSTEM.

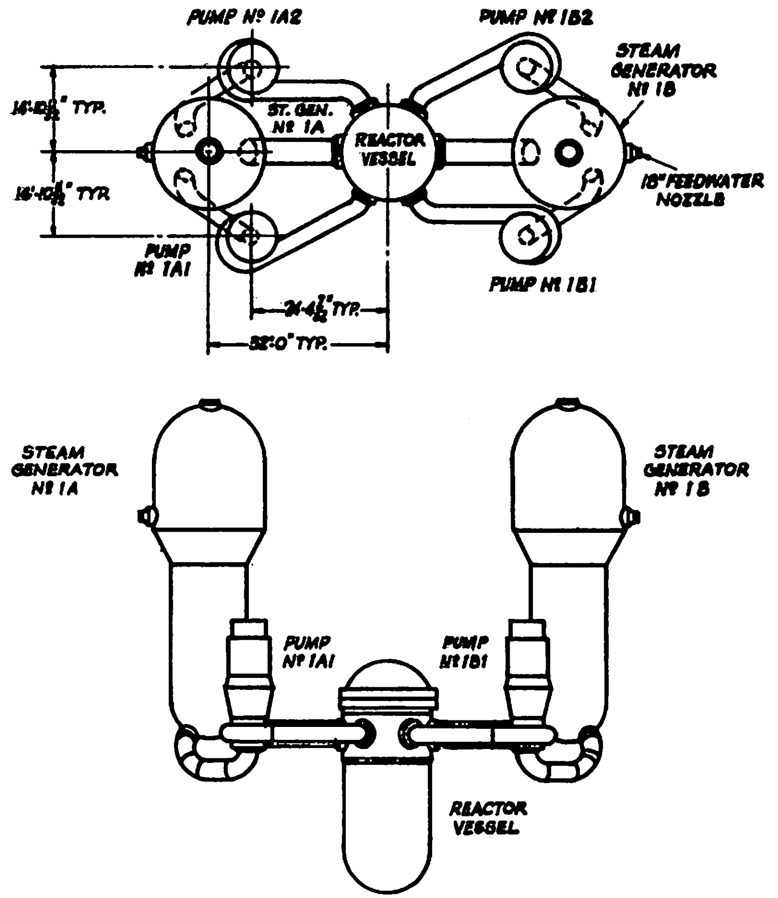

FIGURE 2. TYPICAL LAYOUT OF COMBUSTION ENGINEERING TWO LOOP SYSTEM.

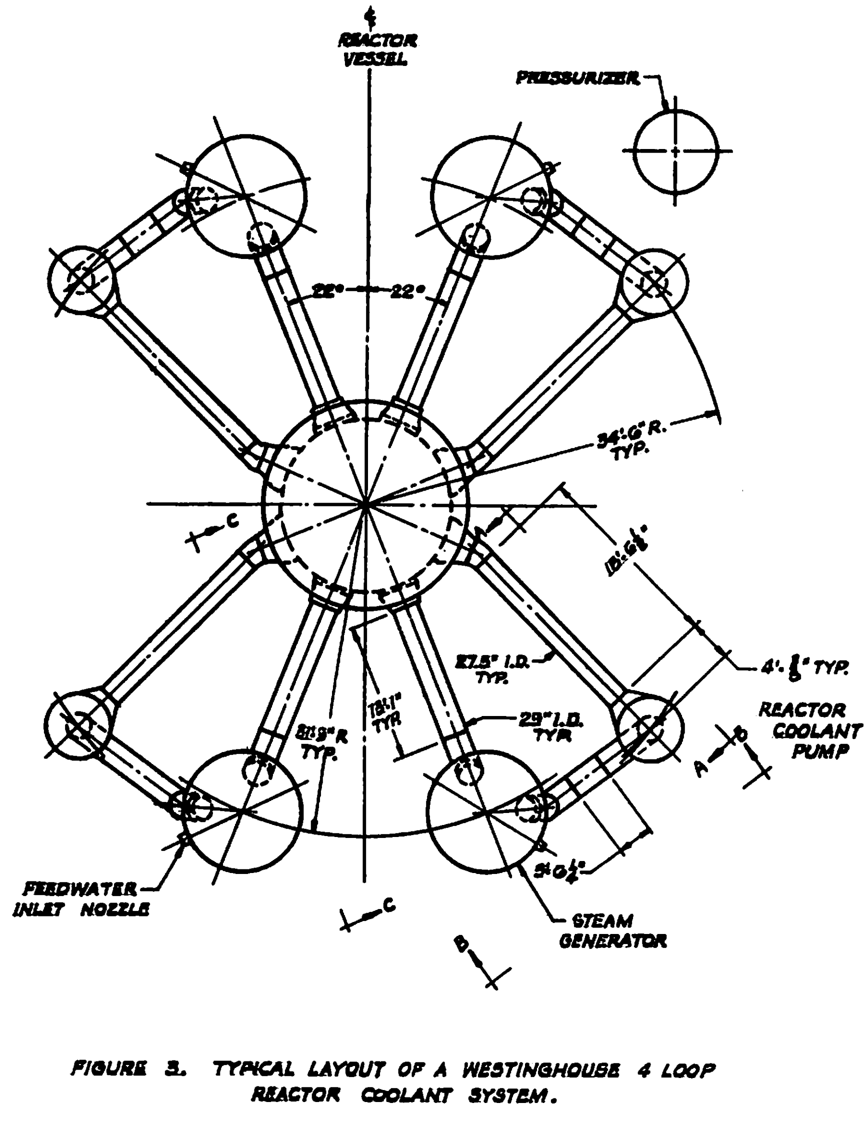

FIGURE 3. TYPICAL LAYOUT OF A WESTINGHOUSE 4 LOOP REACTOR COOLANT SYSTEM.

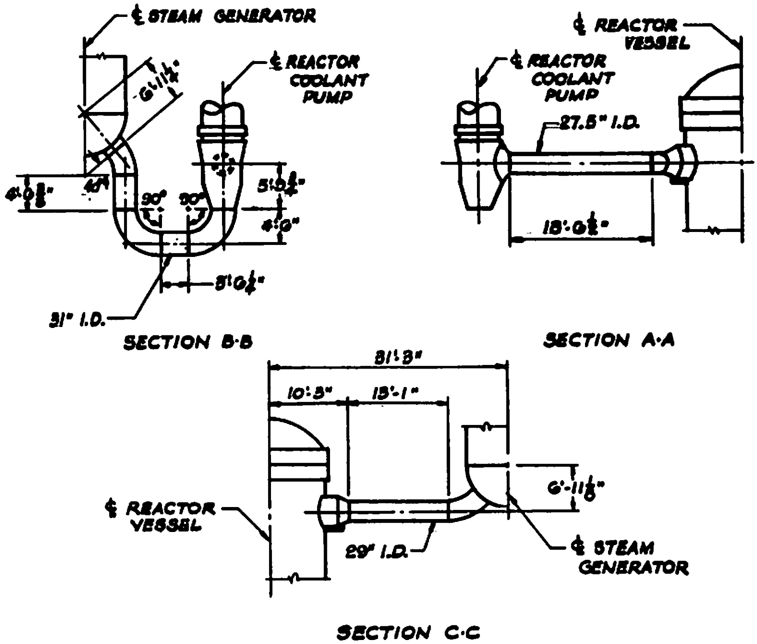

FIGURE 3, CONT. TYPICAL LAYOUT OF A WESTINGHOUSE 4 LOOP REACTOR COOLANT SYSTEM.

DISCUSSION

Q G. RIGAMONTI, U. S. A.

Kind of pipe break (guillotine/longitudinal) and location of the break along the pipe are very important for the supports design. Could you tell which is the policy followed by the different manufacturers ?

A J. D. STEVENSON, U. S. A.

Various manufacturers have different positions with regard to type of pipe break and location of pipe rupture which they have been presenting with various supporting material to the USAEC with the intent of limiting the location and type of breaks for their particular systems. The intent has been to eliminate consideration of slot type breaks and limit breaks only to points of high stress. The USAEC has to date made no decision on the matter.

Q K. KITADE, Japan

I would like to know your comment to the following : Regarding to the design criteria for supports of main coolant piping, I think it is not necessary to combine seismic load with loss of coolant accident, because the period of loss of coolant accident, especially affect to supports, is very short, thus from the view point of probability occurrence of loss of coolant accident and the earthquake at the same time is negligible.

A J. D. STEVENSON, U. S. A.

I agree with your position if it can be said the pipe break event is really independent of the earthquake load. Since the earthquake is specifically considered in design this in my opinion would tend toward independence. However, to date the USAEC has not shared this view.