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STRUCTURAL ANALYSIS OF REACTOR COOLANT LOOP FOR THE SOUTH TEXAS PROJECT UNITS NO. 1 AND 2

VOLUME 1

ANALYSIS OF THE REACTOR COOLANT LOOP PIPING

D. V. Calhoun

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APPROVED: D. F. Miller

D. F. Miller, Manager
Reactor Coolant Loop Analysis

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WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

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SECTION 1
INTRODUCTION

The South Texas Project nuclear power plant reactor coolant loops for Units 1 and 2 are subjected to a very detailed structural and mechanical evaluation to ensure that the public's health and safety are protected. This detailed evaluation compares the results obtained from piping system analyses with the acceptance criteria, ASME Boiler and Pressure Vessel Code Section III Nuclear Power Plant Components[1] (hereafter referred to as the code) for the conditions stated in the South Texas Project Piping Design Specification ASME III Code Class 1 ANS Safety Class 1[2] (hereafter referred to as the design specification). This report concerns itself with the structural evaluation of the reactor coolant loop and primary equipment supports system under all design loading conditions. In particular, volume 1 contains the piping stress evaluation and system analysis description, volume 2 the primary equipment support evaluation, and volume 3 the branch nozzles evaluation and the pipe evaluation in the close vicinity of the nozzles.

Normal operation and safe shutdown of a nuclear plant depends upon the design adequacy and structural integrity of the reactor coolant loop. To demonstrate design adequacy and structural integrity of the reactor coolant loop, analyses are performed for loading under normal conditions, seismic disturbances, and postulated loss of coolant accident conditions. The results of these analyses are compared with the allowable stresses of the code in accordance with the design specification. The results of this information are reported in volume 1 of the stress report.

A nuclear power plant, based on the closed cycle pressurized water reactor (PWR) concept, utilizes two separate fluid systems which interface at a heat exchanger. These systems are known as the primary and secondary systems. Heat is produced in the core by the fission process and is transferred from the primary coolant to the secondary system through a heat exchanger normally referred to as a steam generator. The primary coolant cycle is completed when the water is pumped back to the reactor vessel by the reactor coolant pump (figure 1-1).

In the Westinghouse pressurized water reactor (PWR) system, the primary system is designed for a pressure of 2485 psig and a reactor design temperature of 650°F. The secondary system

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1. ASME Boiler and Pressure Vessel Code Section III "Nuclear Power Plant Components," ASME, New York, 1974, up to Winter 1975 addenda.

2. Westinghouse Equipment Specification 953385 Rev 0, "Piping Design Specification ASME III Code Class 1, ANS Safety Class 1 – for Houston Lighting & Power Company, South Texas Project Units 1 and 2," Balkey, K. R. (Westinghouse Proprietary).

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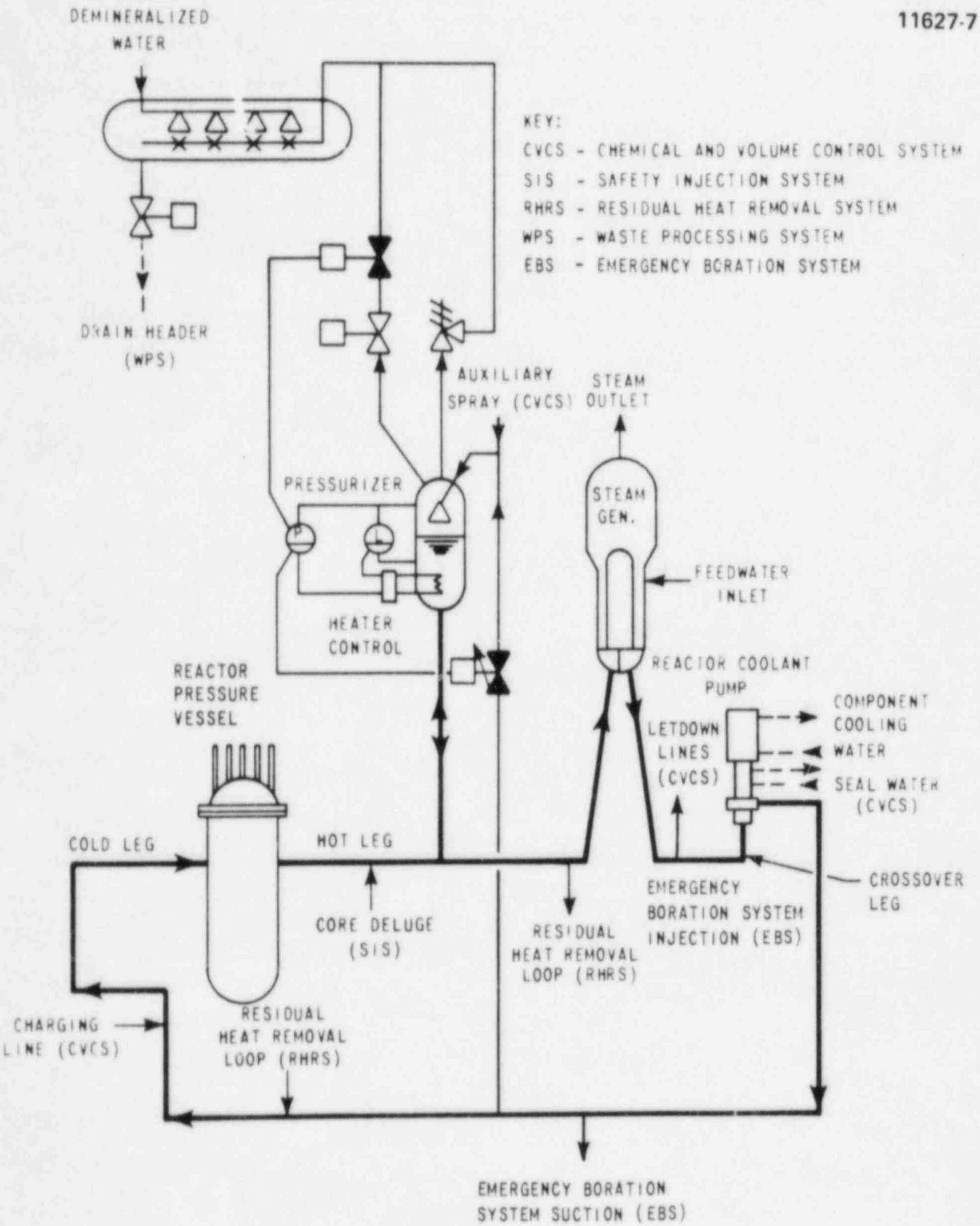

Figure 1-1. Reactor Coolant System, Flow Diagram

operates at a lower temperature and pressure than the primary system to allow the transfer of heat and to promote the formation of high quality saturated steam.

The Westinghouse PWR consists of closed reactor coolant loops as shown in figure 1-2. Each reactor coolant loop contains one coolant pump and one steam generator. The reactor coolant pumps are Westinghouse vertical, single-steam, mixed flow pumps of the shaft seal type, the steam generators are Westinghouse vertical U-tube units. The steam generator consists of two basic parts: an evaporator section, and a moisture separation section. The evaporator section features a U-tube bundle where heat from the reactor is transferred through the tube walls to convert pure secondary-side feedwater into steam. The moisture separation section consists of a set of moisture separators which remove entrained water from the steam.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation, limits pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

Auxiliary system components are provided to charge the reactor coolant system, add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove decay heat when the reactor is shut down, and provide for emergency safety injection.

The chemical and volume control system (CVCS) performs the following functions:

- Fills the reactor coolant system
- Provides a source of high pressure water for pressurizing the reactor coolant system when cold
- Maintains the water level in the pressurizer when the reactor coolant system is hot
- Reduces the concentration of corrosion and fission products in the reactor coolant
- Adjusts the boric acid concentration control
- Provides high pressure seal water for the reactor coolant pump seals

The residual heat removal system (RHRS) transfers heat energy from both the core and the reactor coolant system during plant shutdown and refueling operations. The system is also used (in conjunction with the safety injection system) for emergency core cooling under the postulated pipe rupture accident conditions.

The primary function of the safety injection system (SIS) is to supply borated water to the reactor coolant system to limit fuel rod cladding temperature in the postulated but unlikely

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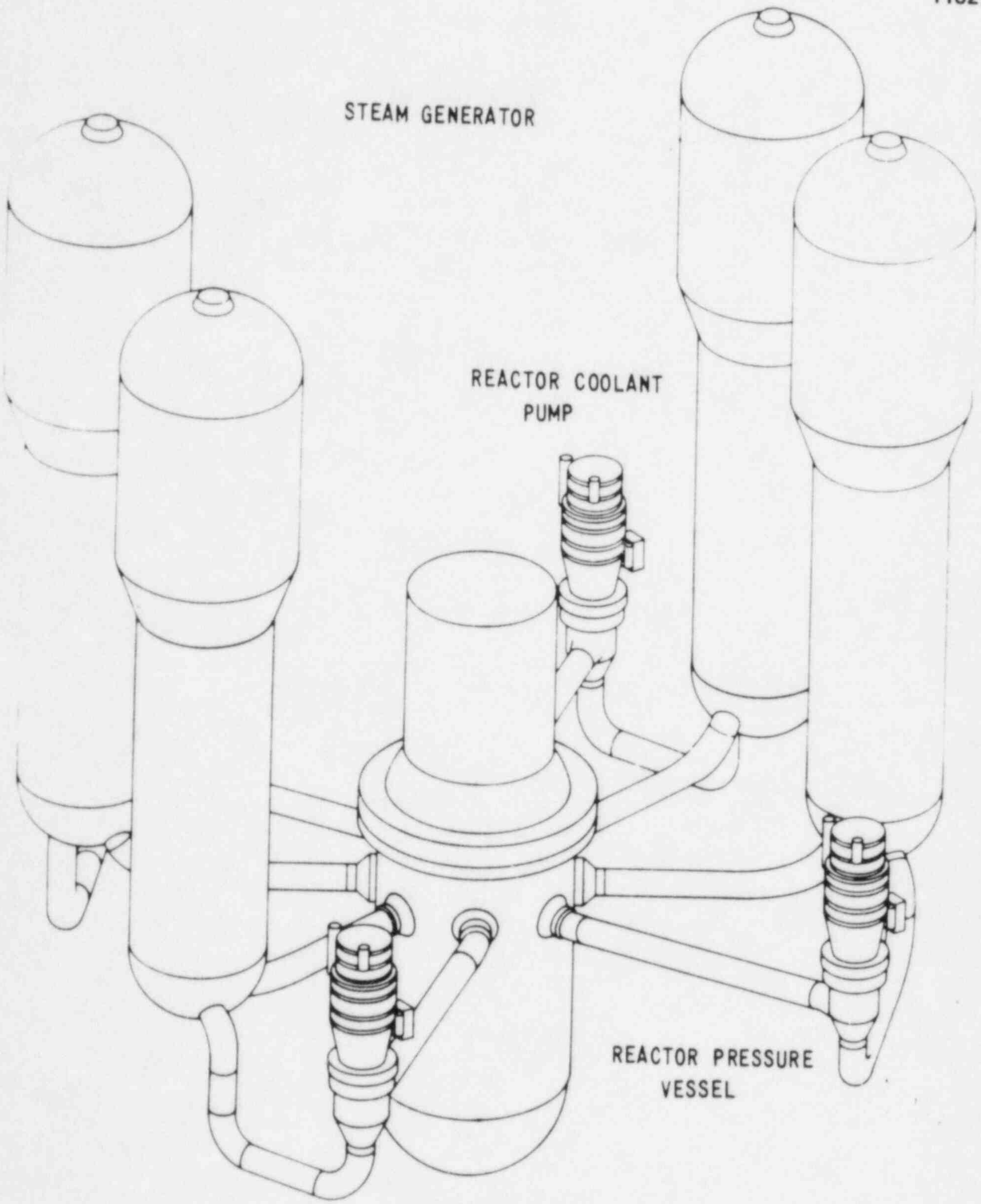

Figure 1-2. Simplified Diagram of the NSSS

event of a loss of coolant accident. A secondary SIS function is to provide a means for introducing a spray of borated water into the containment as an additional dynamic heat sink.

The Emergency Boration System (EBS) protects the Reactor Coolant System against the effects of an incident causing a loss of fluid from the secondary side (steam side), identified as a steam break. The EBS provides negative reactivity in the form of concentrated boric acid to counteract the reactivity increase resulting from the steam break.

The reactor coolant loop and primary equipment supports system are part of the nuclear steam supply system (NSSS). Figure 1-3 shows a simplified plan view of the NSSS.

This report presents the computation of stresses in the reactor coolant loop piping for all loading conditions stipulated in the design specification. In addition, loads on supports for the loading conditions stated above are generated for use in the support evaluation. (See volume 2 of this report.) Finally, this report compares the piping stresses with the acceptance criteria as stated in the code.

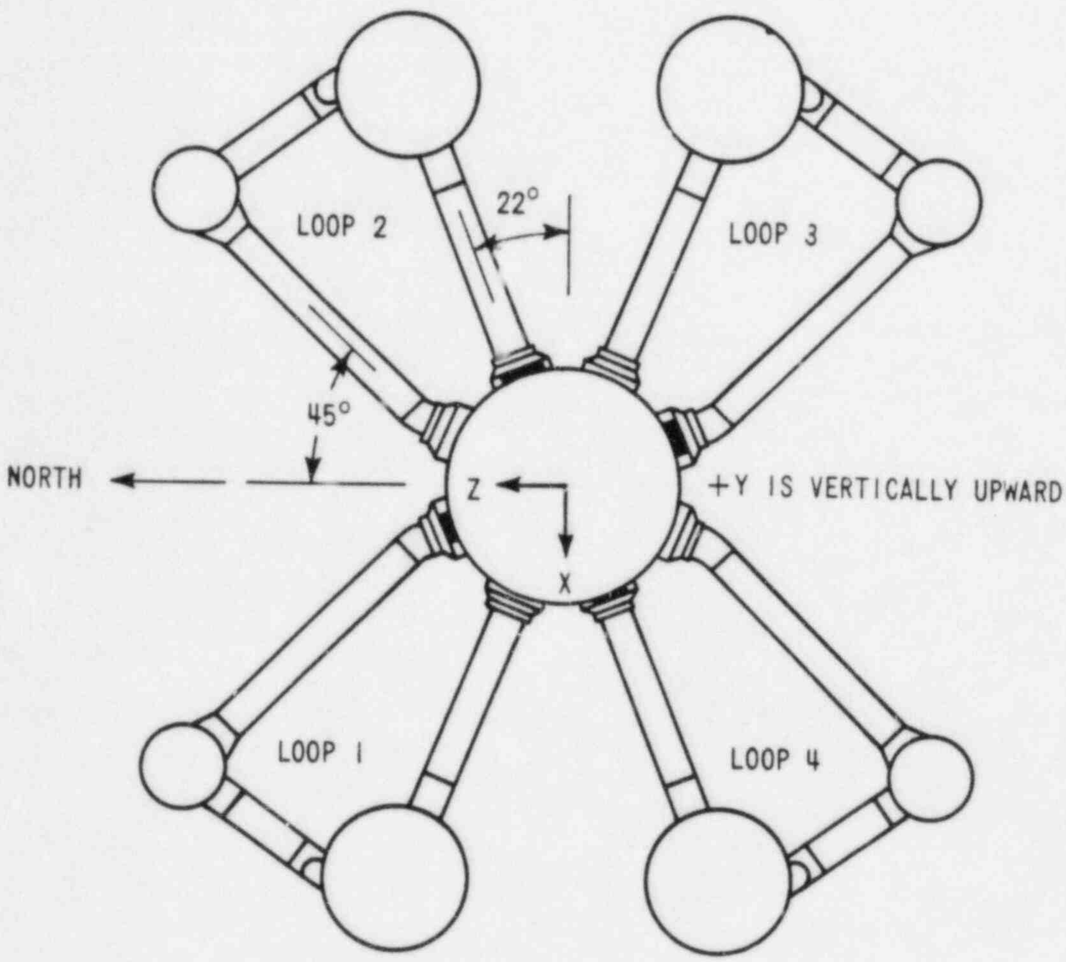

Figure 1-3. Reactor Coolant Loop Coordinate System

(a,c)