SNUPPS
Standardized Nuclear Unit
Power Plant System

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Nicholas A. Petrick
Executive Director

September 9, 1981

SLNRC 81-95 FILE: 0541
SUBJ: NRC Request for Information-
Mechanical Engineering

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos.: STN 50-482, STN 50-483, and STN 50-486

Ref: NRC (Youngblood) letter to UE (Bryan) and KGE (Koester), dated August 25, 1981, Same subject

Dear Mr. Denton:

The referenced letter requested additional information for the SNUPPS FSAR in the area of mechanical engineering. The enclosure to this letter provides the requested information and will be incorporated in Revision 7 to the SNUPPS FSAR.

Very truly yours,

N A Petrick

Nicholas A. Petrick

RLS/dck/3a29

Enclosures

cc: J. K. Bryan, UE
G. L. Koester, KGE
D. T. McPhee, KCPL
W. A. Hansen, NRC/Cal
T. E. Vandel, NRC/WC
D. F. Schnell, UE

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Q210.2 The applicant states that all circumferential breaks in the RCS piping are assumed to result in a limited separation such that the maximum flow area is less than a full break area. The applicant must provide the design information assumed for each location where limited break areas are postulated including gap size, restraint stiffness, blowdown force, and maximum restraint deflection. The results of the time-history analysis (if used) should include the break area vs. time and mass flux rate vs. time which were used to calculate the subcompartment pressurization.

In addition, all restraint locations on the RCS piping must be shown.

RESPONSE

In the reactor coolant loop analysis described in Section 3.6, limited break areas are assumed at the reactor vessel inlet and outlet nozzles. At these locations, the break is limited because of the physical constraint built into the plant design. At this location, the reactor coolant piping restraints located in the shield wall annulus limit the break opening area. A description of these restraints specifically for the SNUPPS units is contained in Section 5.4.14.

In the reactor coolant system analysis, all other circumferential breaks are assumed to be double-ended. However, because of the physical configuration of the plant, these breaks are also limited in area. The specific restraint configurations which limit the break opening are described in Section 5.4.14.

Refer to revised Sections 3.6 and 5.4.14.

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in order to verify the design basis break locations in the reactor coolant loop noted therein.

At all postulated circumferential break locations, the maximum loop piping displacements, as determined by the dynamic RCS analysis or the location of pipe restraints, are such that the separation results in a limited flow area. However, in the reactor coolant loop analysis, limited break areas are only postulated at the reactor vessel inlet and outlet nozzles. At these locations, the break area is limited to approximately one square foot. This reduced break area is justified based on the configuration of the plant. Specifically, reactor coolant piping restraints located in the shield wall annulus (as described in Section 5.4.14) limit the movement of the reactor coolant pipe such that a full double-ended break could not develop. For all other break locations in the reactor coolant loop, full double-ended break locations are assumed. Longitudinal breaks are assumed to have an opening area equal to one flow area of the pipe.

2. Pipe breaks are postulated to occur in the following locations in Class 1 piping runs or branch runs outside the primary reactor coolant loops and pressurizer surge line as follows:

   (a) The terminal ends of the piping or branch run.

   (b) Any intermediate locations between the terminal ends where stresses, calculated using equations (12) and (13) of the ASME B&PV Code, Section III, Subsection NB, exceed 2.4 Sm, where Sm is the design stress intensity, as given in the ASME B&PV Code, and the stress range calculated, using equation (10) of the ASME B&PV code, exceeds 2.4 Sm.

   (c) Any intermediate locations between terminal ends where the cumulative usage factor, derived from the piping fatigue analysis, under the loadings associated with the OBE and operational plant conditions, exceeds 0.1.

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(d) Additional locations of maximum stress intensity or cumulative usage factor to assure a minimum of two break locations between terminal ends.

A complete discussion of the reactor coolant loop break location is provided in Reference 1.

b. ASME B&PV code, Section III - Class 2 and 3 Piping Within Protective Structures

1. Breaks are postulated to occur at terminal ends, including:

   (a) Piping-pressure vessel or equipment nozzle intersection

   (b) High-energy/moderate-energy boundary

   (c) Pipe to anchor intersection

generator outlet nozzle. This restraint is attached to the secondary shield wall and extends horizontally to the vertical run of the crossover leg pipe.

b. Hot leg

A restraint is located near the 50-degree elbow in the hot leg to prevent excessive displacement of the hot leg following a postulated guillotine break at the steam generator inlet nozzle. This restraint consists of structural steel members which transmit loads to the concrete structure. This restraint is shown in Figure 5.4-20.

c. Hot leg and cold leg lateral restraints

A restraint on each reactor coolant system hot leg and cold leg is located near the reactor vessel safe-end to reactor coolant system piping weld with the reactor vessel primary shield wall to prevent excessive displacement of either the hot leg or the cold leg following a postulated guillotine break at the reactor vessel safe-end to piping weld. These restraints are shown in Figure 5.4-21.

5.4.14.3 Design Evaluation

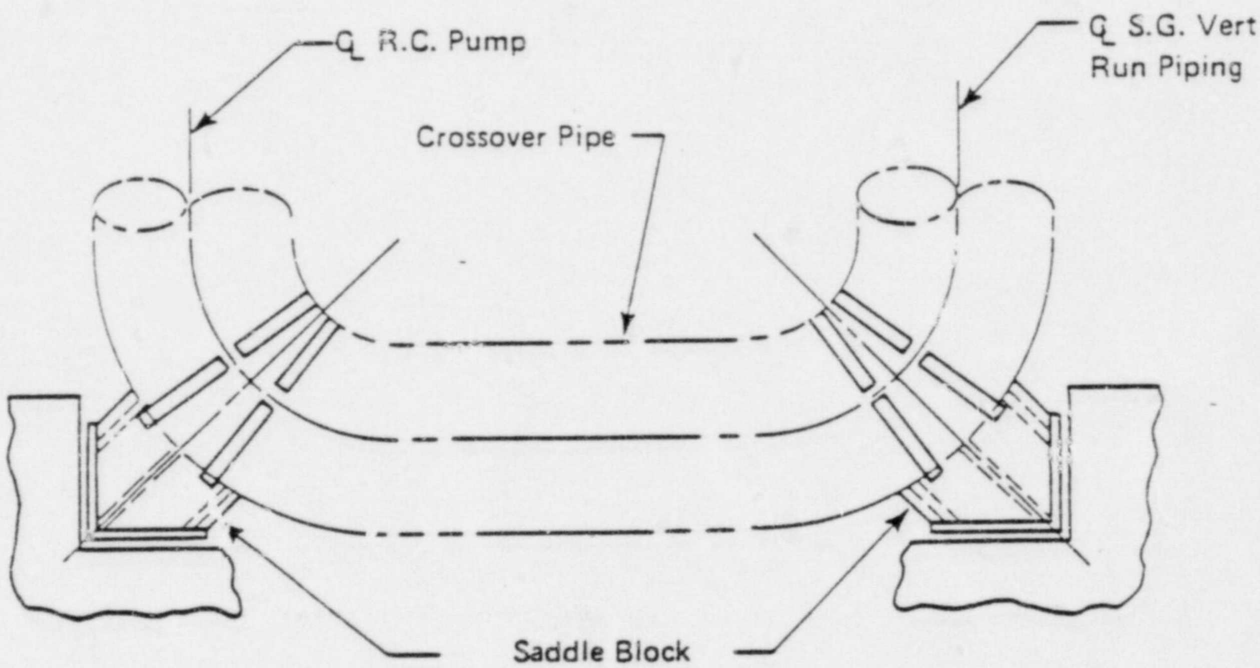
Detailed evaluation ensures the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of a large or small seismic disturbance and/or LOCA conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values as described in Section 3.9(N).1.4.

The safe shutdown earthquake and design basis LOCA, resulting in a rapid depressurization of the the system, are required design conditions for public health and safety. The methods used for the analysis of the safe shutdown earthquake and LOCA conditions are given in Sections 3.9(N).1.4.

5.4.14.4 Tests and Inspections

Nondestructive examinations are performed in accordance with the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF.

Figure 5.4-18



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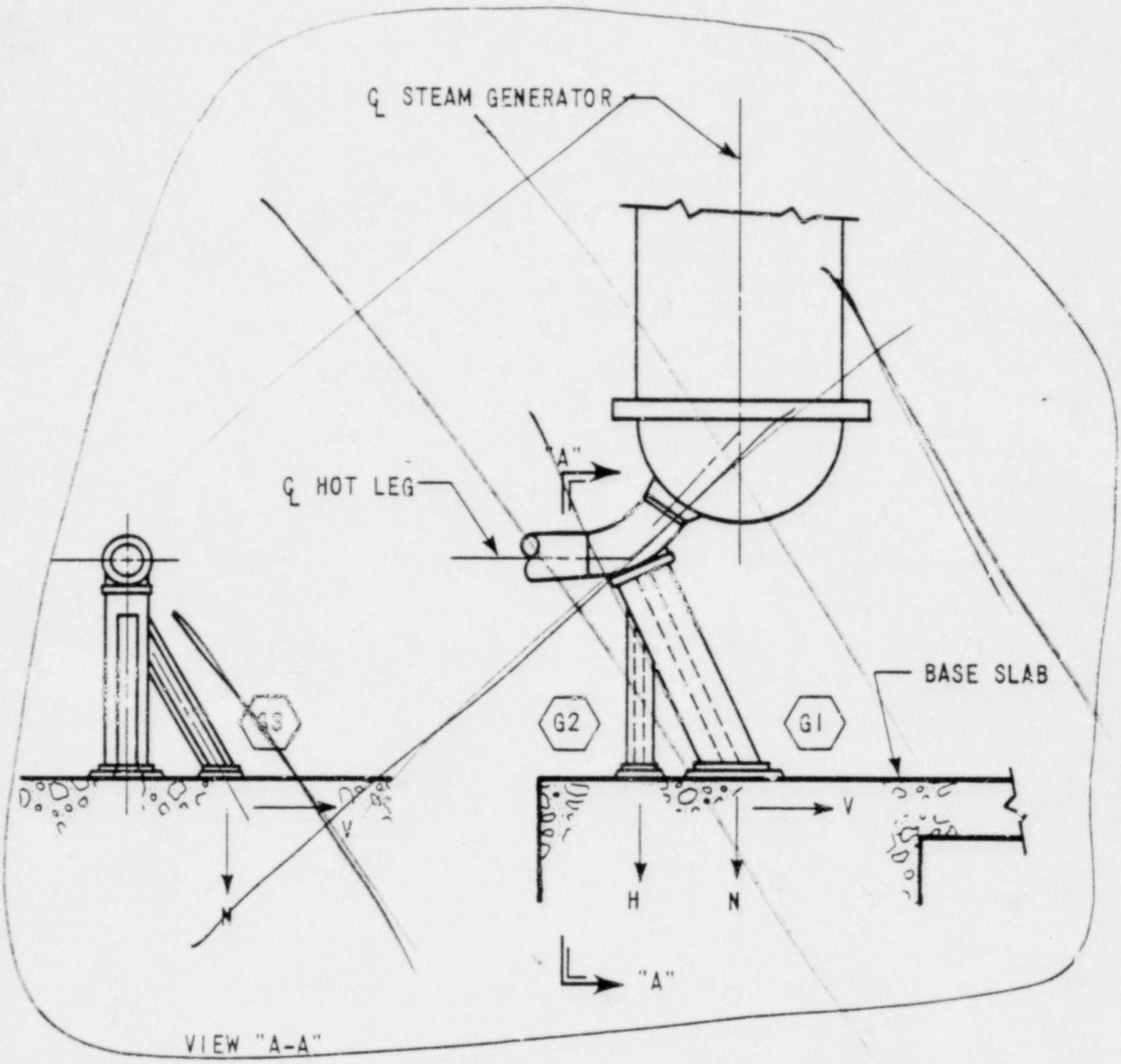


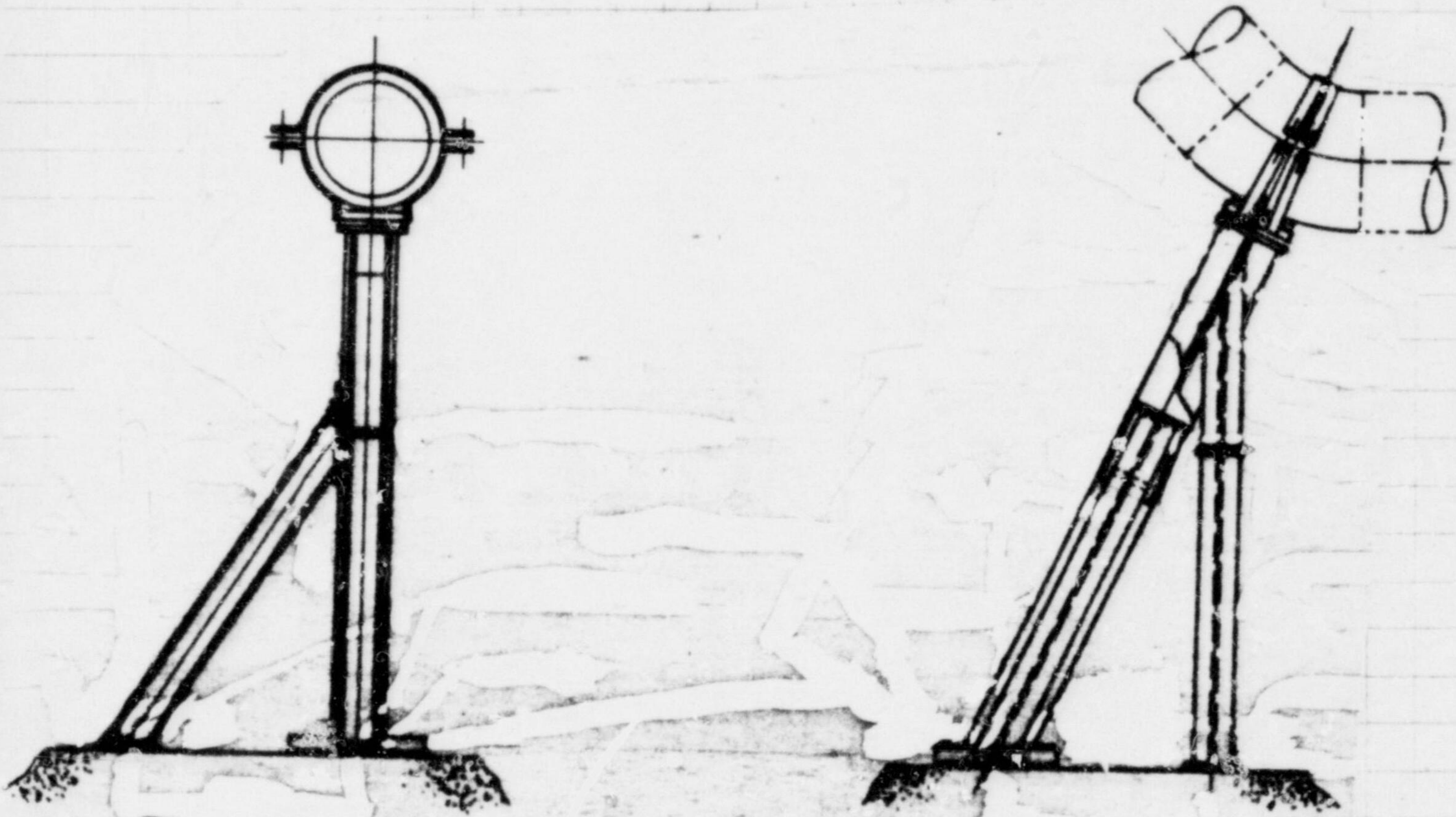
Replace with attached

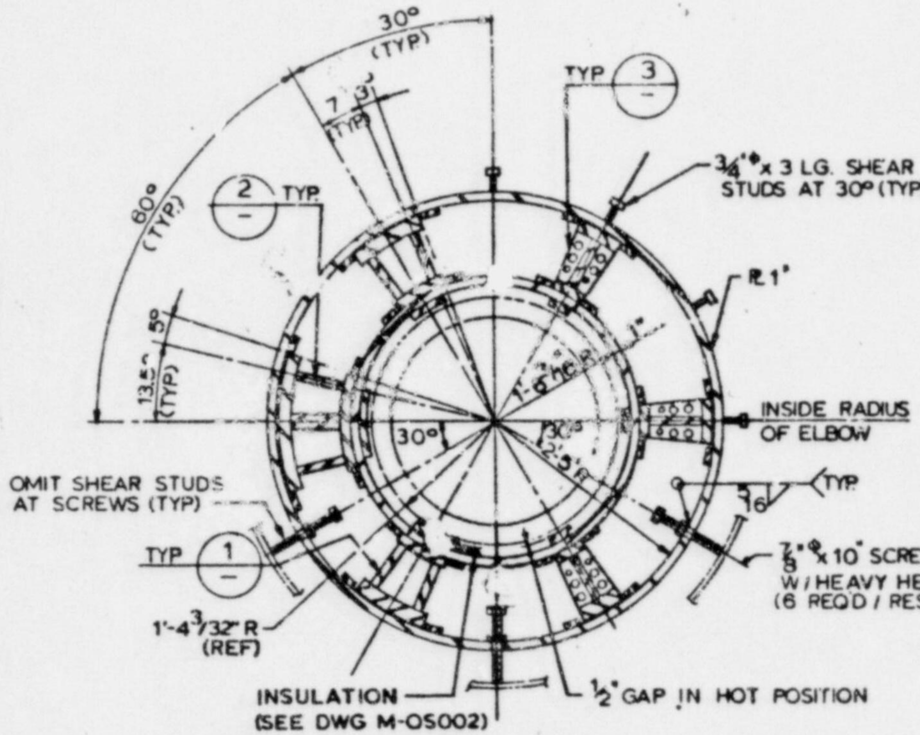
SNUPPS

FIGURE 5.4-20

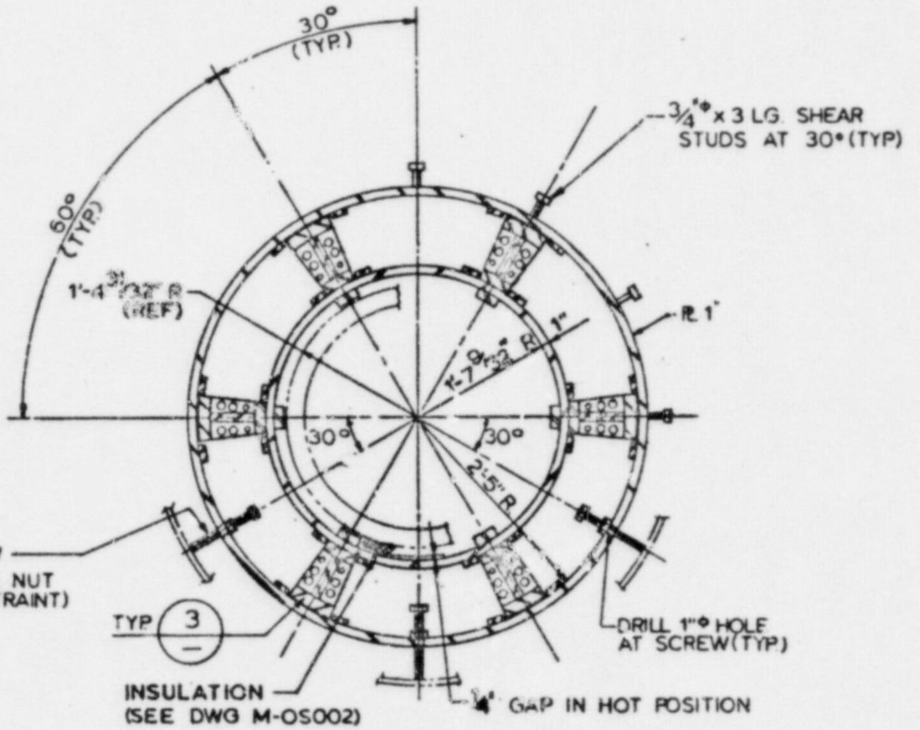
HOT LEG RESTRAINT

Figure 5.4-20





TYPICAL COLD PIPE



TYPICAL HOT PIPE

Hot & Cold Leg Lateral Restraints

F 5.4-21