SUSQUEHANNA STEAM ELECTRIC STATION
DESIGN ASSESSMENT REPORT (DAR)

Revision 1

PREFACE

This Report contains data, descriptions and anaylsis relative to the adequacy of the Susquehanna Steam Electric Station design to accommodate loads resulting from safety relief valve (SRV) discharge and/or loss-of-coolant accident (LOCA) conditions.

DAR TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

CHAPTER 1

GENERAL INFORMATION

TABLE OF CONTENTS

CHAPTER 1

FIGURES

CHAPTER 1

TABLES

1.0 GENERAL INFORMATION

1.1 PURPOSE AND ORGANIZATION OF REPORT

The purpose of this report is to present evidence that the Susquehanna Steam Electric Station (SSES) design margins are adequate should the plant be subjected to the recently defined thermohydrodynamic loads which result from safety relief valve (SRV) operations and/or discharges during a loss-of-coolant accident (LOCA) in a GE boiling water reactor (BWR).

1.2 HISTORY OF PROBLEM

In April 1972 at the German AEG-Kraftwerk Union Wurgassen Nuclear Plant, a boiling water reactor (BWR) safety relief valve (SRV) was opened during startup testing and failed to close. The reactor remained at full pressure, and the valve discharged reactor steam into the containment suppression chamber until the suppression pool water heated from just above ambient to almost 170°C (in approximately 30 minutes). Pulsating condensation developed and large impulsive forces with substantial underpressure amplitudes acted upon the containment, eventually causing leakage from the bottom liner plate. Therefore, concern was expressed that the structural integrity of other BWR pressure containment systems could be sensitive to SRV induced dynamic loads.

The Nuclear Regulatory Commission (NRC) issued Bulletin 74-14 to all BWR owners on November 14, 1974 to alert them to the potential problems of condensation instability (Wurgassen effect) due to SRV operation. The NRC requested verification that BWR suppression pools had been designed to withstand loads similar to those which were being experienced. In January 1975 the General Electric - Nuclear Energy Program Division (GE-NEPD) identified the following dynamic loading conditions which had not been fully considered in the design criteria of Mark II BWR containments:

a. Main steam SRV discharge thermo-hydrodynamic phenomena.

b. Design basis accident (DBA): loss-of-coolant accident (LOCA) hydrodynamic phenomena

Following the GE announcement, the containment construction sequence for the SSES was altered to enable the Pennsylvania Power and Light Company (PP&L) and its architect-engineer, Bechtel Power Corporation, to ascertain the effect of these phenomena on the existing SSES design. A task force was formed in March 1975 with representatives from Bechtel-San Francisco, GE-NEPD, PP&L, and Philadelphia Electric Company to evaluate existing design criteria with respect to the newly defined SRV and DBA-LOCA loadings. In May 1975 Bechtel completed a preliminary study incorporating the effects of the new phenomena in the design criteria for the SSES suppression chamber structures and safety related equipment. As a result of this investigation, it was decided that the following civil-structural modifications were to be incorporated immediately in the containment design to aid in load transfer and add additional conservatism to the existing design:

a. The number of reinforcing bars in the suppression chamber vertical walls was increased.

b. The number of embedments in the suppression chamber walls for downcomer/piping restraints was increased to accommodate future requirements.

c. Anchor bolts were placed on the underside of the diaphragm slab to accommodate additional supports for the SRV discharge piping for horizontal runs should they be needed.

d. Additional anchor bolts were placed within the drywell wall to allow installation of additional snubbers and pipe restraints, if required.

e. The diaphragm slab shear reinforcement was changed from a 45° to a 90° orientation (with respect to the horizontal plane) to accommodate the most conservative pool swell uplift loadings yet predicted.

It became evident that a complex technical issue existed for all Mark II plants, and PP&L sought to create a unified utility group to address the matter. A Mark II BWR containment owners group was formed in June 1975 to define precisely the suppression pool dynamic loads and explore ways to assess their impact. As the direct result of action taken by the Mark II containment owners organization, a generic Dynamic Forcing Function Information Report, NEDE-21061P Rev. 1, which was also known as the DFFIR, was issued jointly by GE-NEPD and Sargent and Lundy for the Mark II owners in September 1975.

Based on the analytical techniques included in the DFFIR, a preliminary SSES unique containment design assessment was submitted by PP&L to the Nuclear Regulatory Commission (NRC) on March 15, 1976.

As the body of the useful supportive data increased, Revision 2 of the DFFIR was issued jointly by GE-NEPD and Sargent and Lundy for the Mark II containment owners group on September 1, 1976, as NEDO/NEDE 21061, Rev. 2. It was at this time renamed the DFFR.

The licensing documentation considered for the SSES is summarized in Table 1-1.

1.3 SSES CONTAINMENT PROGRAM

PP&L is a member of the Mark II owners group that was formed in June, 1975 to define and investigate the dynamic loads due to SRV discharge and LOCA. The Mark II owners group containment program concentrated initially on the tasks required for the licensing of the lead plants (Zimmer, LaSalle, and Shoreham). This phase of work, called the short term program, is complete and a longer term program is underway. The final goal of the Mark II program is to evolve a complete DFFR which will support the plant-unique DARs submitted by each plant for its license to operate.

After gaining some understanding of the containment loads through the initial Mark II work, PP&L decided to find a qualified consultant to supplement in-house technical resources and assist in the determination of a realistic course of action for Susquehanna. In November, 1976, Stanford Research Institute, now called SRI International (SRI), was selected, and an information exchange between SRI and PP&L ensued to determine what caused the greatest loads on the containment structure. After conducting a complete review of known data from the Mark II program and other knowledgeable persons and organizations, PP&L and SRI decided that the loads from main steam safety relief valve (SRV) discharge were the key loads to be controlled. A study of possible methods of controlling the load and a review of what activities were occurring in Europe led PP&L and SRI to the conclusion that an SRV discharge mitigating device (quencher) should be employed to reduce this loading on the Susquehanna containment. Although the Mark II owners group had quencher-related tasks in their program, these tasks were not sufficiently timely to satisfy SSES-construction schedule needs.

From reviewing the work done in Europe by such firms as ASEATOM, MARVIKEN, and Kraftwerk Union, PP&L discovered that all known quencher designs were based on data from Kraftwerk Union (KWU). Thus, in March, 1977, SRI, Bechtel (the SSES Architect/Engineer) and PP&L visited KWU for discussion and tour of quencher-related facilities. In late July, 1977, PP&L employed the services of KWU to design a SSES-unique quencher device.

Kraftwork Union provided PP&L a package of significant design and test reports pertaining to the quencher development to demonstrate design adequacy and quality of their device (refer to Table 1-1). These documents were submitted to the NRC in January, 1978. The quencher load specification was submitted to the NRC in April, 1978. To verify KWU's design approach, a full-scale SSES unique unit cell test, as described in Chapter 8, was performed by KWU for PP&L. The documentation of this test series and verification of the design specification was submitted in March, 1979. Subsequently the quencher design by KWU for use on SSES has been adopted as the SRV discharge used by six of the seven other Mark II owners and the SSES program has become the generic Mark II program.

The definition of LOCA loads (Section 4.2) is in basic accordance with the Mark II program. In addition though, PP&L has decided to conduct a series of transient steam blowdown tests in a modified GKM II test tank in Mannheim, Germany (refer to Chapter 9). These tests will provide data to resolve NRC concerns on the differences in vent configuration between the original GE 4T facility and a prototypical Mark II containment and to verify the condensation oscillation load specification used on the SSES design.

Table 1-1 provides a summary of the documentation supporting the SSES licensing.

In addition, Table 1-4 provides a comparison of the SSES program for SRV and LOCA loading with the NUREG 0487 acceptance criteria, Lead Plant Program and Generic Long Term Program. In accordance with the directions of the NRC staff at the October 19, 1978 meeting with the Mark II Owners Group these positions assume that the use of the SRSS method of load combination will be accepted for use on the Mark II containments.

1.4 PLANT DESCRIPTION

The SSES, Units 1 and 2, is being built in Salem Township, Luzerne County, about 5 miles northeast of the Borough of Berwick. Two generating units of approximately 1,100 megawatts each are scheduled for operation: Unit 1 for November 1, 1980, and Unit 2 for May 1, 1982. General Electric is supplying the nuclear steam supply systems; Bechtel Power Corporation is the architect-engineer and constructor.

The reactor building contains the major nuclear systems and equipment. The nuclear reactors for Units 1 and 2 are boiling water, direct cycle types with a rated heat output of 11.2×10^9 Btu/hr. Each reactor supplies 13.4×10^6 lb/hr of steam to the tandem compound, double flow turbines.

1.4.1 Primary Containment

The containment is a reinforced concrete structure consisting of a cylindrical suppression chamber beneath a truncated conical drywell. Figure 1-1 shows the geometry of the containment and internal structures. The conical portion of the primary containment (drywell) encloses the reactor vessel, reactor coolant recirculation loops, and associated components of the reactor coolant system. The drywell is separated from the wetwell, ie, the pressure suppression chamber and pool, by the drywell floor, also named the diaphragm slab. Major systems and components in the containment include the vent pipe system (downcomers) connecting the drywell and wetwell, isolation valves, vacuum relief system, containment cooling systems, and other service equipment. The cone and cylinder form a structurally integrated reinforced concrete vessel, lined with steel plate and closed at the top of the drywell with a steel domed head. The carbon steel liner plate is anchored to the concrete by structural steel members embedded in the concrete and welded to the plate.

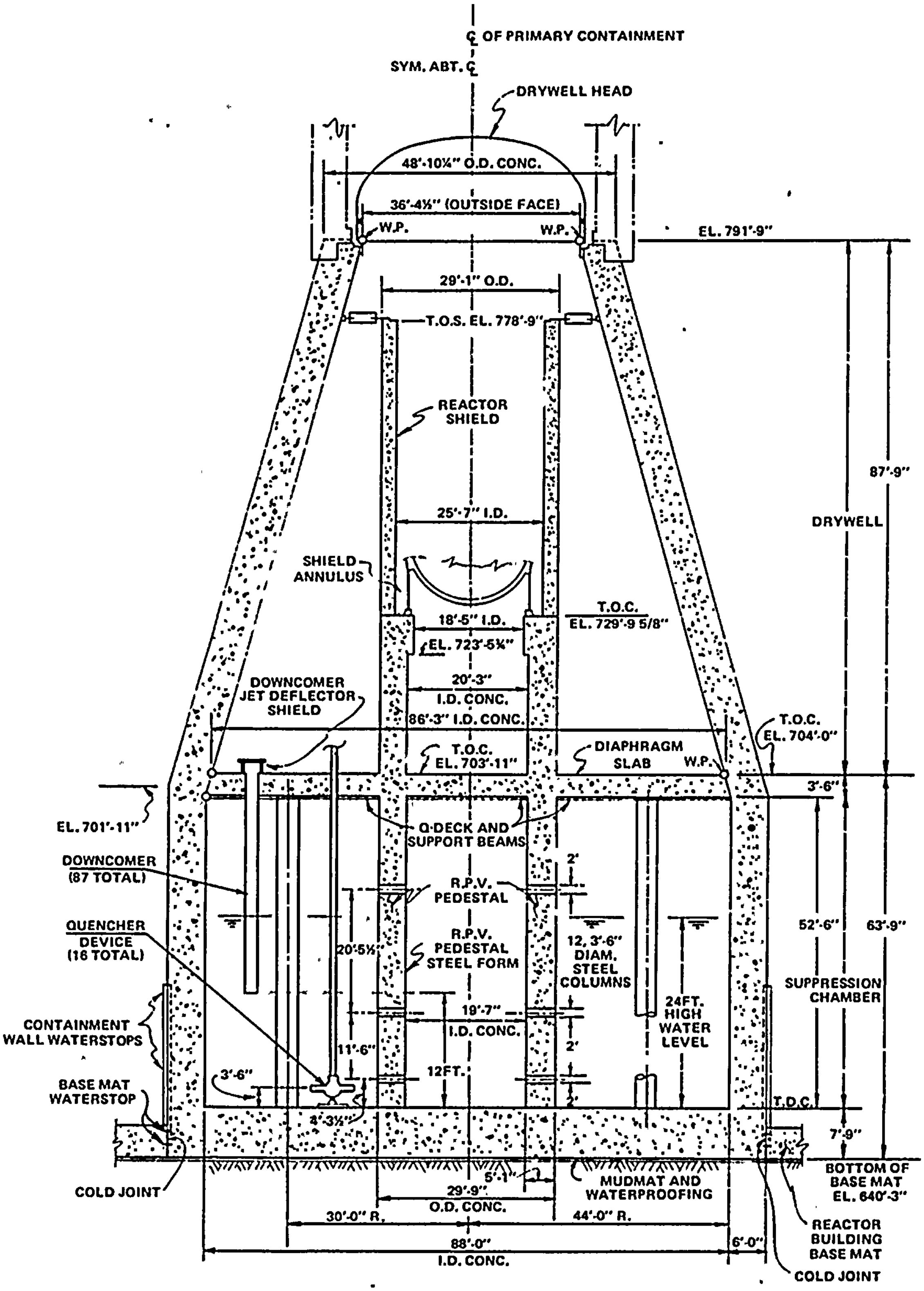
The entire containment is structurally separated from the surrounding reactor building except at the base foundation slab (a reinforced concrete mat, top lined with a carbon steel liner plate) where a cold joint between the two adjoining foundation slabs is provided. The containment structure dimensions and parameters are listed in Tables 1-2 and 1-3. A detailed plant description can be found in the SSES FSAR, Section 3.8.

1.4.1.1 Penetrations

Services and communication between the inside and outside of the containment are made possible by penetrations through the containment wall. The basic types of penetrations are the drywell head, access hatches (equipment hatches, personnel lock, suppression chamber access hatches, CRD removal hatch), electrical penetrations, and pipe penetrations. The piping penetrations consist basically of a pipe with plate flange welded to it. The plate flange is embedded in the concrete wall and provides an anchorage for the penetration to resist normal operating and accident pipe reaction loads.

1.4.1.2 Internal Structures

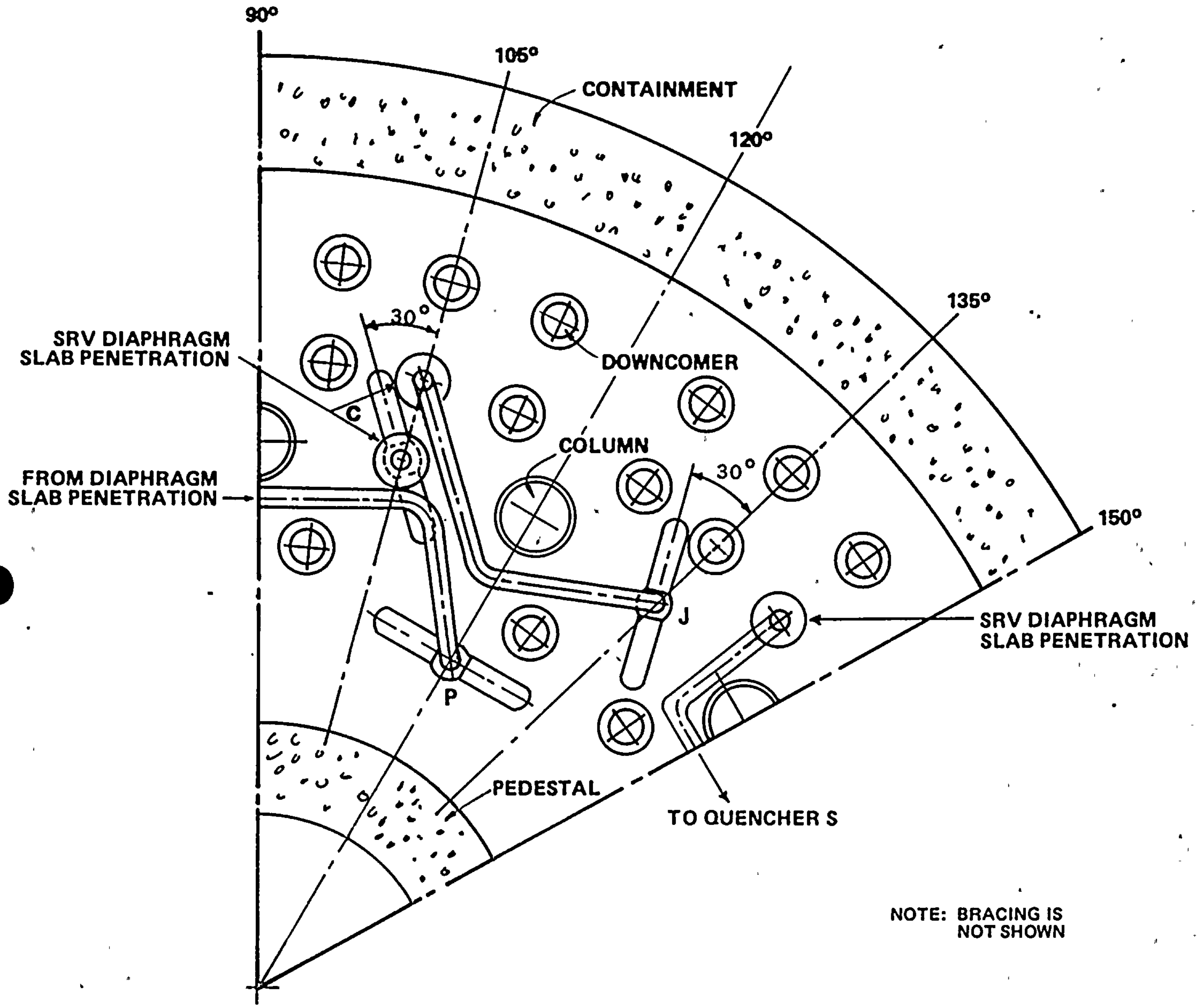
The internal structures consist of reinforced concrete and structural steel and have the major functions of supporting and shielding the reactor vessel, supporting the piping and equipment, and forming the pressure suppression boundary. These structures include the drywell floor (diaphragm slab), the reactor pedestal (a concentric cylindrical reinforced concrete shell resting on the containment base foundation slab and supporting the reactor vessel), the reactor shield wall, the suppression chamber columns (hollow steel pipe columns supporting the diaphragm slab), the drywell platforms, the seismic trusses, the quencher supports, and the reactor steam supply system supports. See Figures 1-1 through 1-4 and Tables 1-2 and 1-3.



SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
DESIGN ASSESSMENT REPORT

CROSS SECTION OF CONTAINMENT

FIGURE 1-1



NOTE: BRACING IS NOT SHOWN

Rev. 2, 5/80

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
DESIGN ASSESSMENT REPORT

SUPPRESSION CHAMBER
PARTIAL PLAN

FIGURE 1-2