DTV - [illegible] & Test

FINAL

SYSTEM DESCRIPTION

(Index No. 53)

CONTAINMENT SYSTEM

JERSEY CENTRAL POWER & LIGHT COMPANY
THREE MILE ISLAND NUCLEAR STATION
UNIT NO. 2

Issue Date
May, 1976

Prepared by: John Hooper

Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, N.J.
07649

198 020

TABLE OF CONTENTS

FOR

CONTAINMENT SYSTEM

198 021

1.0 INTRODUCTION

1.1 SYSTEM FUNCTIONS

The primary function of the Containment System is to minimize the consequences of the maximum credible accident* to an acceptable limit, thereby protecting the public against potential radiation hazards. Thus the Containment System for Three Mile Island Nuclear Station - Unit 2 comprises the Containment Structure (or more commonly called the Reactor Building) together with the Engineered Safety Features as controlled by the Safety Features Actuation System provided to mitigate the consequences of an accident. The Reactor Building also functions as a shield for the Nuclear Steam Supply System against aircraft impact and other external missiles. During full power operating conditions, the Reactor Building functions to attenuate any radiation escaping from the primary-secondary shield complex, so that the radiation level outside the Reactor Building is kept within the limits set forth in 10CFR 20.

1.2 SUMMARY DESCRIPTION OF THE SYSTEM

1.2.1 The Reactor Building

The Reactor Building is a prestressed (post-tensioned), reinforced concrete structure composed of a cylindrical wall and a shallow dome roof on a conventionally reinforced flat foundation mat. The structure provides the final biological shielding for normal operating and accident conditions. It encloses the reactor, the reactor coolant pumps, steam generators, pressurizer, and associated piping and other equipment.

*After analyzing various credible accident events resulting from component failures, design deficiencies, and operating errors, the maximum credible accident is found to be one following the "Loss of Coolant Accident" (LOCA), as a result of double-ended pipe break in the main coolant system. Consequences of this accident have been used as the basis for design of Containment System Components, and the accident has been termed as "Design Basis Accident (DBA)".

198 022

1.2.1.1 Liner and Penetrations

The inside face of the Reactor Building concrete is lined with carbon steel to ensure a high degree of leak tightness during operating and post-accident conditions. Nominal plate thickness for the steel liner is 1/2 inch for dome and ring girder, 3/8 inch for the cylindrical wall and ¼ inch for the base. The liner plate and the concrete shell are penetrated by various openings such as equipment access, personnel access lock and other mechanical and electrical penetrations.

The configuration, basic dimensions and internal components of the Reactor Building are shown in Figure 1.

1.2.1.2 Interior Structure

The Reactor Building interior structure comprises the following: (see Figure 1)

a. The basement floor at el. 282'-6", the intermediate floor at el. 305'-0", and the operating floor at el. 347'-6".

b. The primary shield wall, a reinforced concrete structure surrounding the reactor vessel and extending upward from the Reactor Building basement floor to form the walls of fuel transfer pool.

c. The secondary shield wall, a reinforced concrete structure surrounding the reactor coolant equipment, including piping, pumps and steam generators.

d. Equipment supports, pipe supports and pipe whip restraints.

1.2.2 Safety Features

198 023

The engineered safety features include containment spray systems, containment air recirculation and cooling systems, emergency coolant injection, core flooding and containment isolation system. Containment air recirculation and cooling units, backed up by reactor building sprays, are provided for emergency atmosphere cooling.

Chemical additives contained in the spray systems will reduce the post-accident concentration of iodine and other radioactive materials in the building atmosphere. Emergency coolant injection and core flooding protect the fuel cladding by cooling the reactor core and covering it with borated water to provide sufficient negative reactivity to maintain shutdown. These features are provided to mitigate the consequences of the postulated accident by removing heat from the reactor core and the Reactor Building atmosphere, reducing the concentration of radioactivity in the Reactor Building atmosphere, inserting negative reactivity into the reactor, and isolating the Reactor Building. The energy releases from the LOCA and as modified by the contribution of the safety features could result in a condition not exceeding the selected design pressure of 60 psig and peak temperature of 286 F.

1.3 SYSTEM DESIGN REQUIREMENTS

1.3.1 The Reactor Building

In order to fulfill its primary function, the Reactor Building, access openings and penetrations, are required to be designed to accommodate in excess of the design accident internal pressure of 55.7 psig and a maximum temperature of 286F, without exceeding the design leakage rate of 0.2 percent by weight of contained atmosphere in 24 hours.

Specifically the Reactor Building and its components are designed to withstand the following conditions:

A. Operating Condition

198 024

This condition includes the dead loads, the anticipated live loads, the prestress load, load induced by temperature gradient through the walls in summer and winter, in credible combination with loads due to 80 mph wind or due to the maximum expected seismic disturbances at the site, one half the Safe Shutdown Earthquake (½ SSE), having a horizontal ground acceleration of 0.06g and vertical ground acceleration of 0.04g). The design of the Reactor Building also includes the loads due to test pressure of 69 psig and due to transient temperatures during start-up and shut-down conditions.

B. Accident Conditions

In the design of the Reactor Building, a pressure of 60 psig and associated transient temperatures have been considered in combination with normal operating loads and the loads developed as a result of adverse environmental conditions. The adverse environmental loadings in this connection are those due to a seismic disturbance of twice that assumed during the operating condition (the "Safe Shutdown Earthquake (SSE)", having a horizontal ground acceleration of 0.12g and vertical ground acceleration of 0.08g). Moreover, a tornado loading of 360 mph has been considered in combination with the normal operating loads. In order to ensure the adequate margin of safety against the failure of the Reactor Building to fulfill its functional requirements, the Reactor Building and its components are designed to have an ultimate capacity to withstand internal pressure of 90 psig and associated temperatures together with normal operating loads. The following codes, standards and recommended practices provided the guidelines in designing the Reactor Building.

a. Regulations for Protection from Fire or Panic- Commonwealth of Pennsylvania.

198 025

b. Building Code Requirement for Reinforced Concrete - ACI 318

c. AISC Manual of Steel Construction

d. ASME Boiler and Pressure Vessel code: Section III, Nuclear Vessels; Section VIII, Unfired Pressure Vessels; Section IX, Welding Qualification (Applicable Portions).

1.3.2. Interior Structure

The intermediate and operating floors are designed for the combinations of dead load and anticipated live loads - which include the weight due to Reactor Vessel Head (in an assigned area), the core flooding tanks, piping and other equipment. The primary shield wall is designed for maximum pressure differential of 139.5 psi (with an ultimate capacity of 210 psig) and the secondary shield wall is designed for the pressure differential of 35.25 psi (with an ultimate capacity of 45 psig) and pipe rupture load. The refueling pool walls are designed for hydrostatic pressure. The supports for equipment are designed for normal operating loads combined with rupture of any pipe plus design basis earthquake.

2.0 DETAILED DESCRIPTION OF SYSTEM

2.1 COMPONENTS

The following are the major components of the Reactor Building.

2.1.1 Concrete Shell

The concrete shell is adequately prestressed to eliminate tensile stresses due to membrane forces from the design loads. Moreover, the strain levels in the materials of construction are kept within such limits, that the integrity of the liner is maintained under all credible load combinations (Section 1.3.1). A tendon access gallery below the foundation mat is provided to facilitate the installation, tensioning and grouting of the tendons.

198 026

The concrete for the Reactor Building shell and foundation mat has a minimum ultimate compressive strength of 5000 psi. The reinforcement in the Reactor Building conforms to requirements of ASTM A615, Grade 60.

Production and delivery of the concrete was in accordance with B&R Specification 2555-20, Production and Delivery of Concrete. The placement and testing of the concrete and the reinforcing steel, and the method of cadwelding of the reinforcing bars were in accordance with B&R Specification 2555-54, Reactor Building Concrete.

2.1.2 Prestressing System

The prestressing system used for the Reactor Building is the S/H 54-5 Tendon-Anchorage System as developed by Stressteel Corporation. A tendon consists of 54 - 1/2 inch diameter strands. Strands conform to ASTM A416, Grade 270, low relaxation strands as manufactured by CF&I Roebling Corporation.

The component parts of anchorage hardware consisted of:

1. Bearing plate ASTM A516, Grade 70
2. Splay plate AISI 1040
3. Wedge plate ASTM A517, Grade F
4. Wedges AISI 1117 (Case hardened after machining)

Arrangement and spacing of tendons ensures the following minimum effective forces at 40 years:

Cylinder-Hoop 742 k/ft. (156 tendons)

-Vertical 500k/ft. (167 tendons)

Dome - 42 tendons in each of the three bands developing 700 K/ft. to 750 K/ft. in the meridional direction and 300 K/ft. to 750 K/ft. in the hoop direction.

The tendon is held in designated tension by means of wedges and

198 027

wedge plates. The prestress load is transferred to the concrete through the bearing plate. The tendon duct is 5½ inch inside diameter, galvanized steel, ribbed, which is semi-rigid tube except in areas of high curvature where flexible tube has been used. The ducts are fabricated and installed to conform to the requirements of B&R specification 2555-54. After installation and tensioning of tendons, the tendon ducts are filled with Portland cement grout. Basicity, which is the inherent property of this kind of grout, provides a passivating environment around the prestressed steel and excludes aggressive agents thus precluding corrosion of the prestressing steel.

2.1.3 Liner and Penetrations

The liner is designed to ensure that containment integrity is not violated and leakage is limited to a maximum of 0.2 percent by weight in 24 hours at the design basis accident pressure of 55.7 psig. The liner is anchored to the concrete shell by means of tee-shapped members(ST 3I) running vertically and meridionally along the cylinder and dome, respectively. Liner plate material conforms to ASME SA516, Grade 55, and penetration material (including the reinforcing and face plates) conforms to ASME SA516, Grade 70 and meets the nil-ductility requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Paragraph N-1211. The liner and penetrations were fabricated and constructed according to B&R specification 2555-44, Reactor Building Liner.

The Reactor Building Penetrations Forced Air Cooling System provides cooling which limits the two feedwater and four main steam penetration concrete to 150F. Additional details are discussed in System Description No. 35B.

198 028

2.1.4 Interior Structure

The primary shield wall in conjunction with secondary shield wall, will attenuate the radiation level from sources within the reactor vessel and the reactor coolant system, in order to allow limited access to the Reactor Building during normal full power operation. The neutron and gamma-ray heating of the primary shield will be dissipated to the Reactor Building cooling system.

Reactor Building cooling fans circulate air through ducts in and around the primary cavity, to restrict concrete temperatures in the primary shield below 150F. Audible and visual alarms and a multipoint temperature recorder to monitor primary shield concrete temperatures are located in the Control Room. Additional details are discussed in the system description for Reactor Building Ventilation and Purge Systems, Index No. 35A. The material that is used for primary and secondary shield walls and the intermediate and operating floor slabs is reinforced concrete having an effective density (including reinforcing) of at least 145 lbs/cu.ft.

The primary and secondary shield walls serve as the refueling structure and as restraints for the reactor coolant components under pipe rupture conditions.

Primary and secondary shield walls, together with 3'-0" precast top slabs, over the Reactor Cavity Area, also serve as a barrier to prevent potential internal missiles from reaching the containment liner. Precast slabs are removable for refueling and repair, maintenance, or removal of equipment.

198 029

2.2 INSTRUMENTS, CONTROLS, ALARMS AND PROTECTIVE DEVICES

Control instrumentation, alarms and protective devices associated with the atmospheric radiation monitoring in the Containment are described in the system description for Radiation Monitoring System, Index No. 52.

Instrumentation, provided to detect temperatures, pressures and humidity in the containment atmosphere and including temperature monitoring of the primary shield concrete is described in Table 11 of the System Description for Reactor Building Ventilation and Purge Systems, Index No. 35A. Monitoring of penetration pressurization is provided and is described in System Description, Index No. 26, Reactor Building Penetrations, Air and Nitrogen Pressurization. The protective devices used to permit access to the containment during normal operation are discussed below. Operation of the Safety Features Actuation System as it controls the Engineered Safety Features is described in System Description No. 50.

Two personnel access openings are provided in the Containment Structure. The openings are designed with double doors to form airlocks such that access to the interior of the Reactor Building can be made during operation without breach of containment. Both doors for each lock open toward the interior of the Reactor Building. Each lock has internal and external operating wheels for the interior and exterior doors as well as means to indicate the position of the opposite door. Doors are mechanically interlocked so that one cannot be opened unless the other is completely closed, latched and sealed. Design of air locks also incorporates a locking device on each inside door which is actuated when the pressure differential across the door exceeds 2 psi. This feature is intended to prevent sudden increases or decreases in pressure in the lock, when containment pressure is high due to testing or as a result of a reactor incident.

198 030

Each lock has a lighting system and a communication system. This communication system is connected to the total plant communication system for surveillance and emergency use.

3.0 PRINCIPAL MODES OF OPERATION

3.1 STARTUP

The "containment integrity" (as defined in Section 16 of the FSAR) has to be established before startup of the plant. The equipment access hatch cover must be in place and as a minimum, one of the doors of the personnel access locks must be closed. The gate valves and/or blind flanges for each fuel transfer tube must be in closed positions. The flanges of penetration for Incore Instrumentation Cable Replacement must be in place. Any penetration dismounted for repairs or testing must be replaced.

3.2 NORMAL OPERATION

During normal operation, at least one door in each of the personnel locks must be in the locked position. All access to the Reactor Building is to be on a limited basis; i.e., the occupancy time for an individual entering the Reactor Building is to be based on health physics survey of the Reactor Building atmosphere and the specific areas to be occupied. (NOTE: Section 12 of the FSAR describes the Reactor Building shield design criteria and conditions of occupancy, when the plant is in full operation.)

3.3 SHUTDOWN

The equipment access hatch cover can be removed and/or both doors of the personnel locks opened only during cold shutdown and when a health physics survey of the Reactor Building atmosphere has shown that it is safe to breach containment.

However, during the handling of fuel in the Reactor Building with irradiated fuel in the reactor building, at least one door on each personnel access lock shall be closed and the equipment hatch cover shall be in place and secure.

3.4 Special or Infrequent Operation

3.4.1 Post Operational Leakage Rate Tests

Post-operational leakage rate tests will be conducted periodically to ensure that the integrity of containment as defined in paragraph 3.1 is maintained and to determine if any leakage problems have developed since the previous integrated leakage rate test. These tests will consist of integrated leakage rate tests of the containment system and individual leakage rate test of certain components which penetrate or seal the boundary of the containment system. The tests shall be performed at regular intervals, after major repairs or whenever there is any indication of gross leakage. These tests are discussed in Section 16 of the FSAR and in the System Description for Leak Rate Testing System, Index No. 54.

3.4.2 Structural Proof Testing and Prestressing System

The prestressing tendons in theReactor Building are grouted and their anchorages are protected against the influence of external corrosive agents. Individual tendons therefore, are not accessible for inspection, even at anchorages, without some degree of destruction of the protection provided. Therefore, the continual structural integrity of the Reactor Building will be ascertained by periodic pressure proof tests rather than by direct inspection. These tests are to be performed during the 2nd, 10th, and 20th years (approximately) after start of commercial operation of the unit. They shall be conducted at routine shutdown periods and it is acceptable that they be sequenced with leak rate tests.

Procedure for performing these tests including pressure level, measurements, and visual inspection shall be similar to that of initial proof test. Observations recorded during these tests shall be compared with those recorded during initial proof test and evaluated to assess any change in the behavior of the structure. Details of these tests are discussed in Appendix 3D of the Final Safety Analysis Report.

3.5 Emergency

The containment system is an engineered safety feature which functions to minimize the potential hazardous consequences of a LOCA. The structure, along with other engineered safety feature systems, provides this protection.

4.0 HAZARDS AND PRECAUTIONS

The following conditions are considered to be potentially hazardous and are to be reported so that an evaluation of the condition can be made and corrective action, if required, can be instituted. A record of the condition is to be maintained for future reference in determining further degradations.

a. Deterioration or cracking of shell concrete.

198 033

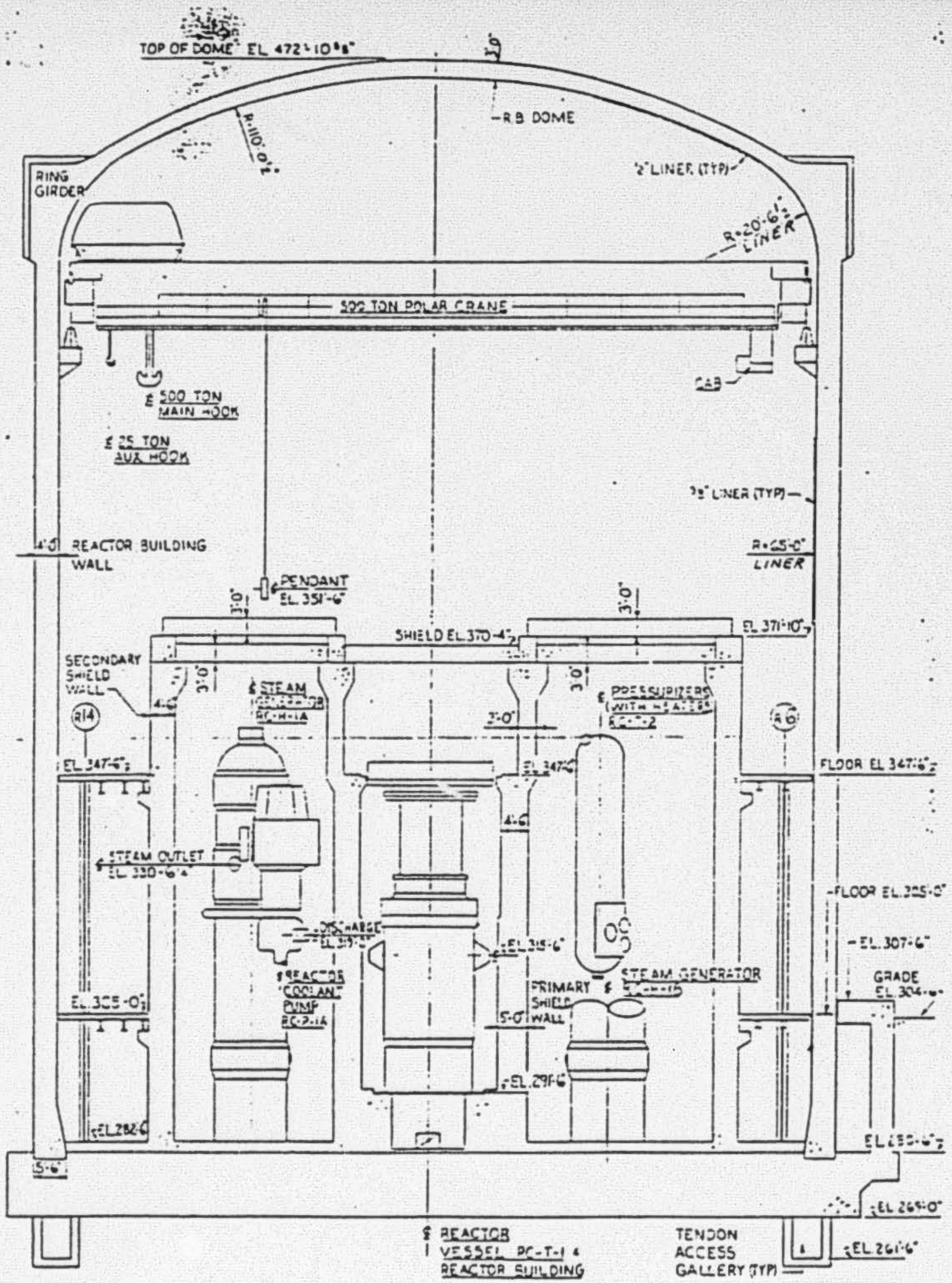
b. Rust or rust streaks on the Reactor Building shell due to rusting of ferrous material imbedded in the Reactor Building shell concrete.

c. Corrosion or peeling of paint on the Reactor Building liner and penetrations.

The following precautions will be observed before pressure testing the Reactor Building:

a. All light bulbs shall be removed.
b. All smoke detectors shall be disconnected electrically.
c. For the polar crane; all control panels, switch box doors, gear care inspection covers, test box lid, pendant housing, master switch covers, and similar light duty enclosures shall be opened.

198 034



CONTAINMENT STRUCTURE

FIGURE 1

198 035

TMI DOCUMENTS

DOCUMENT NO: TMI-C 3C.J

COPY MADE ON 5/8/79 OF DOCUMENT PROVIDED BY

METROPOLITAN EDISON COMPANY.

Wilda R. Mullinix, NRC

7906140407

198 019