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NUCLEAR PRODUCTION

TELEPHONE
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January 6, 1983

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

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

In order to facilitate the completion of the review of the Catawba FSAR, Duke Power Company is transmitting herewith responses or revised responses to open items of the following technical review branches.

Attachment 1 - Mechanical Engineering
Attachment 2 - HGEB/GSB
Attachment 3 - Auxiliary Systems
Attachment 4 - Power Systems
Attachment 5 - Accident Evaluation
Attachment 6 - Containment Systems
Attachment 7 - Core Performance
Attachment 8 - Licensee Qualification
Attachment 9 - Quality Assurance

These responses will be included in FSAR Revision 7.

Very truly yours,

Hal B Tucker

Hal B. Tucker

ROS/php
Attachments

B001

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. P. K. Van Doorn
NRC Resident Inspector
Catawba Nuclear Station

Mr. Robert Guild, Esq.
Attorney-at-Law
P. O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135½ Devine Street
Columbia, South Carolina 29205

Mr. Jesse L. Riley
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28207

Mr. Henry A. Presler, Chairman
Charlotte-Mecklenburg Environmental Coalition
943 Henley Place
Charlotte, North Carolina 28207

Attachment 1

Mechanical Engineering Branch

Response to TMI Concerns

conditions at any level of coolant activity. The design of this system significantly reduces radiation exposures during sample collection under accident conditions.

In addition to the reactor coolant sample line, a containment atmosphere sample line will be routed to a new accident level sampling panel. The containment atmosphere sample will be obtained from the hydrogen analyzer sample lines. Sample line length is minimized and large radius bends are used to minimize plate out and provide an accurate effluent sample.

Procedures for collection and transport of reactor coolant, sump water, and containment air samples under post-accident conditions have been revised to incorporate actions to be taken to minimize radiation exposures. These procedures specify the preplanning to be performed as well as modifications and approvals required prior to sample collection. Samples can be collected within one hour in all instances where personnel exposure does not exceed 3 rem/qtr whole body and 18 3/4 rem/qtr to the extremities. If the predicted personnel exposures exceed the above, samples can still be collected and analyzed but not within one hour. The time required to install additional shielding and allow sample collection while minimizing personnel exposure will be dependent upon the nature of the event. The analytical procedures have been reviewed and determined to be adequate for the expected sample activity levels.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Duke has modified the Catawba training program in order to place increased emphasis on the operation and significance of any Catawba systems or instrumentation which could be used to monitor and control accidents in which the core may be severely damaged. This additional training identified the vital instrumentation which supplies the operator with needed information in a degraded core situation. The training also identifies alternate methods of obtaining this information as well as specific instruction in the interpretation of instrument readings in degraded core situations.

II.D.1 RELIEF AND SAFETY VALVE TEST REQUIREMENTS

EPRI PWR Safety and Relief Valve Test Program will be used by Duke to respond to NRC recommendations in NUREG-0737. The Catawba valves covered by the EPRI program are pressurizer safety valves (Dresser type 6-31749A), pressurizer PORV (Control Components, Inc.), and PORV block valves (Rockwell Equiwedge gate valve). Testing is complete for the three types of valves. Duke has received complete EPRI reports covering the PORV's and block valves but has not received all test data for the safety valve tests. Review of EPRI test data should be complete by March 15, 1983.

the system for dead load analysis. Static or dynamic equilibrium equations are formulated using the direct stiffness method, in which element stiffness matrices are formed according to virtual work principles and assembled to form a global stiffness matrix for the system, relating external forces and moment to joint displacements and rotations. Appropriate stiffness modifications for curved components are included. Diagonal mass and damping matrices are assumed.

Static equilibrium equations are solved using Gaussian reduction techniques on the global stiffness matrix. For dynamic problems, the equilibrium equations may be solved using either step-by-step direct integration of the coupled equations of motion, or by first calculating natural frequencies and mode shapes and transforming the system into a set of uncoupled equations of motion. For seismic analysis of piping systems the latter approach is typically used in the dynamic analysis technique known as the response spectrum mode superposition method. In this technique, the earthquake excitation is characterized by acceleration response spectra, and the total response of the system is evaluated as a combination of the individual responses of the significant natural modes of vibration of the system. Natural frequencies and mode shapes are calculated using the determinant search technique. The method of combination of model responses can be selected from any one of those specified in Regulatory Guide 1.92. Earthquakes acting in all three directions simultaneously may be computed.

SUPERPIPE has been thoroughly verified for a comprehensive set of sample problems. This has included bench marking by EDS against the ASME Sample Problems 1 and 6 contained in ASME publication "Pressure Vessel and Piping 1972, Computer Program Verification", and against a Class 1 sample problem contained in ASME publication "Sample Analysis of a Piping System, Class 1 Nuclear", 1972. Extensive bench marking has also been performed by EDS against the programs, PISOL1A and PISOL3A which are well recognized and utilized throughout the industry. Additionally, the program has been bench marked by EDS against the programs such as NUPIPE, ADLPIPE, PIPESD and EDSGAP. SUPERPIPE has been used on a number of domestic and foreign plants. These include South Texas, McGuire 1, and San Onofre 1 and 2 (United States); Tihange 2 (Belgium); Kernkraftwerk Kruemmel, and Kernkraftwerk Phillipsburg (Germany); Kerndraftwerk Iran (Iran); Almarax, Cofrentes and Valdecaballeros (Spain); and, Leibstadt (Switzerland).

MEB Item 63 | Additional descriptions of computer codes used and their verification is provided in Table 3.9.1-6.

3.9.1.2.3 Programs Supplied by NSSS Vendor

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. These are described and verified in References 1 and 2.

1. WESTDYN-7 - static and dynamic analysis of redundant piping systems.

2. FIXFM - time-history displacement response of three-dimensional structures.

3. WESDYN-2 - piping system stress analysis from time histroy displacement data.

Table 3.9.1-6 (Page 1)
Computer Programs Used in Analysis

Application: ANSYS

A. Author: Swanson Analysis Systems, Inc.
P. O. Box 65
Houston, Pennsylvania 15342

Source: Control Data Corporation

Version: Revision 3

Facility: CDC Rockville

B. Description: Large-scale finite-element program for structural, heat transfer and fluid-flow analysis. ANSYS performs linear and nonlinear elastic analysis of structures subjected to static loads (pressure, temperature, concentrated forces and prescribed displacements) and dynamic excitations (transient and harmonic). The program considers the effects of plasticity, creep, swelling and large deformations. Transient and steady-state heat transfer analyses consider conduction, convection and radiation effects. Coupled thermal-fluid, coupled thermal-electric and wave-motion analysis capabilities are available. Structural and heat transfer analyses can be made in one, two or three dimensions, including axisymmetric and plane problems.

Extent and Limitation of its application: The ANSYS computer program is used to perform static elastic finite element analysis on pipe supports and welded attachments to piping.

C. Verification: The Control Data Corporation verified this version of the ANSYS program by a comparison of one hundred twenty six test problems with published analytical results.

Table 3.9.1-6 (Page 2)
Computer Programs Used in Analysis

Application: ANSYS

A. Author: Swanson Analysis Systems, Inc.
P. O. Box 65
Houston, Pennsylvania 15342

Source: University Computing Company

Version: Revision 3 and 4

Facility: UCC Dallas

B. Description: Large scale finite-element program for structural, heat transfer and fluid-flow analysis. ANSYS performs linear and nonlinear elastic analysis of structures subjected to static loads (pressure, temperature, concentrated forces and prescribed displacements) and dynamic excitations (transient and harmonic). The program considers the effects of plasticity, creep, swelling and large deformations. Transient and steady-state heat transfer analyses consider conduction, convection and radiation effects. Coupled thermal-fluid, coupled thermal-electric and wave-motion analysis capabilities are available. Structural and heat transfer analyses can be made in one, two or three dimensions, including axisymmetric and plane problems.

Extent and Limitation of its application: The ANSYS computer program is used to perform static elastic finite element analysis on pipe supports and welded attachments to piping.

C. Verification: The University of Computing Company verified this version of the ANSYS program by a comparison of hand calculations and analytical results published in literature.

Table 3.9.1-6 (Page 3)
Computer Programs Used in Analysis

Application: STARDYNE

A. Author: STARDYNE Project Office
System Development Corporation
2500 Colorado Avenue
Santa Monica, California 90406

Source: Control Data Corporation

Version: Level 03/01/81 and Level 02/04/82

Facility: CDC Rockville

B. Description: Finite element static and dynamic structural analysis. A STARDYNE static analysis will predict the stresses and deflections resulting from pressure, temperature, concentrated forces and enforced displacements. Dynamic analysis will predict the node displacements, velocities, accelerations, element forces and stresses from transient, harmonic, random or shock excitations. STARDYNE is user oriented, containing automatic node and element generation features that reduce the effort required to generate input. Plots of the original model and deformed structural shapes help the user evaluate results. Contour plots show surface stress for two-dimensional elements. The program creates time histories of element forces and stresses, and of node displacements, velocities, and accelerations.

Extent and Limitation of its application: The STARDYNE computer program is used to perform static elastic finite element analysis on pipe supports and welded attachments to piping.

C. Verification: The Control Data Corporation verified the computer program using the following methods:

Level 03/01/81 Three tests problems used in verification which have an MEB acceptable similar program; two tests problems have hand calculations; and one test problem has published analytical results.

Level 02/04/82 Three test problems used in verification which have an MEB acceptable similar program; two test problems have hand calculations; and 33 test problems have published analytical results.

Table 3.9.1-6 (Page 4)
Computer Programs Used in Analysis

Application: UNISTRUC II

A. Author: Jack Washham
Control Data Corporation
Application Resource Center
4201 Lexington Avenue North
Arden Hills, Minn. 55112

Source: Control Data Corporation

Version: 09/04/81

Facility: CDC Rockville and Duke Power CDC

B. Description: The UNISTRUC program is an interactive graphic pre- and post-processor used to reduce time needed for structural design and analysis. With UNISTRUC an engineer can quickly and automatically generate a finite element model at an interactive graphics terminal (instead of hand coding), check the model visually and submit it to one or a number of finite element applications for analysis. Results can be scanned for data needed to interpret the solution; modifications, if needed, can be submitted immediately.

Extent and Limitation of its application: The UNISTRUC program is used as a preprocessor/postprocessor for the finite element programs ANSYS and STARDYNE.

C. Verification: The Control Data Corporation verified the UNISTRUC program. This application is a pre- and post-processor for finite element applications NASTRAN, ANSYS, STARDNYNE, EASE2, and GTSTRUDL. The testing of UNISTRUC involves translating test problems into each of these five applications' format and then spot checking the results of each of the five applications against the others.

Application: GENERIC

A. Author: Teledyne Engineering Services
303 Bear Hill Road
Waltham, Massachusetts 02154

Source: Teledyne Engineering Services

Version: Revision B June 11, 1979

Facility: UCC Dallas

B. Description: The purpose of the GENERIC program is to generate a finite element model of a typical steel baseplate which is secured to a concrete slab through the use of anchor bolts. The GENERIC program is a preprocessor/ postprocessor to the ANSYS structural analysis program. The preprocessor performs the finite element generation and the postprocessor computes and tabulates anchor bolt loads, maximum plate deflections, loads in the concrete elements and shear elements, as well as the average bending stresses across the length and width of the plate.

Extent and Limitation of its application: The GENERIC program is used to analyze typical base-plate designs.

C. Verification: Teledyne performed the verification of the GENERIC program. Both experimental and analytical work was performed in this generic program. Shear-tension interaction test and cyclic test of concrete expansion anchors was performed and a pre and post processor to an existing finite element program were developed to facilitate baseplate analysis.

Table 3.9.1-6 (Page 6)
Computer Programs Used in Analysis

Application: PSBASEPLATE

A. Author: Jeff Swanson
Design Associates International
4105 Lexington Avenue North
Arden Hills, Minn. 55112

Source: Control Data Corporation

Version: 2.0

Facility: CDC Rockville

B. Description: The program PSBASEPLATE is a preprocessor/postprocessor to the STARDYNE computer code for the specific purpose of analyzing flexible baseplates.

Extent and Limitation of its application: The PSBASEPLATE program is used to analyze support base plates.

C. Verification: The Control Data Corporation verified PSBASEPLATE by comparing ten test problems with hand calculations and by running nine additional test problems used to check diagnostics. Duke Power verified PSBASEPLATE by comparing one test problem with the program GENERIC I. The Duke Power test problem included plate, frame, stiffners, anchor bolts, and a ground weld.

Table 3.9.1-6 (Page 7)
Computer Programs Used in Analysis

Application: BASEPLATE II

A. Author: Richard S. Holland
Ernst, Armand, and Botti Associates, Inc.
60 Hickory Drive
Waltham, Massachusetts 02154

Source: Control Data Corporation

Version: 1.0

Facility: CDC Rockville

B. Description: The program BASEPLATE II is a preprocessor/postprocessor to the STARDYNE computer code for the specific purpose of analyzing flexible baseplates.

Extent and Limitation of its application: The BASEPLATE II program is used to analyze support baseplates.

C. Verification: The Control Data Corporation verified BASEPLATE II by running six test problems to ensure the correct generation of the STARDYNE input data and correct interpretation of STARDYNE output data by comparison to STARDYNE data format. Duke Power verified the program by comparing one test problem with the program GENERIC I. The Duke Power test problem included plate, frame, stiffness, anchor bolts, and a ground weld.

Table 3.9.1-6 (Page 8)
Computer Programs Used in Analysis

Application: RELAP IV

A. Author: Aerojet Nuclear Company
P. O. Box 1625
Idaho Falls, Idaho 83415

Source: Control Data Corporation

Version: Mod 5 Revision 4.504

Facility: CDC Rockville

B. Description: Reactor Loss-of-Coolant Accident Program. Calculates one-dimensional, unsteady multiphase flow in complex pipe networks. Developed for analyzing large-break loss-of-coolant accidents, RELAP IV simulates the effects of hardware, such as pumps and valves, non-adiabatic surfaces, and the reactor core on the fluid behavior.

Extent and Limitation of its application: The RELAP IV computer program is used to develop pressure and temperature conditions in the Auxiliary Building as a result of different pipe rupture events. Also, the program is used to determine flow rates of several pipes to calculate flood levels in the Auxiliary and Reactor Buildings.

C. Verification: The Control Data Corporation verified RELAP IV by comparing test problems with those supplied in published literature.

Table 3.9.1-6 (Page 9)
Computer Programs Used in Analysis

Application: RELAP IV

A. Author: Aerojet Nuclear Company
P. O. Box 1625
Idaho Falls, Idaho 83415

Source: University Computing Company

Version: Mod 5 Revision

Facility: UCC Dallas

B. Description: Reactor Loss-of-Coolant Accident Program. Calculates one-dimensional, unsteady multiphase flow in complex pipe networks. Developed for analyzing large-break loss-of-coolant accidents, RELAP IV simulates the effects of hardware, such as pumps and valves, non-adiabatic surfaces, and the reactor core on the fluid behavior.

Extent and Limitation of its application: The RELAP IV computer program is used to develop pressure and temperature conditions in the Auxiliary Building as a result of different pipe rupture events. Also, the program is used to determine flow rates of several pipes to calculate flood levels in the Auxiliary and Reactor Buildings.

C. Verification: The University of Computing Company verified RELAP IV by a comparison of hand calculations and analytical results published in literature.

Table 3.9.1-6 (Page 10)
Computer Programs Used in Analysis

Application: RELAP V

A. <u>Author</u>: EG & G Idaho, Inc.
P. O. Box 1625
Idaho Falls, Idaho 83415

<u>Source</u>: Control Data Corporation

<u>Version</u>: 2.10

<u>Facility</u>: CDC Rockville

B. <u>Description</u>: Reactor Transient Analysis Program. An advanced thermal-hydraulics program for analyzing complex transients in nuclear reactors and piping network. One-dimensional methodology calculates unsteady steam and/or water flow. The equations can calculate non-homogeneous, non-equilibrium conditions between steam and liquid phases. Models simulate hardware such as pumps and valves, non-adiabatic walls, and reactor control systems.

<u>Extent and Limitation</u> of its application: The computer program RELAP V is used to calculate piping system transient forces resulting from a postulated pipe rupture, valve actuation, or pump transients.

C. <u>Verification</u>: The Control Data Corporation verified the program by test results compared to those supplied by INEL (EG & G Idaho, Inc.) the author; in addition, the Edward's Pipe Problem compares RELAP V data to experimental data.

Table 3.9.1-6 (Page 11)
Computer Programs Used in Analysis

Application: PRTHRUST/PIPERUP

A. Author: QUADREX Corporation
1700 Dell Avenue
Kandell, California 95008

Source: Control Data Corporation

Version: 1.3.1

Facility: CDC Rockville

B. Description: Nonlinear analysis for piping systems subjected to postulated ruptures. PRTHRUST calculates blowdown forces for both longitudinal and circumferential breaks and provides for modeling of all major components. PIPERUP analyzes an elastic-plastic solution while considering the effects of strain hardening, gaps and pipewhip restraints.

Extent and Limitation of its application: The computer program PRTHRUST is used to calculate piping system transient forces resulting from one of the following: Postulated Pipe Rupture; Control, Relief, or Stop Valve Actuation; or Pump Transients. The computer program PIPERUP is used to perform elastic plastic analysis of piping systems subjected to concentrated static or dynamic time history forcing functions.

C. Verification: The Quadrex corporation performed the verification of PRTHRUST/PIPERUP. The Control Data Corporation ran two test problems to check the workability of the program.

Table 3.9.1-6 (Page 12)
Computer Programs Used in Analysis

Application: PRTHRUST/PIPEUP

A. Author: QUADREX Corporation
1700 Dell Avenue
Kandell, California 95008

Source: University Computing Company

Version: 1.3.1

Facility: UCC Dallas

B. Description: Nonlinear analysis for piping systems subjected to postulated ruptures. PRTHRUST calculates blowdown forces for both longitudinal and circumferential breaks and provides for modeling of all major components. PIPERUP analyzes an elastic-plastic solution while considering the effects of strain hardening, gaps and pipewhip restraints.

Extent and Limitations of its application: The computer program PRTHRUST is used to calculate piping system transient forces resulting from one of the following: Postulated Pipe Rupture; Control, Relief, or Stop Valve Actuation; or Pump Transients. The computer program PIPERUP is used to perform elastic plastic analysis of piping systems subjected to concentrated static or dynamic time history forcing functions.

C. Verification: The Quadrex corporation performed the verification of PRTHRUST/PIPERUP. University Computing Company will perform on their own verification in early 1983.

Table 3.9.1-6 (Page 13)
Computer Programs Used in Analysis

Application: SUPERPIPE

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: February 28, 1978; May 15, 1978;
July 1, 1978; October 1, 1978;
November 15, 1979; January 31, 1982;
and June 28, 1982

Facility: EDS Nuclear, Inc., CDC Cybernet Twin Cities,
UCC Dallas, Duke Power
CDC, and Duke Power - IBM

B. Description: SUPERPIPE is a computer program for the structural analysis and code compliance evaluation of piping systems, with particular emphasis on Class 1, 2, and 3 nuclear power piping designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

The piping system may be modeled by straight pipes, curved pipes, elbows, reducers, tees, branch connections, and other commonly-used piping attachments.

Principal features of the program include the following:

i) Static and thermal stress analysis by the direct stiffness method.
ii) Frequencies and mode-shapes computation, using the subspace interaction method or the Q-R method.
iii) Dynamic response analysis by the modal-superposition time-history analysis, the direct-integration time-history analysis or the response spectrum analysis.
iv) Seismic analysis of multiple-excited piping systems by the "multiple-response-spectrum approach."
v) Combination of directional and modal responses according to USNRC Regulatory Guide 1.92.
vi) Code compliance evaluation according to ASME Boiler and Pressure Vessel Code, Section III, for Class 1, 2, and 3 nuclear piping (user's choice of code addendum).
vii) Determination of pipe-break locations.
viii) Fatigue damage computations.
ix) Built-in library of standard material properties, cross-sectional properties, flexibility factors, and stress intensification factors.
x) Restart options to store, recall, and modify piping geometry, and to perform analysis and code compliance evaluation in stages.
xi) Options to print-out support loads and displacement summaries, nozzle and penetration load summaries, and code compliance summary.

xii) Options to plot piping geometry and mode shapes.

Extent and Limitation of its application: All routines of the SUPERPIPE program are used as detailed in the description.

C. Verification: The program has been bench-marked against the EDS program PISOL and foreign programs like NUPIPE and PIPESD.

This program has been verified by bench-marking to an ASME sample problem, by comparison to detailed analysis performed manually, by comparison to results achieved using similar programs, as described above, and by comparison to results achieved using the previous version of SUPERPIPE. The bench-mark problems specified in NUREG CR-1677 have been evaluated using this program and the results have been transmitted to the NRC.

Application: ANSYS

A. Author: Swanson Analysis Systems, Inc.
P. O. Box 65
Houston, Pennsylvania 15342

Source: Swanson Analysis Systems

Version: Revision 3

Facility: EDS Nuclear, Inc. and CDC Cybernet Twin Cities

B. Description: ANSYS is a general purpose program for structural, heat-transfer and fluid-flow analysis. In structural analysis, the program can consider static and dynamic; elastic, plastic, creep and swelling small and large deflection conditions. In the heat-transfer or fluid-flow analysis, it can consider linear and nonlinear; steady-state and transient conditions. The program has the capability of analyzing piping systems, two-dimensional axisymmetric solids, three-dimensional solids and axisymmetric and three-dimensional shells.

Extent and Limitation of its application: ANSYS is used to perform finite element static elastic analysis on piping components to obtain stress levels for NB 3600.

C. Verification: Verification consists of comparing program calculated solutions with 1) known theoretical solutions, 2) experimental results, and 3) other calculated solutions. The readily available theoretical solutions used for comparison to the program solutions are found in numerous published documents. Several of the documents are listed below:

1) "Strength of Materials, Part I, Elementary Theory and Problems" by S. Timoshenko.

2) "Strength of Materials Part II, Advanced Theory and Problems" by S. Timoshenko.

3) "An Introduction to the Mechanics of Solids" by S. H. Crandall and N. C. Dahl.

4) "Formulas for Stress and Strain" by R. J. Roark.

Table 3.9.1-6 (Page 16)
Computer Programs Used in Analysis

Application: EDSSAAS

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94101

Source: SAASII/EDS Nuclear, Inc.

Version: August 31, 1979

Facility: CDC Cybernet Twin Cities and EDS Nuclear, Inc.

B. Description: EDSSAAS is a finite element program designed for the stress analysis of axisymmetric solids subjected to axisymmetric load. It is specifically useful for the stress analysis of components (nozzles, valves, etc.). The program is a modified version of SAASII, developed by the Aerospace Corporation, San Bernardino, California.

Extent and Limitations of its applications: EDSSAAS is used to perform finite element static elastic analysis on piping components to obtain stress levels for NB 3600.

C. Verification: The program is currently verified to handle elastic analyses of soilds modeled with quadrilateral and triangular 2-D elements with orthotropic temperature dependent material properties. The axisymmetric loading can be mechanical (pressure and discrete nodal loads) and/or thermal (nodal temperatures).

The program has been verified by running test cases using SAASII and comparing the results. This version of EDSSAAS was verified by rerunning problems used for the verification of the previous version (v. 7/26/77) and comparing results.

Table 3.9.1-6 (Page 17)
Computer Programs Used in Analysis

Application: SPECT1A

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: January 20, 1978

Facility: EDS Nuclear, Inc. and CDC Cybernet
Twin Cities

B. Description: SPECT1A modifies and combines period or frequency based digitized seismic response spectra and produces punched card decks for subsequent use in SUPERPIPE. Punched card sets of time-value/acceleration pairs may be linearized, enveloped, averaged, and spread using various options.

Extent and Limitations of its application: All routines of the SPECT1A program are used as detailed in the description.

C. Verification: The purpose of this program is data manipulation only. No calculations as such are performed. The program has therefore been verified by comparing the output to data manipulated manually.

Table 3.9.1-6 (Page 18)
Computer Programs Used in Analysis

Application: TRANS2A

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: March 15, 1977

Facility: EDS Nuclear, Inc. and CDC Cybernet Twin Cities

B. Description: TRANS2A is a computer program which determines radial temperature distributions and gradient in a pipe wall experiencing fluid temperature excursions. TRANS2A determines these temperature distributions and gradients by solution of the unsteady one-dimensional axisymmetric heat transfer equation.

Extent and Limitation of its application: All routines of the TRANS2A program are used as detailed in the description.

C. Verification: TRANS2A has been extensively tested and compared with independent results for sample problems. TRANS2A temperature distributions agree favorably with calculations using TRANS1A and the EDS proprietary finite-element program TAPAS. Calculations for thermal gradient terms ΔT_1 and ΔT_2 compare favorably with results from TRANS1A and with the values derived from charts published by McNeill and Brock in "Engineering Data File - Charts for Transient Temperature in Pipes," 1971.

Application: EDSMESH

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: August 31, 1979

Facility: CDC Cybernet Twin Cities and EDS Nuclear, Inc.

B. Description: EDSMESH is a general purpose, two dimensional finite element mesh generator. With minimal user input, the program generates a finite element mesh, plots the mesh and writes the mesh information on tape for use in finite element analysis programs. It is specifically designed for use with EDS Component Analysis Package (GAP1) consisting of EDSSAAS, EDSASAAS and EDSPOT.

Extent and Limitation of its application: All routines of the EDSMESH program are used as detailed in the description.

C. Verification: This program has been verified by developing sample finite element meshes with the program and comparing these models to previously generated finite element meshes.

Application: EDSLIN

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: August 1, 1980

Facility: CDC Cybernet Twin Cities and EDS Nuclear, Inc.

B. Description: EDSLIN is a post processor to finite element stress and thermal analyses of axisymmetric solids (components), it linearizes stress and/or temperature profiles across user-specified cuts through the thickness of the component, for compatibility with the ASME Section III NB-3000 code definitions.

The purpose underlying the program is the correlation of finite element results (stress/temperature) to the ASME Code stress categories (NB-3200) and temperature terms (NB-3650).

Extent and Limitations of its application: All routines of the EDSLIN program are used as detailed in the description.

C. Verification: The program has been verified by comparison to detailed analysis performed manually. The manually performed analysis consisted of:

(1) determination of stress/temperature at each internal point along the section

(2) determination of average temperature and linear temperature distrubution across the section.

(3) determination of principal stresses and stress intensities for membrane, linear and total stresses.

Application: EDSFLOW

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: RELAP IV Mod 5 / EDS Nuclear

Version: 1.01 February 12, 1980

Facility: CDC Cybernet Twin Cities

B. Description: EDSFLOW is a computer program for the analysis of the thermal/hydraulic behavior of light water reactor systems subjected to postulated transients such as those resulting from loss of coolant, pump failure, or rapid depressurization. It is a modified version of the RELAP4/MOD5 program developed for NRC by the Idaho National Engineering Laboratory.

Extent and Limitation of its application: EDSFLOW is used primarily for time-history analysis of hydraulic forces on piping systems during rapid transient (i.e. safety valve blowdown). It is also used for transient containment building subcompartment pressure/temperature analysis.

C. Verification: EDSFLOW has been verified by comparison to RELAP4/MOD5 thermal/hydraulic predictions for two pressurizer safety/relief valve discharge transients and one PWR main steam hammer analysis.

Application: FRCON4

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear

Version: February 8, 1978

Facility: CDC Cybernet Twin Cities

B. Description: FRCON4 is a computer program for the computation of the time-histories of the unbalanced forces during water and/or steam piping systems transients.

The program reads the thermal-hydraulic transient time-histories generated (on tape) by the EDSFLOW program and computes the corresponding force time-histories by the integral force equation technique. The calculated force time-histories are written on tape, in a format that is compatible with the input formats of the structural dynamic analysis programs EDSGAP and SUPERPIPE.

Extent and Limitation of its application: All routines of the FRCON4 program are used as detailed in the description.

C. Verification: The program has been verified by comparison to Moody test data for superheated steam discharge into a pipe. It has also been compared to hand calculations, tables and graphs published by Moody for other fluid conditions liquid and subcooled and saturated steam with regard to blowdown thrust forces.

Application: CATNAP

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: May 26, 1978

Facility: EDS Nuclear, Inc. and
CDC Cybernet Twin Cities

B. Description: CATNAP is a special-purpose program which evaluates load combinations for defined service conditions induced on a nozzle due to the stiffness of the attached piping. Pipe stiffness, RCL and RPV seismic displacements, LOCA displacements, seismic inertia moments, and gravity moments are input to the program. Design condition and faulted condition evaluations are performed at the centerline intersection of the RCL and auxiliary pipe; seismic primary-plus-secondary stress (i.e., fatigue), and thermal expansion evaluations are performed at the auxiliary line safe-end-weld. "Worst-case" resultant moments conditions are obtained through evaluation of all possible permutations of the sign combination of unsigned and conditionally signed displacements and rotations in combination with signed moments. Output from the program summarizes actual worst-case resultant moment and ratio to allowable for the defined service conditions.

Extent and Limitations of its application: All routines of the CATNAP program are used as detailed in the description.

C. Verification: The purpose of this program is data manipulation only. No calculations as such are performed. The program has therefore been verified by comparing the output to data manipulated manually.

Application: SIMPWHIP

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: August 15, 1979

Facility: EDS Nuclear, Inc. and UCC Dallas

B. Description: SIMPWHIP is a computer program for the simplified time-history analysis of pipe whip problems. The program provides a conservative estimate of the restraint forces and deformations, during the initial impact phase up to the time the piping system initially stops after impact with the restraints. SIMPWHIP can handle both circumferential and longitudinal pipe rupture events. The program assumes the motions of the system to be two-dimensional and in the small-deflection range.

Extent and Limitations of its application: All routines of the SIMPWHIP program are used as detailed in the description.

C. Verification: The program has been verified by comparison with the results of the hand calculated problem.

Application: PWHIP

A. Author: EDS Nuclear, Inc.
220 Montgomery Street
San Francisco, California 94104

Source: EDS Nuclear, Inc.

Version: October 20, 1977

Facility: EDS Nuclear, Inc.

B. Description: PWHIP is a computer program for the nonlinear dynamic response analysis of three-dimensional beam/piping systems restrained with energy-absorbing devices.

Principle features of the program include the following:

i) The finite element library consists of:
   a) Two different types of inelastic beam elements, differing in post-yield behavior.
   b) Energy-absorbing restraint elements (inelastic tension element - i.e., U-bar element, inelastic compression element, and friction disc element).

ii) Both inelastic material nonlinearities and geometric "gap" nonlinearities can be included in the analysis.

iii) Static loads may be applied to the structure prior to the application of the dynamic loads, provided the structural behavior is elastic during the static loading phase.

iv) Large deformation of the energy-absorbing devices are permitted; but the response of the main structure - beam/pipe elements - is restricted to small deformation.

v) Damping may be specified as mass-proportional or stiffness-proportional.

Extent and Limitation of its application: All routines of the PWHIP program are used as detailed in the description.

C. Verification: PWHIP has been verified by comparison analyses with closed form text book solutions, and solutions generated by the general finite program ANSYS.

Table 3.9.1-6 (Page 26)
Computer Programs Used in Analysis

Application: MCAUTO STRUDL

A. Author: McDonnell Douglas Automation Company
Box 516
St. Louis, Missouri

Source: MCAUTO

Version: Release 2.7 through 4.7

Facility: Duke Power

B. Description: Large scale general purpose finite element program for structural analysis.

Extent and Limitation of its application: MCAUTO STRUDL is used to perform static elastic analysis of pipe supports.

C. Verification: MCAUTO STRUDL has been verified by comparison of the results with either hand calculations, closed form solutions found in standard text books or solutions from other programs.

Table 3.9.1-6 (Page 27)
Computer Programs Used in Analysis

Application: ANSYS

A. Author: Swanson Analysis Systems, Inc.
P. O. Box 65
Houston, Pennsylvania 15342

Source: Swanson Analysis Systems, Inc.

Version: Revision 3 and 4

Facility: Duke Power

B. Description: Large-scale finite-element program for structural, heat transfer and fluid-flow analysis. ANSYS performs linear and nonlinear elastic analysis of structures subjected to static loads (pressure, temperature, concentrated forces and prescribed displacements) and dynamic excitations (transient and harmonic). The program considers the effects of plasticity, creep, swelling and large deformations. Transient and steady-state heat transfer analyses consider conduciton, convection and radiation effects. Coupled thermal-fluid, coupled thermal-electric and wave-motion analysis capabilities are available. Structural and heat transfer analyses can be made in one, two or three dimensions, including axisymmetric and plane problems.

Extent and Limitation of its application: The ANSYS computer program is used to perform static elastic finite element analysis on pipe supports and welded attachments to piping.

C. Verification: The ANSYS program has been verified by comparison with a series of test problems of a published analytical solutions.

Attachment 2

HGEB/GSB

VII in the region and the overburden amplification that contributed to those maximum observed surface intensities.

The historic earthquake which is assumed to be the largest earthquake to occur in the Piedmont is VII MM. This is larger than the historic maximum surface intensity at the site, which is VI-VII MM from the August 31, 1886 Charleston earthquake.

Q230.3 The NRC position allows the maximum earthquake determined for the seismotectonic province to coincide with the Safe Shutdown Earthquake. On this basis, the acceptable Safe Shutdown Earthquake for Catawba would be VII MM. (The Safe Shutdown Earthquake actually assumed for Catawba is VII-VIII MM, or 1/2 intensity unit higher than the maximum historical epicentral intensity experienced in the site seismotectonic region). For intensity VII MM, the "Trend of the Mean" relationship of the data of Trifunac and Brady (Reference 62) yields 0.12g.

The design response spectra are discussed in Section 2.5.2.8.

2.5.2.7 Operating Basis Earthquake

Appendix A of 10 CFR 100 provides that the Operating Basis Earthquake (OBE) shall be specified by the Applicant and shall be defined by response spectra. The OBE acceleration value recommended for foundations on closely jointed and slightly weathered rock is 0.08g. This is a conservative value, and is slightly more than one-half the SSE.

2.5.2.8 Design Response Spectra

Four synthetic earthquake records with maximum accelerations of 0.15g are developed to generate the response spectra for the Safe Shutdown Earthquake. In simulating the earthquakes, a maximum duration of 20 seconds is used in the model, of which 0 to 2 seconds is used for the rising period, 2 to 7 seconds for the constant acceleration period, and 7 to 20 seconds for the receding period. The shape of the response spectra of the simulated earthquakes for a single degree of freedom approximates the "Spectrum Curve" discussed by Newmark (Reference 64).

The numerical average of the response of the four earthquakes is used to generate the response spectra for 0.5, 1.0, 2.0 and 5.0 percent damping. The design response spectrum is a smoothed curve drawn through the averaged spectra. Figure 2.5.2-5 gives comparisons of the smoothed and the averaged spectra for 0.5, 1.0, 2.0 and 5.0 percent damping. Figures 2.5.2-6 and 2.5.2-7 give the smoothed response spectra for 0.5, 1.0, 2.0 and 5.0 percent for 0.08g and 0.15g ground acceleration, respectively.

Certain Category I tanks and other small structures are not founded on "continuous" rock as described in Section 2.5.4.10. These structures are designed using the design response spectra for 25 ft of soil (fill) above rock as shown on Figure 2.5.2-8. The spectra on Figure 2.5.2-8 are computed

using a lumped mass model; the mathematics for the method are described in standard texts such as Chapter 2 of Newmark and Rosenblueth (Reference 105). The soil parameters used in the lumped mass model are shear modulus, G, equals 576 KSF, and soil damping, D at 10 percent of critical. These parameters are equivalent to strain-corrected soil properties (G_{used} and D_{used}) obtained when the program SHAKE (Reference 106) and the equivalent linear method are used to perform amplification analyses.

The soil conditions at structures listed in Table 2.5.4-4 and for which the amplified design response spectra on Figure 2.5.2-8 are used for design are summarized on Figure 2.5.2-8A. As can be seen from Figure 2.5.2-8A, the site conditions are such that only the Unit 2 Diesel Fuel Oil Tanks and the Unit 2 Above Ground Storage Tank actually justify consideration of a design response spectrum other than the one for rock on Figure 2.5.2-7. All other structures are supported on fill concrete extending to rock, or on partially weathered rock sufficiently thin (less than about 20 ft deep) that amplification of the bedrock motion is not of engineering concern. (Sites having 20 ft or less of overburden that overlies rock or rock-like material whose shear wave velocity equals or exceeds 2500 ft per second are considered rock sites, References 107 and 108.)

For the two structures where conditions actually warrant consideration of the amplified spectrum on Figure 2.5.2-8, horizontal response spectra are calculated for the soil columns at the specific structure for comparison to the design spectrum. The program SHAKE is utilized for these calculations using the soil parameters shown on Figure 2.5.2-8B. The strain dependency of shear modulus and damping of all the soil materials in the soil columns is assumed to conform to the average curve for sand found in Seed and Idriss, 1970 (Reference 102). The strain dependency of the shear modulus of the compacted crushed stone backfill is calculated using information contained in Reference 109. The shear wave velocity (shear modulus) at low strain of the compacted crushed stone is computed from information contained in Reference 109. The damping in the crushed stone is assumed to be about 15 percent less than the average sand damping at the equivalent confining pressure and shear strain.

The numerical average of the response due to input of each of the four synthetic earthquakes at an exposure of the rock-like material is used to plot the response spectra for actual site conditions as shown on Figure 2.5.2-8C. The response spectrum for which the structures are designed is also shown on Figure 2.5.2-8C for comparison with the response spectra computed for actual site conditions.

The computed response spectra for the actual site conditions at the Unit 2 Diesel Fuel Oil Tanks is below the design response spectrum except for periods above 0.4 seconds. The fundamental period of these tanks is less than 0.4 seconds. The computed response spectrum for the Unit 2 Above Ground Storage Tank is below the design response spectrum for periods less than 0.7 seconds; the fundamental period of the Above Ground Storage Tank is less than 0.7 seconds. Thus, the design spectrum exceeds the computed spectra for actual site conditions within the period range of both these facilities and the use

of the design response spectrum (25 ft of overburden) is conservative for the design.

Table 2.5.4-4 lists the foundation design condition and the response spectrum used for each Category I structure.

2.5.3 SURFACE FAULTING

There is no geologic evidence of (capable) surface faulting in the Piedmont, the tectonic province in which the site is located. Therefore, a design basis for surface faulting is not applicable to this site.

2.5.3.1 Geologic Conditions of the Site

The geologic conditions in the region surrounding the site and of the site itself are discussed in Sections 2.5.1.1 and 2.5.1.2, respectively.

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95. Fisher, G. W., et al., Studies of Appalachian Geology: Central and Southern, Interscience Publishers, 1970, Tectonic Map.

96. Hunt, C. B., Physiography of the United States, W. H. Freeman and Co., San Francisco, 1967, pp 137-204, Figure 1.1.

97. Keith, A. and Sterrett, D. B., Description of the Gaffney and Kings Mountain Quadrangles, U. S. Geol. Survey Geologic Atlas of the U. S., Folio 222, 1931.

98. Law Engineering Testing Company, Geology of Cherokee Nuclear Station, Duke Power Company, 1974.

99. Stromquist, A. A., Choquette, P. W. and Sundelius, H. W., Geologic Map of the Denton Quadrangle, Central North Carolina, U. S. G. S. Map GQ-872, 1971.

100. Stuckey, J. L. and Conrad, S. G., Explanatory Text for Geologic Map of North Carolina, North Carolina Dept. of Cons. and Dev., Bull. 71, 1958, 51 pp.

Note: References 90 through 100 are not cited in the text.

101. Duke Power Company, Final Geologic Report on Brecciated Zones. March 1, 1976.

102. Seed, H. B. and Idriss, I. M., Soil Moduli and Damping Factors for Dynamic Response Analysis, Earthquake Engineering Research Center, Report No. EERC-70-10, December 1970.

103. Seed, H. B., Idriss, I. M., Makdisi, F., and Banerjee, N., Representation of Irregular Stress Time Histories by Equivalent Uniform Stress Series in Liquefaction Analyses, Report No. EERC-75-29, College of Engineering, University of California, Berkeley, October 1975.

104. Wroth, C. P., In Situ Measurement of Initial Stress and Deformation Characteristics, Session IV, ASCE Geotechnical Engineering Division Special Conference on In Situ Measurement of Soil Properties, Raleigh, North Carolina, June, pp. 1-14.

105. Newmark, N. M. and Rosenblueth, E., Fundamentals of Earthquake Engineering, Civil Engineering and Engineering Mechanics Series, Prentice-Hall, Inc. Englewood Cliffs, New Jersey, 1971.

106. Schnabel, P. B.; Lysmer, J.; and Seed, H. B., "SHAKE, A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites" Report No. EERC 72-12, December 1972, College of Engineering, Univ. of California, Berkeley, California.

107. Seed, H. B., Ugas, C. and Lysmer, J. (1974), "Site Development Spectra for Earthquake Resistant Design," Report No. EERC-74-12, College of Engineering, University of California, Berkeley, November, 1974.

108. Seed, H. B., Ugas, C. and Lysmer, Jr. (1976b), "Site Dependent Spectra for Earthquake Resistant Design," Bulletin of the Seismological Society of America, Vol. 66, pp. 221-244.

109. Hardin, B. O., "Shear Modulus of Gravels," UKY TR74-73-CE19, Soil Mechanics Series No. 16, University of Kentucky, College of Engineering Dept. of Civil Engineering, September, 1973.

Attachment 3

Auxiliary Systems Branch

410.6 (3.6.1) Provide justification and documentation for your position that non-liquid and non-water carrying piping systems over 200°F not be considered high energy systems. The information provided in FSAR Table 3.6.1-3 is insufficient justification. It is our position that all plant fluid systems which operate at temperatures exceeding 200°F or pressures exceeding 275 psig during normal plant conditions be considered as high energy systems and protection afforded for safety related systems from their failure. Provide the results of the necessary analysis to assure compliance with the guidance of Standard Review Plan (SRP) Section 3.6.1.

Response:

Refer to Section 3.6.1.1.2.

410.7 (3.6.1) Provide an analysis of the effects on safety-related systems of failures in all high- or moderate-energy piping systems in accordance with the applicable criteria of Standard Review Plan (SRP) Section 3.6.1 and identify the criteria you are applying to your plant design. Expand Tables 3.6.1-1 and 3.6.1-2 to correlate interactions between systems. For each safety-related system, indicate which high- and/or moderate-energy system can affect its safety function and identify the specific protective method provided for the safety-related system from the postulated failures. Include area layout drawings depicting the locations of failures affecting safety-related systems giving dimensions, locations, and protective method for each postulated break or crack in a high-or moderate-energy system. Include the assumptions used in your analysis such as flowrate through postulated cracks, room volumes, sump capacities, and floor drainage system capacities. Your analysis should verify the capability to sustain any postulated high energy piping system failure concurrent with any single active failure.

Response:

The information required to respond to this question was provided by letter dated March 9, 1982 from W. O. Parker, Jr. to H. R. Denton.

This analysis considered single active failures in accordance with BTP ASB 3-1.

The Auxiliary Steam System, which is seismically supported inside the Auxiliary Building, was treated as moderate energy piping in this analysis. Environmental qualification of equipment is discussed in Section 3.11.

Recommendation GS-4

Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place.

Response

Transfer of the auxiliary feedwater supply from the normal to the safety grade assured supply occurs automatically. The instrumentation and controls utilized in the switchover logic are safety grade.

Recommendation GS-5

The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train, independent of any AC power source.

Response

The auxiliary feedwater system at Catawba is capable of automatic initiation and of providing the required flow for 2 hours independent of any AC power source. This is accomplished by means of the turbine-driven auxiliary feedwater pump and DC powered instrumentation and controls.

Recommendation GS-6

The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

(1) Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

(2) The licensee should propose Technical Specifications to assure that, prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

Response

(1) Procedures used to manipulate valves which are important to safety of the auxiliary feedwater system require that such valves be manipulated by one operator and independently verified by another operator. (See response to TMI Item I.C.6 in Table 1.9-1).

(2) Prior to unit startup following any cold shutdown of 30 days or longer and at least once per refueling cycle, the CA System is given either a manual or an automatic initiation signal in order to verify the normal flow path. This commitment will be included in the Technical Specifications.

Recommendation GS-7

The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade.

Response

The Catawba auxiliary feedwater system employs safety-grade automatic initiation signals and circuits. Automatic initiation of the system is discussed in Section 10.4.9.2.

Recommendation GS-8

The licensee should install a system to automatically initiate AFW system flow.

Response

See response to Recommendation GS-7.

Additional Short-Term Recommendations

Recommendation

The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

Response

As noted in the response in Recommendation GS-4, the Catawba design utilizes an automatic transfer of the auxiliary feedwater supply to the assured supply, the Nuclear Service Water System. Redundant level instrumentation is provided as discussed in Section 9.2.1.5.5. In addition to this, single channel, non-safety-grade level indication and low level alarms are provided in the control room for each of the normal auxiliary feedwater sources (upper surge tank, condenser hotwell and auxiliary feedwater condensate storage tank).

Recommendation

The licensee should perform a 72 hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72 hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

Attachment 4

Power Systems Branch

1. Maintenance circuit: This circuit consists of phone jacks located throughout the plant which can be patched together to establish communication between areas as necessary.

2. Refueling circuit: This circuit consists of sound-powered phone stations connecting areas required for refueling operations.

3. Emergency circuit: This circuit consists of sound-powered phone stations connecting the auxiliary shutdown panel with areas of the plant that may require local operation during an emergency shutdown.

Q430.54 The sound powered telephone systems are powered from the diesel backed emergency AC lighting panelboards located in their respective areas. The locations of the emergency sound-powered telephone stations are indicated on Figures 9.5.2-1 through 9.5.2-16.

9.5.2.2.4 Emergency Offsite Communication

Emergency offsite communication independent of the PABX system is provided by public telephone lines and Duke microwave lines connected directly to specific telephones in critical areas of the station. Emergency telephones are color coded to distinguish them from the intraplant telephones. The locations of emergency public and microwave telephones are indicated on Figures 9.5.2-1 through 9.5.2-16. Additionally, a security radio system is provided in accordance with 10CFR73.55(f), and a crisis management radio system is provided in accordance with NUREG 0654.

Q430.54 Emergency Notification System (ENS) "red phones" are located in the control room, Technical Support Center, and Crisis Management Center. These phones provide a communications link with the NRC-Bethesda and NRC-Region II. The ENS is independent of the PABX, using lines connected to AT&T Long Lines Division. This system is powered from a non-essential, battery backed bus.

Health Physics Network (HPN) phones are located in the Technical Support Center and Crisis Management Center. These phones provide a communications link with the NRC's health physics personnel. This system is tied to the Bell System and receives power from a non-essential, battery backed bus.

Ringdown phones are located in the control room, Technical Support Center, and Crisis Management Center and provide silmutaneous communications with Mecklenburg, Gaston, and York Counties. The system uses local Bell lines and receives power from a non-essential, battery backed bus. A separate system of ringdown phones is provided for communications with North Carolina and South Carolina.

A radio system provides an independent communications link to the States, counties, General Office, and field monitoring teams. A base station radio is located in the control room; remote units are located in the Technical Support Center, Crisis Management Center and Duke Power General Office in Charlotte, N. C. The counties and States have portable radios with fixed antenna exterior to the building. Each radio is powered by a non-essential AC source and has a built in battery backup.

9.5.2.2.5 Power Supply Separation

The Intraplant Telephone System (PABX) is powered from three (3) battery backed, 120VAC, 1 Ø power panelboards that are located in the Auxiliary Building and are independent of each other. These panelboards are 1KXPA, 1KXPB, and 2KXPA and feed the switch via cables 1ECI541, 1ECI542, and 1ECI585 respectfully. The minimum separation distance between panelboards is 25 feet. Cables 1ECI541 and 1ECI542 are run in the same cable tray from the Auxiliary Building to the Telephone Equipment Room in the Administration Building, however, Cable 1ECI585 is run in a separate tray system in excess of 20 feet from the 1ECI541 and 1ECI542 cables to a point in the Service Building, El. 594'-0" at Column U-7, where all three cables enter a final tray run to the Telephone Equipment Room.

The Intraplant Public Address (PA) System is powered from two (2) shared Motor Control Centers that are located independent of each other. Motor Control Center SMXC is located in the Auxiliary Building and feeds PA Power Panelboard PAP-1 also located in the Auxiliary Building. Motor Control Center SMXS is located in the Service Building and feeds PA Power Panelboard PAP-2 also located in the Service Building. Therefore, the PA is fed from two (2) sources located in different buildings.

The Emergency Sound-Power Telephone System is powered on a per Unit basis, via a double feed circuit through a power transfer contactor. For Unit 1, the system is fed from diesel-backed Emergency Lighting Panelboard 1ELB1 and Normal Lighting Panelboard 1LA6. Panelboard 1ELB1 is located in the Auxiliary Building, EL. 560'-0". Panelboard 1LA6 is located in the Auxiliary Building, El. 543'-0", thus separation is one floor elevation (17 feet). Unit 2 is similarly installed. The cabling running from 1ELB1 and 1LA6 to the Power Transfer Contactor is 2/C #12ALS. Q430.54

See Figures 9.5.2-17 and 9.5.2-18 for one-line representations of the telephone systems.

9.5.2.3 <u>Communication During Transient and Accident Conditions</u>

In order to achieve a safe cold shutdown, it may be necessary for plant personnel to communicate with the control room or the emergency shutdown panel from selected working stations. These work stations and the communication systems available at each station are identified in Tables 9.5.2-1 and -2. The types and locations of these communication devices/stations are indicated on Figures 9.5.2-1 through 9.5.2-16.

The emergency sound-powered telephone system is the means of communication intended for use during accident conditions. Effective communication is provided by the emergency sound-powered phones in background noise levels as high as 110 dBA. PABX handsets are also available at all of the subject work stations and can be effectively used in noise levels or approximately 90 to 95 dBA.

Hand held radio transceivers are provided for use by station personnel in routine and emergency situations. A fifty watt fixed location repeater base

assures general coverage of the Turbine Building, Auxiliary Building, and Containment Building areas. The redundant repeaters receive emergency power, while individual radios have their own batteries.

9.5.2.4 Inspection and Testing

Q430.54

All communication systems are inspected and checked for operability after installation to assure proper operation and coverage. After a unit is operational, plant noise levels will be measured during normal and simulated accident conditions. Based on these measurements, an evaluation will be made to determine the need for sound isolation booths or noise-cancelling devices.

The communication systems are used routinely and do not require periodic testing.

430.108 Provide the location of underground cabling at Catawba. Describe the proximity of underground Class 1E cables to piping, roads or non-Class 1E cables and describe the qualification and design criteria used for the underground system.

Response:

All Class 1E cables in yard areas are located in conduit banks buried 5 feet minimum below grade, except those cables routed to the Refueling Water Storage Tank are in conduit located within the seismic Category I pipe trench. There are no direct buried Class 1E cables.

Conduit banks under roadways and railroads are encased in reinforced concrete capable of supporting H50 truck loading or E80 railroad loading.

For moisture and freeze protection, all conduit banks are sloped 1/8 inch per foot between manholes. The manholes have either gravity drains or sump pumps which discharge to the yard drainage system.

Where conduit is not encased in concrete, expansion joints are provided between manholes, and between building structures and manholes at intervals not exceeding 200 feet. Also, 4" diameter schedule 40 heavy-wall PVC conduit is used.

For protection of the conduit bank locations, concrete monuments at grade level located every 50 feet on centerline of the conduit bank are provided for identification.

The major Class 1E conduit crossing with non-safety piping is the crossing over the condenser cooling water lines which are moderate energy (70 psi lines). All other crossings are with gravity yard drainage or small process lines. It is anticipated that a break in any CCW line would have minimal affect on the Class 1E conduit since there are no structures built over these areas to prevent near vertical seepage from occuring. Thus, any postulated loss of water will be detected at yard level due to the impervious zone of group I earth backfill confining these areas. Therefore, only minor undermining of the Class 1E conduits could be expected and due to their inherent tensile strength, the cables would be able to span the resulting weakened area without interruption of their power function.

430.109 The load sequencing system at Catawba has an "accelerated sequence" feature which shortens the time intervals between load sequencing steps if system voltage and frequency setpoints are satisfied. Describe this feature in the Catawba FSAR and provide an analysis showing that the loads will be satisfactorily sequenced with no degradation to either the loads or the diesel generator.

Response:

See revised Section 8.3.1.1.2.

Attachment 5

Accident Evaluation Branch

Table 12.3.3-6 (Page 1)

Comparison of Containment Purge Filter System
With Regulatory Guide 1.52
Revision 2, March 1978

Paragraph	Compliance Status
C-1-a	The Containment Purge Filter System is not an engineered safety feature. System design is based on normal plant operation and shutdown modes. System design does; however, ensure a safe release path from the containment in the event of a fuel handling accident inside containment with the purge system operating.
C-1-b	Regulatory Guides 1.3 and 1.4 do not apply as the Containment Purge System is Technical Specification limited such that a LOCA is not postulated while the system is in operation.
Q450.8 C-1-c	See paragraph c-1-b.
C-1-d	In Compliance
C-1-e	In Compliance
C-2-a	System design provides two (2) 50% capacity filter trains and fans. System design does not include domisters or HEPA filters downstream of the adsorbers.
C-2-b	See paragraph c-2-a.
C-2-c	In Compliance
C-2-d	The Containment Purge System is isolated during the pressure surge resulting from a postulated LOCA. The system is not required to operate during or after the postulated LOCA.
C-2-e	In Compliance
C-2-f	In Compliance
C-2-g	System instrumentation consists of local flow and pressure drop indication. System discharge flow is totalized and indicated locally. System discharge flow is recorded at the control room.

Table 12.3.3-6 (Page 2)

Comparison of Containment Purge Filter System
With Regulatory Guide 1.52
Revision 2, March 1978

Paragraph	Compliance Status
C-2-h	Not applicable - the system is not designated as Class 1E electrical equipment.
C-2-i	The Containment Purge System operates continuously during the postulated fuel handling accident. The system is automatically isolated during the postulated LOCA.
C-2-j	Filter trains will not be removed as intact units. Gasketless filter adsorbers are used - which permits the fluidizing of carbon for external filling and removal. In this manner, we comply with ALARA recommendations.
C-2-k	In Compliance
C-2-l	In Compliance
C-3-a	System design does not include domisters.
C-3-b	Electrical components are not designated to Class 1E equipment.
C-3-c	Prefilters are tested in accordance with ASHRAE Standard 52.68 and carry UL Class 2 labels.
C-3-d	In Compliance
C-3-e	In Compliance
C-3-f	In Compliance
C-3-g	In Compliance
C-3-h	In Compliance
C-3-i	In Compliance
C-3-j	In Compliance

Attachment 6

Containment Systems Branch

The recombiner consists of an inlet preheater section, a heater-recombination section, and a discharge mixing chamber. The preheater section consists of a shroud placed around the central heater section to take advantage of heat conduction through the walls to preheat the incoming air. The heater section consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters. Four vertically stacked electric heater assemblies are integral to the heater section with each assembly containing individual heating elements. The discharge mixing chamber consists of the mixing chamber and exhaust louvers. The recombiner is provided with an outer enclosure to protect the unit from containment spray water.

The unit is manufactured of corrosion resistant, high temperature material. The electric hydrogen recombiner uses commercial type electric resistance heaters sheathed with Incoloy-800 which is an excellent corrosion resistant material for this service. These recombiner heaters operate at significantly lower power densities than in commercial practice.

Containment atmosphere is circulated through the recombiner by natural circulation. Air is drawn into the recombiner and passes first through the preheater section which serves the dual function of reducing heat losses from the recombiner and of preheating the air. The warmed air passes through an orifice plate and then enters the electric heater section where it is heated to approximately 1150-1400°F, a temperature sufficent to cause hydrogen recombination with the containment oxygen. Tests have verified that the recombination is not a catalytic surface effect associated with the heaters but occurs due to the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, saturating of the unit by fission products will not occur. Table 6.2.5-2 gives the recombiner design parameters. Operation of the recombiner is done manually from a control panel located outside the containment. The recombiner power supply panel and control panel are shown schematically in Figure 6.2.5-2. The power panel for the recombiner contains an isolation transformer plus an SCR controller to regulate power into the recombiner. This equipment is not exposed to the post-loss-of-coolant accident environment. To control the recombination process, the correct power input which will bring the recombiner above the threshold temperature for recombination will be set on the controller. The correct power required for recombination depends upon containment atmosphere conditions and will be determined when recombiner operation is required. For equipment test and periodic checkout, a thermcouple readout instrument is also provided in the control panel for monitoring temperatures in the recombiner.

Reference 1 provides a description of the testing of a full scale prototype electric hydrogen recombiner.

6.2.5.2.2 Containment Hydrogen Purge System

The Containment Hydrogen Purge System is shown in Figure 6.2.5-3.

The Containment Hydrogen Purge System consists of a containment hydrogen purge inlet blower, which blows air from the Auxiliary Building through a 4 inch pipe into the upper compartment of the containment. Another 4 in pipe originating in the upper compartment of the containment purges air from the containment to the annulus. A control valve located in the line can be throttled from the control room to the desired purge rate. The purged air from the containment mixes with the air in the annulus. The recirculation header of the Annulus Ventilation System guarantees a mix in the annulus volume prior to discharge, thus reducing the offsite dose. Since the annulus is kept at a negative pressure flow through the purge line can always be assured. The design flow of the hydrogen purge inlet blower is 100scfm, which is the same as one hydrogen recombiner. The blower has sufficient developed head to overcome piping losses and containment pressure. Redundancy is not required for the Containment Hydrogen Purge System, since it is a backup system to the redundant hydrogen recombiners. The hydrogen purge inlet blower and valves required for operation are manually operated from the control room.

Since the VY lines are closed at all times during power operation, Branch Technical Position CSB 6-4 does not apply to this system.

Equipment design data is presented in Table 6.2.5-3. for the Containment Hydrogen Sample and Purge System. Piping between containment isolation valves meet ASME III class 2 codes. Hydrogen sample piping in the Auxiliary Building meet ASME III class 3 codes.

6.2.5.3 Design Evaluation

6.2.5.3.1 Electric Hydrogen Recombiners

An analysis of potential hydrogen generation following the loss of coolant accident is summarized in Figures 6.2.5-4 and 6.2.5-5 and Table 6.2.5-4. The summary shows that, although the hydrogen production rate decreases with time after the accident total hydrogen accumulation can exceed the lower flammability limit of 4 volume percent unless positive measures are taken. Following a loss of coolant accident, the electric recombiner provides that means to prevent hydrogen concentration from exceeding safe levels.

For the purpose of showing that the electric recombiner is capable of maintaining safe hydrogen concentrations, analysis was performed using the NRC Regulatory Guide 1.7 Model. The result for the containment volume is shown in Figure 6.2.5-6.

during normal plant operations including start-up and shutdown transients. Containment pressure fluctuations due to postulated accidents are mitigated by safety related systems, rather than the Containment Air Release and Addition System.

Capacity is based on consideration of the estimated in leakage rate of air into the containment, the limits of 10CFR50 and Appendix I for routine releases, and the limits of 10CFR100 assuming concurrent loss of coolant accident and fan discharge. Also the capacity is such that the differential pressure from lower containment to upper containment is small enough to prevent the ice condenser doors from opening.

The containment air release fans are automatically shut off when the containment pressure is depressed to 0 psig to limit the impact on ECCS back pressure analysis and Containment negative pressure analysis.

The Containment isolation valves will automatically shut on an engineered safety signal to prevent containment air from being purged to the atmosphere during a loss of coolant accident or a steam break accident.

A comparison of the design of the VQ system to Branch Technical Position CSB 6-4 is given in Table 9.5.10-1.

9.5.10.2 System Description

The Unit 1 Containment Air Release and Addition System is shown on Figure 9.5.10-1. Each unit is provided with an identical system. Each system consists of two redundant containment air release filters for removing radioactive iodines and other radioactive particulates, two redundant containment air release fans, and associated valves, piping, and instrumentation.

Containment pressure increase may result from heatup of the containment atmosphere during start-up, leakage of air supplied to control and other air operated valves, or Nitrogen or Breathing Air System leakage.

Containment pressure reduction is accomplished by removing air from the upper containment. Containment pressure build-up is expected to result from leakage in the lower containment, but upper containment exhaust is preferred since it reduces the consequences of a loss of coolant accident concurrent with discharge. The open area between upper and lower containment is adequate to allow for equilibration of upper and lower containment for the range of pressure under consideration.

When the pressure inside the Containment increases to a set value an alarm is set off in the control room. The containment air release fans are placed in full recirculation by operator action by opening valves 1VQ2A, 1VQ3B, 1VQ15B, and 1VQ16A, and then starting the fans. After it is determined that the pressure increase was not due to a loss of coolant or steam break accident, and the containment air chemistry has been determined, the containment air is purged to the unit vent by opening valve 1VQ10 to a desired purge rate. A flow totalizer records the volume of air purged from the Containment. The flow rate and volume of air purged from the Containment will be coordinated with station release air purged from the containment will be coordinated with station release limits concerning the release of airborne activity. Valve 1VQ10 will automatically close on a high radiation signal from a radiation monitor located in the unit vent. When the pressure is decreased to 0 psig valve 1VQ10 will automatically shut.

Table 9.5.10-1

Comparison of VQ System to BTP CSB 6-4

BTP	Disposition
1a.	Actuators will close the containment isolation valves assuming full containment pressure differential and resultant flow.
b.	The system as shown in Figure 9.5.10-1 contains only one supply and one return line.
c.	The lines are nominal 4" pipe.
d.	The design of the containment penetrations is listed in Table 6.2.3-1 and 6.2.4-1.
e.	The containment isolation valves close on receipt of a "T" signal (phase A isolation). One of the parameters that can initiate a "T" signal is high containment airborne activity (see Figure 7.2.1-1, page 8).
f.	The containment isolation valves close within 5 seconds.
g.	The pipes which connect the containment isolation valves with the containment is Duke Class F (i.e., ANSI B31.1 pipe, seismically qualified). The pipes are open to upper containment atmosphere which will afford virtually complete isolation from high energy pipe break generated debris. In addition, the opening of the pipes are covered by a 40 mesh screen that is held in place between a pair of flanges.
2.	This system is designed only to control containment pressure during normal operation.
3.	This system is not designed or used to purge the containment to reduce airborne activity.
4.	The containment isolation valves, manual block valves and test connections are located in the pipe tunnel in lower containment.
5a.	See response to position 1.e. for valve closure signals. The amount of radiation that can realistically be expected to be released through this flow path is insignificant.
b.	The VQ system utilizes schedule 40 pipe which precludes rupture due to application of containment design pressure (15 psig). The only safety related equipment in the system are the containment isolation valves.
c.	If the system is in operation at the start of an accident the amount of air lost while the valves are closing is insignificant.
d.	An allowable leak rate for these valves will be developed in the type "C" test program.

TABLE 6.2.3-1 (Page 1)

Potential Bypass Leak Paths Through Containment Isolation Valves

Penetration Item Number	Service (Note 1)	Process Fluid	Essential (Note 7)	Design Features to Prevent Bypass Leakage — Attached Closed Systems (Note 2): Presents a Seismic Category 1 Closed Pressure Boundary to Containment Atmosphere Following a LOCA	Presents a Seismic Category 1 Closed Pressure Boundary to Environment Following a LOCA	Designed to Quality Group B or C Standards	Design Pressure Equals or Exceeds Containment Design Pressure (Note 4)	Design Temperature Equals or Exceeds Containment Design Temperature (Note 4)	Protected from Effects of Pipe Whip, Missiles, and Jet Forces Resulting From a LOCA	Pressure Boundary Maintained During Normal Plant Operation	Both Valves Served by Seal Water System	Leakage Path Terminates in Annulus	Remarks	Potential Bypass Leakage Path (Note 2)
01	Pressurizer Relief Tank Makeup	Water	No				X	X		X				Yes
02	Nitrogen to Pressurizer Relief Tank	Nitrogen	No				X	X		X				Yes
03	NC Pump Motor Drain Tank Pump Dischg.	Oil	No		X		X	X		X				Yes
04	NV Letdown Line	Water	No		X	X	X	X	X	X				No
05	Pressurizer Aux. Spray Transient Line	Water	No		X	X	X	X	X	X				No
06	NV Charging Line	Water	Note 8		X	X	X	X	X	X				No
07	NC Pump Seal Water Return	Water	No		X	X	X	X	X	X				No
08-11	NC Pump Seal Inj. Water 1A-1D	Water	Yes		X	X	X	X	X	X			Note 6	No
12	Reactor Makeup Water Tank Flush Hdr.	Water	No				X			X				Yes
13-14	Ice Condenser Ice Blowing Air	Air	No									X	Note 5	No
15	Ice Condenser Glycol Pump Dischg.	Glycol	No				X			X				Yes
16	Ice Condenser Glycol Pump Suction	Glycol	No				X			X				Yes
20	Containment H2 Purge Inlet Blower Dischg	Air	No		X		X	X	X	X			Note 6	Yes
21	Containment H2 Purge Outlet Line	Air	No		X		X	X	X	X		X	Note 6	No
22-23	ND Pump Suction 1A & 1B from Loop	Water	Note 9		X	X	X	X	X	X			Note 6	No

TABLE 6.2.3-1 (Page 2)

Potential Bypass Leak Paths Through Containment Isolation Valves

Penetration Item Number	Service (Note 1)	Process Fluid	Design Features to Prevent Bypass Leakage: Essential (Note 7)	Attached Closed Systems (Note 2): Presents a Seismic Category 1 Closed Pressure Boundary to Containment Atmosphere Following a LOCA	Presents a Seismic Category 1 Closed Pressure Boundary to Environment Following a LOCA	Designed to Quality Group B or C Standards	Design Pressure Equals or Exceeds Containment Design Pressure (Note 4)	Design Temperature Equals or Exceeds Containment Design Temperature (Note 4)	Protected from Effects of Pipe Whip, Missiles, and Jet Forces Resulting From a LOCA	Pressure Boundary Maintained During Normal Plant Operation	Both Valves Served by Seal Water System	Leakage Path Terminates in Annulus	Remarks	Potential Bypass Leakage Path (Note 2)
24	Boron Inj. Tank Line To Cold Legs	Water	Yes		X	X	X	X	X	X			Isolation valves open during LOCA to allow Boron Injection (Note 6)	No
25	Nitrogen to Accumulators	Nitrogen	No				X	X		X				Yes
26	Safety Injection Test Line	Water	No				X	X		X				Yes
27	ND Crossover Dischg. to Hot Legs	Water	Yes		X	X	X	X	X	X			Isolation valves open during LOCA to allow Hot Leg recirculation (Note 6)	No
28-29	NI Pump A & B Dischg. to Hot Legs	Water	Yes		X	X	X	X	X	X			Isolation valves open during LOCA to allow Hot Leg recirculation (Note 6)	No
30-31	ND HX Dischg. A & B to Cold Legs	Water	Yes		X	X	X	X	X	X			Isolation valves open during LOCA to allow Cold Leg recirculation (Note 6)	No
32	NI Pumps A & B Dischg. to Cold Legs	Water	Yes		X	X	X	X	X	X			Isolation valves open during LOCA to allow Cold Leg recirculation (Note 6)	No
33-34	Containment Sump Recirc. Lines A & B	Water	Yes		X	X	X	X	X	X			Isolation valves and penetration located below LOCA recirculation water level (Note 6)	No
35-36	Upper Head Injection Lines	Water	Yes		X	X	X	X	X	X			Isolation valves open during LOCA to allow Boron Injection to upper head (Note 6)	No
37	Upper Head Injection Test Line	Water	No				X	X		X			LOCA to allow Boron Injection to upper head (Note 6)	Yes

TABLE 6.2.3-1 (Page 3)

Potential Bypass Leak Paths Through Containment Isolation Valves

Penetration Item Number	Service (Note 1)	Process Fluid	Design Features to Prevent Bypass Leakage: Essential (Note 7)	Attached Closed Systems (Note 2): Presents a Seismic Category 1 Closed Pressure Boundary to Containment Atmosphere Following a LOCA	Presents a Seismic Category 1 Closed Pressure Boundary to Environment Following a LOCA	Designed to Quality Group B or C Standards	Design Pressure Equals or Exceeds Containment Design Pressure (Note 4)	Design Temperature Equals or Exceeds Containment Design Temperature (Note 4)	Protected from Effects of Pipe Whip, Missiles, and Jet Forces Resulting From a LOCA	Pressure Boundary Maintained During Normal Plant Operation	Both Valves Served by Seal Water System	Leakage Path Terminates in Annulus	Remarks	Potential Bypass Leakage Path (Note 2)
38-41	Containment Spray Lines	Water	Yes		X	X	X		X	X			Isolation valves open during LOCA to allow containment Spray Flow (Note 6)	No No
42,43	ND Containment Spray Lines A & B	Water	Yes		X	X	X		X	X			Isolation valves open during LOCA to allow ND containment spray flow (Note 6)	No
44	NC Drain Tank Gas Space to WG System	Hydrogen	No				X			X				Yes
45	NC Drain Tank HX Dischg.	Water	No				X			X	X			No
46	Vent. Unit Condensate Drain Hdr.	Water	Note 10				X			X	X		3 psi loop seal allows valves to remain open during small leaks inside containment	Yes
47	Cont. Floor Sump & Incore Inst. Sump Dischg.	Water	No				X			X	X			No
48	Steam Generator Drain Pump Dischg.	Water	No				X			X	X			No
49	Equipment Decontamination Line	Water	No				X			X				Yes
50	Fuel Transfer Tube	Water	No				X	X		X			Penetration terminates 30 ft below water level in fuel pool refueling canal	No
51	Refueling Water Pump Suction	Water	No				X			X				Yes
52	Refueling Cavity Fill Line	Water	No				X			X				Yes
53	Pressurizer Sample	Water	No				X	X		X				Yes
54	Reactor Coolant Hot Leg Sample	Water	No				X	X		X				Yes
55	Safety Injection Accumulator Sample	Water	No	X		X	X	X	X	X				No
56-59	Steam Generator Samples 1A, 1B, 1C, 1D,	Water	No	X		X	X	X	X	X				No
60	Comp.Cooling to RC Drain Tank HX	Water	No	X	X	X	X		X	X				No

TABLE 6.2.3-1 (Page 4)

Potential Bypass Leak Paths Through Containment Isolation Valves

Penetration Item Number	Service (Note 1)	Process Fluid	Design Features to Prevent Bypass Leakage: Essential (Note 7)	Attached Closed Systems (Note 2): Presents a Seismic Category 1 Closed Pressure Boundary to Containment Atmosphere Following a LOCA	Presents a Seismic Category 1 Closed Pressure Boundary to Environment Following a LOCA	Designed to Quality Group B or C Standards	Design Pressure Equals or Exceeds Containment Design Pressure (Note 4)	Design Temperature Equals or Exceeds Containment Design Temperature (Note 4)	Protected from Effects of Pipe Whip, Missiles, and Jet Forces Resulting From a LOCA	Pressure Boundary Maintained During Normal Plant Operation	Both Valves Served by Seal Water System	Leakage Path Terminates in Annulus	Remarks	Potential Bypass Leakage Path (Note 2)
61	Comp. Cooling from Drain Tank HX	Water	No	X	X	X	X		X	X	X			No
62	Comp. Cooling to Reactor Vessel Support & RCP Coolers	Water	Note 10	X	X	X	X		X	X				No
63	Comp. Cooling from Reactor Vessel Support & RCP Coolers	Water	Note 10	X	X	X	X		X	X	X			No
64	Comp. Cooling to Excess Letdown HX		No	X	X	X	X		X	X				No
65	Comp. Cooling from Excess Letdown HX	Water	No	X	X	X	X		X	X				No
66	Comp. Cooling to Component Cooling Drain Sump	Water	No				X			X				Yes
67	Nuclear Service Wtr. to NC Pump and Lower Cont. Vent. Units	Water	Note 10				X			X				Yes
68	Nuclear Service Wtr. from NC Pump & Lower Cont. Vent. Units	Water	Note 10				X			X	X			Yes
69	Nuclear Service Wtr. to Upper Cont. Vent.Units In	Water	Note 10				X			X				Yes
70	Nuclear Service Wtr. to Upper Cont. Vent. Units Out	Water	Note 10				X			X	X			Yes
71	Incore Instrumentation Rm Purge In	Air	No				X	X		X				Yes
72	Incore Instrumentation Rm. Purge Out	Air	No				X	X		X			Seismic piping extends through safety related filter trains	No
73-74	Upper Compartment Purge Inlet	Air	No				X	X		X			Note 6	Yes
75-76	Lower Compartment Purge Inlet	Air	No				X	X		X			Note 6	Yes
77-79	Containment Purge Exhaust	Air	No				X	X		X			Seismic piping extends through safety related filter trains, Note 6	No

TABLE 6.2.3-1 (Page 5)

Potential Bypass Leak Paths Through Containment Isolation Valves

Penetration Item Number	Service (Note 1)	Process Fluid	Design Features to Prevent Bypass Leakage										Remarks	Potential Bypass Leakage Path (Note 2)
				Attached Closed Systems (Note 2)										
			Essential (Note 7)	Presents a Seismic Category 1 Closed Pressure Boundary to Containment Atmosphere Following a LOCA	Presents a Seismic Category 1 Closed Pressure Boundary to Environment Following a LOCA	Designed to Quality Group B or C Standards	Design Pressure Equals or Exceeds Containment Design Pressure (Note 4)	Design Temperature Equals or Exceeds Containment Design Temperature (Note 4)	Protected from Effects of Pipe Whip, Missiles, and Jet Forces Resulting From a LOCA	Pressure Boundary Maintained During Normal Plant Operation	Both Valves Served by Seal Water System	Leakage Path Terminates in Annulus		
80-83	Steam Generator 1A-1D Blowdown	Water	No	X		X	X	X	X	X				No
84	Containment Air Release	Air	No				X			X				Yes
85	Containment Air Addition	Air	No					X		X				Yes
86-89	Feedwater	Water	Note 11	X		X	X	X	X	X			Isolation Valves and penetration located below steam generator water level	No
90-93	Aux. Feedwater	Water	Yes	X		X	X	X	X	X			Isolation Valves and penetration located below steam generator water level	No
94-97	Main Steam	Steam	Yes	X		X	X	X	X	X			Isolation Valves and penetration located below steam generator water level	No
98	Interior Fire Protection - Hose Racks	Water	No				X			X				Yes
99	Demineralized Water	Water	No				X			X				Yes
100	Instrument Air	Air	No				X			X				Yes
101	Station Air	Air	No				X			X				Yes
102	Breathing Air	Air	No				X			X				Yes
103-106	Containment Pressure Sensing											X	Note 6	No
107	Equipment Hatch	Air										X(3)		No
108	Personnel Hatch	Air										X(3)		No
109	RCP Oil Fill	Oil	No				X	X		X				Yes
110	Interior Fire Protection - Sprinklers	Water	No				X			X				Yes
111	Cont. Valve. Inj. Water "A" Train	Water	Yes		X	X	X		X	X			Note 6	No

TABLE 6.2.3-1 (Page 6)

Potential Bypass Leak Paths Through Containment Isolation Valves

Penetration Item Number	Service (Note 1)	Process Fluid	Design Features to Prevent Bypass Leakage: Essential (Note 7)	Attached Closed Systems (Note 2): Presents a Seismic Category 1 Closed Pressure Boundary to Containment Atmosphere Following a LOCA	Presents a Seismic Category 1 Closed Pressure Boundary to Environment Following a LOCA	Designed to Quality Group B or C Standards	Design Pressure Equals or Exceeds Containment Design Pressure (Note 4)	Design Temperature Equals or Exceeds Containment Design Temperature (Note 4)	Protected from Effects of Pipe Whip, Missiles, and Jet Forces Resulting From a LOCA	Pressure Boundary Maintained During Normal Plant Operation	Both Valves Served by Seal Water System	Leakage Path Terminates in Annulus	Remarks	Potential Bypass Leakage Path (Note 2)
112	Cont. Valve Inj. Water "B" Train	Water	Yes		X	X	X		X	X			Note 6	No
113	Standby Makeup Pump Discharge Line	Water	No				X	X		X		X	Penetration terminates 30 ft. below water level in fuel pool refueling canal	No

Potential Bypass Leak Paths Through Containment Isolation Valves

NOTES:

1. Reference Table 6.2.4-1 and Figure 6.2.4-1 for penetration listing and associated containment isolation design details.

2. An attached closed system is considered as a boundary which precludes bypass leakage if the system:

   a. Either (1) does not directly communicate, i.e., presents a closed pressure boundary, with the containment atmosphere, or (2) does not directly communicate, i.e., presents a closed pressure boundary, with the environment, following a LOCA.

   b. Closed pressure boundary is designed to Quality Group B or C standards.

   c. System closed pressure boundary is Seismic Category 1.

   d. If the system closed pressure boundary is inside Containment, its design pressure and temperature exceeds or is equal to Containment design pressure and temperature. If the system closed pressure boundary is outside of Containment, its design pressure exceeds or is equal to Containment design pressure.

   e. System closed pressure boundary is designed for protection from the effects of pipe whip, missiles, and any jet forces resulting from the LOCA.

   f. System closed pressure boundary is maintained during normal plant operation.

3. See Section 3.8.2.1 and Figure 3.8.2-1 details.

4. Containment isolation valves and operators will be designed to withstand internal conditions of the process piping and external conditions due to Post-LOCA temperature, pressure, humidity, and radiation.

5. Connected piping is temporary and is removed prior to startup. Penetrations are then provided with permanent weld caps inside and outside. These are effectively tested during Class A Integrated Leak Rate Tests.

6. Although the Containment Isolation System is itself an engineered - safety - feature, these lines belong to systems which perform an engineered - safety - feature function after LOCA.

7. Penetrations listed as non-essential are automatically isolated by closure of their containment isolation valve(s) on receipt of a "T"

Potential Bypass Leak Paths Through Containment Isolation Valves

NOTES: (continued)

signal (i.e. phase A containment isolation) or are normally locked or sealed closed.

8. Yes. Charging line is isolated on receipt of "S" signal (i.e., safety injection initiation).

9. Yes. These valves are closed during power operation. Automatic closure via containment isolation signals is unacceptable since spurrous signal could degrade core cooling.

10. Yes. Valve closes on receipt of a "P" signal (i.e., phase B containment isolation).

11. Yes. Valves close on feedwater isolation signal which is generated by hi-hi steam generator level, safety injection signal, or reactor trip with low T average. The only time a feedwater isolation signal is not generated when a "T" signal is generated is on manual actuation of "T" signal from control room.

Attachment 7

Core Performance Branch

Discussion

The Technical Specifications surveillance requirement for demonstrating the operability of the 125 VDC vital instrumentation and control power batteries is in accordance with the recommendations of Regulatory Guide 1.129.

Regulatory Guide 1.130

Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Revision 1, 10/78).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after April 1, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

Regulatory Guide 1.131

Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (8/77).

Discussion

The qualification testing of electric cables, field splices, and connections is discussed in Section 3.11.

Regulatory Guide 1.132

Site Investigations for Foundations of Nuclear Power Plants (Revision 1, 3/79).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after March 30, 1979) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

Regulatory Guide 1.133

Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Revision 1, 5/81).

Discussion

The Loose Parts Detection System for Catawba is in compliance with the requirements of Regulatory Guide 1.133, Revision 1. This system is described in Section 7.8.8.

Attachment 8

Licensee Qualification Branch

and observation training, (22 weeks minimum). Upon completion of this module, trainees have completed trainng in all topics for reactor operators described in 10 CFR 55.21, 10 CFR 55.23, and NUREG-0737.

(i) License Preparation Senior Operator

This training module reviews material covered in the License Preparation Reactor Operator module with emphasis on the following: Basis of Technical Specifications, Emergency Plan, Operating Characteristics, Fuel Handling, Handling and Disposal of Radioactive Waste, Facility Incident Reports, Core Damage Mitigation, and Supervisory Skills. It is normally presented to selected senior operator license candidates and is conducted at the nuclear stations. This module includes selected materials from Systems and Procedures Specific with emphasis on Safety Related Systems and Components, and Administrative Procedures. (16-18 weeks) Upon completion of this module, trainees have completed training in all topics for reactor operators described in 10 CFR 55.22, 10 CFR 55.23, and NUREG-0737.

(j) Simulator

Simulator training is a topic within the Periodic, Cold Certification, and License Preparation Modules. The simulator exercises are appropriately constructed for the level of training required. All license preparation, cold certification and periodic training modules contain the most advanced and demanding simulator exercises.

Simulator training is conducted on the Duke Power Company McGuire/Catawba simulator which complies with the requirements of Regulatory Guide 1.149. McGuire Nuclear Station is the reference plant for the simulator. However, its control board configuration, vendor, and response to transients satisfy the similarity requirements of 10 CFR 55 Appendix A. Prior to licensing all trainees receive a minimum of four reactor startups utilizing the simulator.

The simulator training associated with License Preparatory Training is scheduled for five weeks including one week for evaluation. Approximately one half of the time is used operating the simulator, while the remaining time is used in the classroom preparing for simulator operations and critiques of previous operations. The simulator training sessions are normally conducted in groups consisting of four trainees per group.

These simulator sessions are designed to train the operators in areas of event assessment, diagnosis, and response to the extent possible using Catawba Nuclear Station procedures on the McGuire simulator. Simulator training is supplemented with observation training at an operating nuclear station and procedure "walk-through" training at Catawba Nuclear Station.

(k) Observation

Observation training associated with Cold Certification is conducted at a Duke Power Company nuclear steam station. Normally the observation training is structured to reinforce the operating practices, procedures and administrative policies associated with nuclear station operations. The cold certification observation check list provides structured guidance for required observation

Attachment 9

Quality Assurance Branch

17.0 QUALITY ASSURANCE

17.0.1 DUKE POWER COMPANY QUALITY ASSURANCE PROGRAM

A discussion of the Duke Power Company quality assurance program is presented in the Duke Power Company Topical Report, "Quality Assurance Program", DUKE-1A.

17.0.2 APPLICABILITY

The Duke Power Company quality assurance program, as described in DUKE-1A, is applicable to items and activities designated as nuclear safety-related. Nuclear safety-related structures, systems, and components are identified in Tables 3.2.1-1, 3.2.1-2, 3.2.2-2, 3.10-1, 3.11-1, 3.11-2, 3.11-3, and 3.11-4.

17.0.2.1 Criteria Utilized in Determining Nuclear Safety-Related Structures Systems and Components

17.0.2.1.1 Criteria for Designation of Mechanical Systems and Components

The criteria used in determining nuclear safety-related mechanical systems and components is discussed in Section 3.2.2. Piping and specific valves in mechanical systems are not classified in Table 3.2.2-2. Instead, system flow diagrams are used to indicate the Duke piping classification. All piping and equipment in classifications A, B, C, and D are considered nuclear safety-
Q260.2 related. Classification F is nuclear safety-related only when it is made Class F to protect a Class A, B, C, or D system. Piping and equipment in classification E, G, and H are considered non safety-related.

Supports for piping and equipment follow the classification of the piping/equipment supported. In some cases safety-related supports are used with non-safety related piping/equipment.

17.0.2.1.2 Criteria for Designation of Electrical Systems and Components

The following criteria are established as a guide in determining nuclear safety-related electrical systems and components:

(a) Electric systems and components that are essential to shut down the reactor and limit the significant release of radioactive material following a design basis event.

(b) Controls and instrumentation systems and components that are utilized to develop a signal which is essential to initiate reactor protection and/or engineered safeguards functions.

(c) Instrumentation systems and components that are essential for post-accident monitoring. (Refer to Chapter 7).

(d) Cables and their support systems that provide power to or control nuclear safety-related components.

3. See revised Table 3.2.2-2 (Page 2).

4. See revised Table 3.2.2-2 (Page 2).

5. See revised Table 3.2.2-2 (Page 3).

6. See revised Table 3.2.2-2 (Page 3).

7. See revised Table 3.2.2-2 (Page 4).

8. See revised Table 3.2.2-2 (Page 4).

9. See revised Table 3.2.2-2 (Page 4).

10. See revised Table 3.2.2-2 (Page 4a).

11. See revised Table 3.2.2-2 (Page 4a).

12. Included in Reactor Vessel Internals, Table 3.2.2-2, Page 4a).

13. Part of Pressurizer (Table 3.2.2-2 (Page 4a)) and Reactor Coolant System piping.

14. The steam generator steam flow restrictors are integral with the steam generator steam outlet nozzle. The flow restrictors are qualified with and subject to the same QA program as the steam generators.

15. See revised Table 3.2.2-2 (Page 5).

16. See revised Table 3.2.2-2 (Page 5).

17. See revised Table 3.2.2-2 (Page 10).

18. See revised Table 3.2.2-2 (Page 10).

19. See revised Table 3.2.2-2 (Page 10).

20. See Section 17.0.2.1.1.

21. See Section 17.0.2.1.1.

22. See revised Table 3.2.2-2 (Page 9).

23. All penetrations are Class B except as discussed in Note 21, Table 3.2.2-2. As discussed in Section 17.0.2.1.1, Class B is nuclear safety-related and therefore under the Appendix B QA program.

24. As shown on Figure 9.4.10-1, ducting for the Containment Air Return and Hydrogen Skimmer System is Class B. As discussed in Section 17.0.2.1.1, Class B is nuclear safety-related and therefore under the Appendix B QA program.

25. See revised Table 3.2.2-2 (Page 8).

26. See revised Table 3.2.1-1 (Page 2).

27. a) and b) See Tables 3.11-1 through 3.11-4. Cable trays are included in Table 3.10-1 (Page 21).

    c) While fire stops are not safety-related and therefore not covered by the Appendix B QA program, they are treated as fire protection-related and are subject to QA Category 3 requirements.

28. a) and b) See Tables 3.11-1 through 3.11-4. Cable trays are included in Table 3.10-1 (Page 21).

29. As discussed in Section 9.4.2.2, this radiation monitor performs no safety function during irradiated fuel handling operations and is therefore not subject to the requirements of the QA program.

    Radiation monitors are discussed in Section 17.0.2.2.

30. See revised Table 3.2.1-1 (Note 6).

b) See revised Table 3.2.1-1 (Note 5).

c) The operational QA program includes provisions to assure that the design and implementation of modifications to the stations structures, systems and components is performed in a manner so as to assure that quality is maintained in a manner commensurate with the remainder of the system which is being modified, or as directed by applicable regulation. In the specific case of modifications to the site and roof drainage systems, the modification process would insure that the flood vulnerability of safety-related structures, systems, and components would not be increased.

d) Safety-related instrumentation and controls described in Sections 7.1 through 7.6 plus safety-related instrumentation and controls for safety-related fluid systems are subject to the pertinent requirements of the QA program described in Chapter 17.

e) The fuel oil recirculation pump and the clean and used lube oil tanks transfer pumps provide supplementary, non-safety related

services which are not essential for the operation of the diesel. Failure or loss of these components during a design basis earthquake would not hinder diesel operation. Under the guidelines set forth in Regulatory Guide 1.29, the fuel oil recirculation pump and the clean and used lube oil tank transfer pumps would not require seismic Category I design classification. Since these components are not safety-related, they are not subject to the requirements of the QA program.

Summary of Criteria - Mechanical System Components

System	Component or System	(2) Scope	(3) Safety Class	(4) Code	(5) QA Reqd.	(6) Location	(7) Rad. Source	(8) Seismic OBE	(8) Seismic DBE	(9) Tornado Wind	(9) Tornado Missile
	Fuel Assemblies	W	NA	-	X	C	X	X	X	X	X
	Control Rod Assemblies	W	2	-	X	C	X	X	X	X	X
|	Burnable Poison Rod Assemblies	W	NNS	-	X	C	X	-	-	X	X
	Reactor Vessel Internals	W	2	III-2	X	C	X	X	X	X	X
	Control Rod Drive Mechanisms										
|	Non Pressure Housing	W	NNS	-	X	C	X	-	-	X	X
	Pressure Housing	W	1	III-1	X	C	-	X	X	X	X
	Reactor Coolant Pumps	W	1	III-1	X	C	X	X	X	X	X
	Steam Generators (tube)	W	1	III-1	X	C	X	X	X	X	X
	(shell)	W	2	III-2	X	C	X	X	X	X	X
	Pressurizer	W	1	III-1	X	C	X	X	X	X	X
	Pressurizer Relief Valves	D	1	III-1	X	C	X	X	X	X	X
	Pressurizer Safety Valves	D	1	III-1	X	C	X	X	X	X	X
	Pressurizer Relief Tank	W	NNS	VIII	X	C	P	X	X	X	X
	RC Pump Motor Drain Tanks	D	NNS	VIII	-	C	P	X	X	X	X
	RC Pump Motor Drain Tank Pump	D	NNS	-	-	C	P	X	X	X	X
	Valves	D	1,2,3,NNS	III-1,2,3	X	C,AB	X	X	X	X	X
				B31.1.0						X	X
	Pressurizer Relief Tank Samp Vessel	D	NNS	(10)	-	AB	X	-	-	X	X

Table 3.2.2-2 (Page 8)

Summary of Criteria - Mechanical System Components

System	Component or System	(2) Scope	(3) Safety Class	(4) Code	(5) QA Reqd.	(6) Location	(7) Rad. Source	(8) Seismic OBE	(8) Seismic DBE	(9) Tornado Wind	(9) Tornado Missile
RY	Exterior Fire Protection										
	Fire Protection Pumps	D	NA	(15)	X	O	-	-	-	-	-
	Valves	D	NA	-	X	O,B	-	-	-	-	-
SA	Main Steam to Auxiliary Equipment										
	Valves (Note 22)	D	2	III-2	X	AB,DH	-	X	X	X	X
SB	Main Steam Bypass to Condenser System										
	Turbine Bypass Valves	D	NA	-	-	TB	-	-	-	-	-
SM	Main Steam System										
	Main Steam Isolation Valves	D	2	III-2	X	DH	-	X	X	X	X
SV	Main Steam Vent to Atmosphere System										
	Power Operated Relief Valves	D	2	III-2	X	DH	-	X	X	X	X
	Safety/Relief Valves	D	2	III-2	X	DH	-	X	X	X	X
TE	Feedwater Pump Turbine Exhaust										
	Valves (Note 22)	D	3	III-3	X	AB,DH	-	X	X	X	X
VA	Auxiliary Building Ventilation System										
	Aux. Building Radiation Air Supply Air Handling Units	D	NNS	-	X	AB	P	-	-	X	X
	Aux. Building General Air Supply Air Handling Units	D	NNS	-	X	AB	P	-	-	X	X
	Aux. Building Filtered Exhaust Filter Trains	D	3	(12)	X	AB	P	X	X	X	X
	Aux. Building Filtered Exhaust Fans	D	3	AMCA(14)	X	AB	P	X	X	X	X
	Auxiliary Building General Exhaust Fans	D	NA	-	-	AB	-	-	-	X	X
	Misc. Aux. Bldg. Air Handling Units By Cont.	D	NA	-	-	AB	P	-	-	X	X
	Auxiliary Building Filtered Exhaust Ducting and Dampers	D	3		X	AB	P	X	X	X	X
VB	Breathing Air System										
	Valves (Note 22)	D	2,NNS,NA	III-2, B31.1.1,	X	SB,AB,RB	-	-	-	-	-
VC	Control Area AC System										
	Control Room Air Handling Units	D	3	III-3 AMCA(14)	X	AB	-P	X	X	X	X
	Control Room Area Pressurizing Filter Train	D	3	(12)	X	AB	-P	X	X	X	X