UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PRC

SEP 19 1984

Docket No.: 50-458

MEMORANDUM FOR: The Atomic Safety and Licensing Board for River Bend Station (B. Paul Cotter, Jr., Richard F. Cole, G. A. Linenberger, Jr.)

FROM: Thomas M. Novak, Assistant Director for Licensing, Division of Licensing

SUBJECT: BOARD NOTIFICATION REGARDING ISSUANCE OF INTEGRATED (IDI) REPORT FOR RIVER BEND STATION (BN 84-156)

This notification is provided in accordance with NRC procedures regarding Board Notifications and is deemed to provide information material and relevant to safety issues in the River Bend Station OL proceeding. The appropriate parties are being informed by copy of this memorandum.

The enclosed letter from Richard C. DeYoung, Director, Office of Inspection and Enforcement, dated August 29, 1984, conveys the results, conclusions and Inspection Report 50-458/84-18 of the integrated design inspection of River Bend Station. The inspection took place from April 9, 1984, through June 1, 1984, and examined activities authorized by NRC Construction Permit No. CPPR-145.

Section 1 of the report provides a summary of the results of the inspection and the conclusions reached by the inspection team. The staff has concluded that the overall design process appeared adequate in each of the engineering disciplines inspected (mechanical systems, mechanical components, civil/structural, electrical power, and instrumentation and control). The staff, however, has a concern regarding the design verification process used for River Bend Station and is of the opinion that certain of the deficiencies identified in the report should have been found and corrected by the design verification process. The staff does not consider that the findings warrant negative conclusions concerning adequacy of the overall design process.

The applicant is required to respond in writing to the deficiencies and unresolved items within 60 days after receipt of the enclosed letter addressing the cause, extent, corrective action and any other information considered relevant.

Albert Schwencer /for

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure: As Stated

cc: ACRS (10)
Parties to the Proceeding
J. Nelson Grace, IE
EDO
See next page

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DISTRIBUTION LIST FOR BOARD NOTIFICATION

River Bend Station, Units 1&2
Docket Nos. 50-458/459

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Rev. 9/10/84

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 29, 1984

Docket No. 50-458/84-18

Gulf States Utilities Company
ATTN: Mr. William J. Cahill, Jr.
Senior Vice President -
River Bend Nuclear Group
P. O. Box 2951
Beaumont, Texas 77704

Gentlemen:

SUBJECT: INTEGRATED DESIGN INSPECTION 50-458/84-18

This letter conveys the results and conclusions of the integrated design inspection of the River Bend nuclear power plant. The inspection was conducted by the NRC's Office of Inspection and Enforcement. The team was composed of personnel from the NRC's Office of Inspection and Enforcement, Office of Nuclear Reactor Regulation and consultants. The inspection took place at the River Bend Station, West Feliciana Parish, Louisiana; Stone and Webster Engineering Corporation, Cherry Hill Operations Center, Cherry Hill, New Jersey; General Electric Company, Nuclear Energy Division, San Jose, California; Reactor Controls, Incorporated, San Jose, California; and Gulf States Utilities Company, Beaumont, Texas. The inspection took place over the period from April 9, 1984 to June 1, 1984. The inspection examined activities authorized by NRC Construction Permit No. CPPR-145. The team inspected areas defining whether (1) regulatory requirements and design bases as specified in the license application had been correctly translated and satisfied as part of specifications, drawings, and procedures, (2) correct design information had been provided internally and externally to the responsible design organizations including a selected off-site subcontractor, (3) design engineers had sufficient technical guidance to perform assigned engineering functions and (4) design controls, as applied to the original design, had also been applied to design changes, including field changes.

The inspection focused on the low pressure coolant injection mode of the residual heat removal system and the automatic depressurization system, although other areas were also covered as delineated in the enclosed inspection report. Activities included examination of design, design bases, design procedures, records, and inspection of the systems as installed at the plant. Emphasis was placed on reviewing the adequacy of design details as a means of measuring how well the design process had functioned for the selected samples.

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Deficiencies regarding errors, procedural violations and inconsistencies are identified in the report. Unresolved items are identified where more information is needed to reach conclusions. Other observations are identified where it was considered appropriate to call your attention to matters which are not deficiencies or unresolved items, but which are recommended for your consideration.

Section 1 of the report provides a summary of the results of the inspection and the conclusions reached by the inspection team. We have concluded that the overall design process appeared adequate in each of the engineering disciplines inspected (mechanical systems, mechanical components, civil/structural, electrical power, and instrumentation and control). However, based on the results of the inspection, including some of the more significant deficiencies identified in Chapter 1 of the report, we have a concern regarding the design verification process used for River Bend. We believe that certain of the deficiencies identified in the report should have been found and corrected by the design verification process. Our concern is heightened by the fact that system descriptions (or system design criteria) were not used to guide the design effort for the River Bend project. Use of such guidance is a standard approach at most architect/engineering firms, including Stone and Webster. It does not appear that the design verification process for the River Bend project was modified to accommodate the fact that the River Bend project did not use system descriptions. Because a great deal of good work was also reviewed, we do not consider that the findings warrant negative conclusions concerning adequacy of the overall design process. Based on these considerations, we have concluded that additional effort is required to provide assurance that the design verification process has been effective in detecting errors. A limited design review should be conducted by off-project Stone and Webster or Gulf States Utilities personnel of a sample of other safety related systems to determine whether or not deficiencies similar to those found by the IDI team can be expected elsewhere.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC Public Document Room unless you notify this office by telephone, within 10 days of the date of this letter, and submit written application to withhold information contained herein within 25 days of the date of this letter. Such applications shall be consistent with the requirements of 10 CFR 2.790(b)(1).

You are requested to respond in writing to the deficiencies and unresolved items within 60 days after receipt of this letter. With respect to the deficiencies identified in the report the response should address the cause, extent, corrective actions and any other information you consider relevant. For unresolved items, the response should provide information concerning acceptability of the specific feature or practice involved or indicate the extent to which corrective action is needed. In such cases the cause and corrective actions and any other information you consider relevant should also be included in the response. The response should be addressed to this office.

Should you have any questions concerning this inspection, please contact me or Mr. Ted Ankrum (301-492-4774) of this office.

Sincerely,

R C DeYoung

Richard C. DeYoung, Director
Office of Inspection and Enforcement

Enclosure:
Inspection Report
50-458/84-18

cc w/enclosure:
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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
Division of Quality Assurance, Safeguards, and Inspection Programs
Quality Assurance Program

Report No.: 50-458/84-18
Docket No.: 50-458
Licensee: Gulf States Utilities Company
P. O. Box 2951
Beaumont, Texas
Facility Name: River Bend Station
Inspection At: River Bend Station, West Feliciana Parish, Louisiana
Stone and Webster Engineering Corporation, Cherry Hill Operations Center, Cherry Hill, New Jersey
General Electric Company, Nuclear Energy Division, San Jose, California
Gulf States Utilities Company, Beaumont, Texas
Reactor Controls Incorporated, San Jose, California
Inspection Conducted: April 9-13, 23-27, April 30-May 4, May 14-18, and June 1, 1984
Inspection Team Members:

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Team Leader	R. E. Architzel, Senior Inspection Speciali t, IE
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	W. Anderson, Senior Mechanical Engineer
Civil and Structural	R. E. Lipinski, Structural Engineer, NRR
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*Inspection in both Electric Power and I&C Disciplines

Ralph E. Architzel — August 16, 1984
Ralph E. Architzel
Team Leader, IE
Date

DP Allison — August 16, 1984
Dennis P. Allison
Team Leader, IE
Date

Approved By: James L. Milhoan — August 16, 1984
James L. Milhoan
Section Chief, Quality Assurance Branch
Date

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TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

LIST OF DETAILS, DEFICIENCIES, UNRESOLVED ITEMS AND OBSERVATIONS

List of Abbreviations

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ASME	American Society of Mechanical Engineers
ANSI	American National Standards Institute
ASTM	American Society for Testing and Materials
BS	Bachelor of Science
CFR	Code of Federal Regulations
FSAR	Final Safety Analysis Report
HVAC	Heating, ventilation and air conditioning
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss of Coolant Accident
MS	Master of Science
NEMA	National Electrical Manufacturers Association
NPSH	Net positive suction head
NRC	U.S. Nuclear Regulatory Commission
P&ID	Piping and instrumentation diagram
PSAR	Preliminary Safety Analysis Report

1. INTRODUCTION AND SUMMARY

1.1 Objectives

In August 1982 the staff of the U.S. Nuclear Regulatory Commission (NRC) undertook a number of initiatives to improve assurance of quality in design and construction of nuclear projects. One of those initiatives was to develop and implement an integrated design inspection program to assess the quality of design activities, including examination of the as-built configuration. The objective was to expand the NRC examination of quality assurance into the design process. The approach is intended to provide a comprehensive examination of the design development and implementation for a selected system. This was the fourth inspection in that program.

1.2 Report Format and Definitions

The areas examined during this inspection are addressed by discipline in the following chapters. Deficiencies, unresolved items, observations, and details are defined below and are included in an appendix to this report.

(1) Deficiencies

Errors, inconsistencies or procedure violations with regard to a specific licensing commitment, specification, procedure, code or regulation are described as deficiencies. Followup action is required for licensee resolution and NRC evaluation of the resolutions.

(2) Unresolved Items

Unresolved items are questions which require more information to reach a conclusion. Unresolved items are described in the text of the following sections. Licensee response and NRC evaluation are required.

(3) Observations

Observations represent cases where it is considered appropriate to call attention to matters that are not deficiencies or unresolved items. They include items recommended for licensee consideration but for which there is no specific regulatory requirement. No licensee response is required.

(4) Details

Detailed descriptions of specific inspection findings or design details which may be of interest to a limited audience have been segregated from the main text of the report. No licensee response is required.

In our evaluation we found many design actions that were being well executed. Some of these positive findings are described in the text of the following sections. They are not flagged and numbered in the text nor listed at the front of this report since followup is not required.

1.3 River Bend Project Organization

Gulf States Utilities Company, co-owner of the River Bend Station with Cajun Electric Power Cooperative, is the applicant for the operating license. As such the Company is responsible for the design, construction and operation of the facility. Stone and Webster Engineering Corporation executes the overall plant design, having been contracted as the architect-engineer, design consultant, and construction manager. General Electric Company designs and provides the nuclear steam supply system and the turbine generators. The nuclear steam supply contract is managed directly by Gulf States. The bulk of Stone and Webster's production design work is performed at their Cherry Hill (New Jersey) Operations Center. This work is structured within the River Bend Project Organization. Technical assistance and oversight within disciplines, engineering assurance guidance, procurement specialists, and other assistance and oversight functions are provided by the Stone and Webster corporate office in Boston, Massachusetts. Stone and Webster's Site Engineering Group performs design activities related to construction and field assistance. Reactor Controls Incorporated was contracted by Stone and Webster to design and construct the piping and supports between the control rod drives in the drywell and their respective hydraulic control units in the containment.

1.4 Inspection Effort

The low pressure coolant injection mode of the residual heat removal system and the automatic depressurization system were selected for this inspection. Considered together, these systems are designed to mitigate a spectrum of loss of coolant accidents. The selected systems were pre-engineered to a large extent by General Electric Company, the nuclear steam system supplier. The engineering details and construction were provided by Stone and Webster Engineering Corporation. The components, functions and interfaces involved are typical of those found in a number of other safety systems.

The inspection was an interoffice NRC effort conducted with contractor assistance. Team members were selected to provide technical expertise and design experience in the disciplines listed. Many of the team members had previous experience as employees of architect-engineering firms or reactor manufacturers working on large commercial nuclear power plants. The others had related design experience on commercial nuclear facilities, test reactors or naval reactors.

A date of February 17, 1984 was established as the cut-off date for evaluation purposes and the team examined the design as it existed on that date. For example, if a document contained an error on February 17, 1984, that error formed the basis for a deficiency even if the document was later corrected. In some cases, more recent information which developed during the inspection was also reviewed to help understand the significance of an item. For instance, a subsequent analysis might show whether or not an error had any effect on the design. Where such information is used in this report it is described as more recent work done during the inspection.

Beginning on March 23, 1984 the inspection team devoted 2 weeks to the initial study of background information and preparation of plans for the inspection. The week of April 9 was spent inspecting at the site and the Architect Engineers' Cherry Hill Operations Center to become familiar with the Gulf States Utilities' and Stone and Webster Engineering Corporation's organizations and interfaces (including those with the General Electric Company) and gather additional background material. The week of April 16 was spent completing study of background information and plans for the inspection. Approximately three and one-half weeks of additional direct inspection activities were conducted at the Stone and Webster Cherry Hill Operations Center, the River Bend Station, General Electric Company Nuclear Energy Division, Reactor Controls Incorporated, and Gulf States Utilities, concluding on May 18, 1984. One team member returned to Stone and Webster during the week of May 21, 1984 to examine the design verification process. The exit meeting was held on June 1, 1984.

The inspection team reviewed the organizations' staffing and procedures and interviewed personnel to determine the responsibilities of and the relationships among the entities involved in the design process. Primary emphasis was placed upon reviewing the adequacy of design details (or products) as a means of measuring how well the design process had functioned in the selected sampling area. In reviewing the design details the team focused on the following items:

(1) Validity of design inputs and assumptions

(2) Validity of design specifications

(3) Validity of analyses

(4) Identification of system interface requirements

(5) Potential indirect effects of changes

(6) Proper component classification

(7) Revision control

(8) Documentation control

(9) Verification of as-built condition

(10) Application of design information transferred between organizations.

(11) Design verification methods

The team inspected five engineering disciplines within the project. The five disciplines were mechanical systems (see Section 2), mechanical components (see Section 3), civil and structural (see Section 4), electric power (see Section 5), and instrumentation and controls (see Section 6).

In some areas the sample was narrowed to include only a part of the selected systems. In other areas the sample was broadened into areas that were not related to the selected systems. More detailed descriptions of the review are provided in following sections of this report.

1.5 Major Conclusions

Although the inspection sampled a very small part of the design effort, the team did review hundreds of specific items as described in subsequent sections of this report. The most significant items are summarized below.

1.5.1 Mechanical Systems

In the mechanical systems area three substantial deficiencies were identified.

Passive failure in the low pressure coolant injection pump suction lines, which might result either in draining the suppression pool water below the minimum acceptable level or flooding emergency core cooling system equipment, was not properly considered (Deficiencies D2.3-2 and D2.3-3).

Resistance orifices in the low pressure coolant injection system were not installed. This decision was based on an incorrect calculation for net positive suction head and low pressure coolant injection pump runout flow rate. Without the resistance orifices there is a possibility of cavitation in the low pressure coolant injection loops during a design basis event (Deficiency D2.3-4).

In the design of the pneumatic supply for the automatic depressurization system valves it was improperly concluded that the safety grade air compressors alone were adequate. The design process failed to identify that the accumulators, associated check valves, and a higher air pressure (provided by the non-safety grade air compressors) are also needed for consistency with the functional requirements in the plant safety analysis. This may be corrected by appropriate analysis, specification of accumulator pressure, and leak testing of check valves. In fact, test and operations personnel had raised questions that might well have led to proper resolution in this manner. However, these questions had not been properly resolved at the time of the inspection and might not have been properly resolved given the misconception in the design process (Deficiency D2.4-6).

There were also deficiencies of lesser individual significance related to the adequacy or control of specific analyses in this area. These deficiencies included documentation errors, calculational errors, and failures to meet interface specifications provided by the nuclear steam system supplier.

The team reviewed other calculations in which no substantial problems were uncovered. These calculations were controlled, comprehensive, and technically correct. As a result, the team concluded that the design process in the mechnical systems area appeared to be adequately controlled.

1.5.2 Mechanical Components

A significant problem was found in this area with the loadings on safety relief valve nozzle flanges and the use of ball joints to reduce these nozzle loadings. Exceeding these nozzle loads could result in reactor coolant system leakage at the inlet flange of the safety relief valves. (Deficiency D3.3-3) A breakdown in the design process was found in the specification and evaluation of the ball joints to be used in the safety relief valve discharge lines. (Deficiency D3.6-2) Problems were found with required tests in the specification, review of test reports submitted by the vendor, transmittal of information to stress analysts, use of the information in the analysis, and feedback for results of the evaluation. A significant reanalysis effort may be required to resolve this problem and some re-design may be required. Use of ball joints is not customary in safety grade piping of nuclear power plants and the problems encountered may be attributed to lack of experience in their application. No similar problems of this extent were found in the design process for customary plant features.

The team also identified deficiencies in the pipe and equipment supports area. Calculations for the stiffness of pipe anchors and restraints attached to structural steel did not include the flexibility of the structural steel. This omission affects the analysis of selected supports. These calculations should be redone to obtain revised stiffnesses and these values should then be used in the final analysis cycle for the as-built piping configuration (Deficiency D3.4-4). Large seismic Category I equipment supported on structural steel was not analyzed in accordance with the FSAR commitment to verify such components can be legitimately decoupled from the response of the structure. Corrective action may require analysis to ensure decoupling is justified, if appropriate (Deficiency D3.5-3).

Despite the substantial amount of analytical rework that apparently will result from our inspection in the mechanical components area, we do not believe that significant hardware changes will be required after confirming the integrity of the existing design. The team was generally impressed by the availability of procedures, the documentation of work, and the ready availability of that documentation. The team concluded that the design process in the mechanical components area appeared to be adequately controlled.

1.5.3 Civil/Structural

The Stone and Webster design for radial shear did not incorporate required "hooks" for the end anchorage of reinforcement steel. Although this was a violation of the ACI Code, the structures were adequate because they either contained sufficient embedment to be considered fully developed or the need for such reinforcement was eliminated by subsequent design changes. The item was not an adverse finding with respect to design control for the River Bend project since the decision to select this design was a knowing decision that was documented, reviewed and accepted by project personnel. However, it does appear to reflect on the generic approach by Stone and Webster to embedment length of the inclined stirrups and indicates a need for assurance that the concrete structures built elsewhere have adequate capacity to resist radial shear. The NRC staff is pursuing this aspect on a generic basis.

In the Auxiliary Building two deficiencies were identified concerning the shear design of the mat (See Deficiencies D4.11-1 and D4.11-2). It is expected that a reanalysis of the mat will demonstrate the adequacy of the structure for shear. Two findings also require further investigation to confirm that the exclusion of control rod drive piping masses in the shield wall calculation and failure to address decoupling of the control rod drive piping from the pipe supports do not have an adverse effect on the design (Deficiency D4.3-2 and Unresolved Item U4.16-1).

There were instances of isolated weak points associated with some of the elements of design control as noted in the particular deficiencies and observations. The team was generally impressed with the quality of design products in the civil/structural area. The personnel interviewed in the Structural Division and the Geotechnical Division were highly qualified engineers with many years of experience on the River Bend Project. The calculations were clear, well referenced and organized. Based on the facts gained during the integrated design inspection, the team concluded that the design process in the civil structural area appeared to be adequately controlled.

1.5.4 Electrical Power

Two deficiencies were identified with respect to terminal block qualification. The documentation for terminal blocks inside containment was inadequate to support qualification. Also, Stone & Webster did not meet General Electric requirements for sealing conduit on terminal boxes inside containment (Deficiencies D5.10-1 and D5.10-2). The problem with terminal block qualification was identified by Stone and Webster prior to the inspection, however, the team felt that the problem was not being addressed in a timely fashion prior to the inspection. These terminal blocks are essential because they are used in redundant safety-related circuits within containment. During the inspection, Gulf States made the decision to bypass these terminal blocks which eliminated the concern regarding terminal block qualification.

A data sheet error in the specification for the Loops A and B low pressure injection valves resulted in the installation of incorrect (unqualified) motor actuators inside the containment (Detail A5.3-1). This problem also had been identified by the licensee prior to the inspection. However, it was obvious in nature and had remained undetected after several previous review of the documentation.

Overall, the results of the inspection in the electrical power discipline were positive. The design change process for the main control room was well controlled and documented. In general, the team was impressed with the accuracy and consistency that Stone & Webster maintained among numerous related electrical design documents. The team was also impressed with the project problem tracking scheme. Calculations and analyses were performed in a competent and professional manner. The team concluded that the design process in the electrical power area appeared to be adequately controlled.

1.5.5 Instrumentation and Control

In the instrumentation and control area the sample systems were expanded to include the Standby Service Water System and a portion of the High Pressure Core Spray System involving cooling of its diesel generator. Significant findings are described below.

Stone and Webster actuation logic for a LOCA for the standby diesel generators and certain Engineered Safety Feature loads required either off-site or on-site AC power to function. This application should have been provided noninterruptible AC or DC power. This design was changed during the inspection and is now considered acceptable. (Deficiency D6.5-2)

The team noted a design philosophy difference between the emergency core cooling systems designed by General Electric and their counterpart safety-related cooling systems designed by Stone and Webster. In the General Electric parts of design, generation of a LOCA signal causes the former systems to either begin accident mitigation or to operate in a mode that clearly demonstrates their immediate readiness to perform. In the Stone and Webster parts of design, the LOCA signal is not used to initiate or demonstrate availability of the latter safety-related systems. The Stone and Webster design approach is considered acceptable provided that high availability of these cooling systems, such as the standby service water system, can be assured during plant operation (see Sections 6.7 and 6.12).

The design of main control room display indications and annunciation alarms was reviewed with emphasis on accident monitoring instrumentation and indication of bypassed or inoperable status of safety-related systems relative to Regulatory Guides 1.97 and 1.47 respectively. Stone and Webster makes only two equipment quality designations; namely, QA CAT. I (i.e., safety-related including Class 1E) or QA Cat. II (i.e., non-safety-related). Some system status accident monitoring variables do not fit either of these two categories, and may require upgrading of some QA Cat. II equipment to be consistent with commitments to implement Regulatory Guide 1.97. (Deficiencies D6.9-1 and D2.3-1)

Stone and Webster design procedures in the Instrumentation and Control area used at Cherry Hill and at the River Bend site appear to be complete and comprehensive. Their use in accomplishing design work at both locations was evident in all of the documents reviewed. The team concluded that the design process in the instrumentation and control area appeared to be adequately controlled.

1.5.6 Overall Conclusions

Several problems identified in the River Bend design appeared to be threads of similar types of findings that crossed discipline boundaries.

The team was concerned about the design process in light of the findings discussed above. The post LOCA flooding calculation failed to properly address the degree of protection against draining the suppression pool or flooding ECCS equipment. The flow restricting orifices that are usually needed to meet General Electric specifications to provide adequate NPSH were eliminated from the residual heat removal system based on an incorrect calculation. The design of the air supply for the automatic depressurization system actuators failed to identify that several elements were necessary for consistency with the safety analysis performed by General Electric. The stress analysis of the safety relief valve discharge line containing ball joints failed to address or meet the interface load limits provided by General Electric. The diesel starting logic failed to fully meet the FSAR commitment to start on either loss of off-site power or LOCA. Because a great deal of good work was also reviewed, the team does not consider that these findings warrant negative conclusions concerning adequacy of the overall design process. However, they do raise questions concerning the design verification process. The team's concern was heightened by the fact that system descriptions (or system design criteria) were not used to guide the design effort for the River Bend project. Use of such guidance is a standard approach at most architect/engineering firms, including Stone and Webster. The design verification process was not modified to accomodate the fact that the River Bend project did not use system descriptions. Based on these considerations and in light of the nature of the findings discussed above, the team concluded that a limited design review should be conducted by off-project Stone & Webster or Gulf States Utilities personnel of a sample of other safety related systems to determine whether or not similar deficiencies can be expected elsewhere. The team recommends that these be selected from both nuclear steam supply systems pre-engineered systems and from Stone and Webster's scope of supply.

Another cross-discipline problem concerned the general quality of the calculation checking and design review documentation process being used by Stone and Webster. Many examples were identified of simple arithmetic errors, transcription problems, lack of ability to identify the individual responsible for detailed checking or review of the particular calculations, etc. Generally these deficiencies were minor in nature or had already been identified by other features of the design process. As a result, the team was not overly concerned with the River Bend design in this regard. The design organizations should investigate their practices in these areas and consider appropriate improvements (see Report Attachment A-1).

Although the aspects of the design process mentioned above raised concerns and questions, in other respects the team considered that the overall design process was well controlled. The team was very impressed with the control of documents within the project. The personnel assigned to the project were considered to be highly motivated and well qualified.

2. MECHANICAL SYSTEMS

The objective of this portion of the inspection was to evaluate the mechanical systems aspects of the design with emphasis on technical adequacy and the control of interface information. The team reviewed the low pressure coolant injection and automatic depressurization system designs and a number of sample areas of work which focused primarily upon the Stone & Webster Power Division.

2.1 Design Information

This section summarizes the basic mechanical systems design information reviewed.

Design commitments to the NRC are contained in the FSAR and related correspondence submitted in support of the operating license application. The basic systems design, design bases, functional requirements, failure analyses, and component data are described in these documents along with general information such as relevant accident analyses, high-energy line break analyses, and seismic requirements. These licensing commitments were prepared and submitted by Gulf States Utilities Company with considerable assistance from Stone & Webster Engineering Corporation and General Electric Company.

The nuclear steam system supplier's basic design specifications and interface information are contained in General Electric Company Design Specifications and Design Specification Data Sheets issued specifically for River Bend Station. These documents include piping and instrument diagrams, functional control diagrams, process and data diagrams, equipment lists, and functional test requirements. The basic systems design and major components are engineered by General Electric in this process. Interface requirements for Stone & Webster and Gulf States Utilities are specified by General Electric. This design basis information has been augmented considerably by correspondence between Gulf States Utilities, Stone & Webster, and General Electric over the life of the project.

Stone & Webster uses basic engineering and interface information provided by General Electric. The Stone & Webster engineering effort includes design of subsystems within the low pressure coolant injection and automatic depressurization systems, pipe routing, and detailed sizing of equipment based on the specific in-plant arrangement. This effort results in calculations which incorporate the information provided by General Electric. These calculations are in some cases performed as verification of General Electric recommendations. In other instances where Stone & Webster has design and equipment purchase responsibility, the calculations provide performance data for the purchase specifications.

2.2 Personnel and Guidance

This section summarizes the basic staffing and guidance information reviewed.

The responsibility for the Stone & Webster design effort related to mechanical systems lies within the Power Division of the project team. The Power Division is organized around the key positions of principal engineer, lead engineer, responsible system engineer, and responsible purchase engineer. Each of these key assignments has a specific set of responsibilities within the design process. Both the Power Division and the site have similar engineering/design project team organizations. The key distinction between the two groups lies in the type of design function performed. The function of the Power Division is to design systems which meet functional or process requirements. The function of the site team is to perform those design activities required to support installation of the systems.

The team briefly surveyed the education and experience of the Stone & Webster personnel associated with the River Bend project from the Power, Nuclear Technology, Materials, and the Hydraulics Divisions.

In the Power Division 35 individuals are assigned to positions ranging from Lead Engineer to Engineering Asssociate. Of these individuals 31 have engineering degrees. The breakdown by degree is as follows:

B.S. Mechanical	16
M.S. Mechanical	5
B.S. Nuclear	1
M.S. Nuclear	1
B.S. Other Diciplines	8
Total	31

Of these engineers 12 have Professional Engineer registrations in one or more states. The Lead Power Engineer has a mechanical engineering degree and has worked for Stone & Webster for 12 years. All of those 12 years have been on the River Bend Project. On the design floor the Power Division designers are supervised by 7 individuals with a total engineering experience of 131 years of which 74 have been on the River Bend Project.

In the Nuclear Technology Division 10 individuals are assigned. Of these individuals 8 have degrees. The breakdown by degree is as follows:

Ph.D. Mechanical	1
M.S. Mechanical	3
M.S. Radiation Health Physics	1
B.S. Electrical	1
B.A. Math	1
B.A. Engineering	1
Total	8

Of these individuals 6 have Professional Engineer registrations in one or more states. The 10 engineers assigned to the Nuclear Technology Division have a total engineering experience of 80 years of which 33 have been on the River Bend Project.

In the Materials Division 4 individuals are assigned. The Lead Materials Engineer has a Bachelors of Science degree in metallurgical engineering and has 22 years of engineering experience of which 5 have been associated with the River Bend Project.

In the Hydraulics Division only one engineer is assigned. He has a Bachelor of Science degree in mechanical engineering, a professional engineer registration, and 12 years of engineering experience of which 4 have been associated with the River Bend Project.

The team's review of guidance was limited to the Power Division. Stone & Webster has a continuing education program in each of their main disciplines. The program is designed to provide information relating to equipment, design approaches and procedures. The subject matter of the seminars is varied annually and reflects the type of work performed within the discipline. Each design center has an individual curriculum that is set by the discipline manager at that location. The instructors are usually experts from within Stone & Webster but on occasion outside experts conduct the seminars. Formal training courses are available in the area of calculation preparation, introduction to systems engineering, pump application, and design checking.

Guidance is also provided in the form of published memoranda, procedures and standards. The team observed the use of Project Management Memoranda and Power Technical Procedures. Project Management Memorandums are issued by the River Bend Project Manager to provide direction for interdisciplinary design activities. For example, the team observed that Project Management Memoranda were used to publish the procedures for conducting high energy line break and moderate energy line crack evaluations. Power Technical Procedures were used to describe Power Division's position in various design details. For example, the team observed that a Power Technical Procedure was used to describe Power Division's position on imple-mentation of the single failure criterion. Unlike some Architect-Engineer companies, Stone & Webster did not have a large compilation of division or company procedures prescribing in detail how various design calculations were to be performed. Instead, the choice of technical approach to calculations such as NPSH, total discharge head, or pipe sizing was left to individual engineers and their supervisors.

2.3 Low Pressure Coolant Injection System Design

OBJECTIVE

The objective of this portion of the inspection was to evaluate the adequacy and control of design information for the low pressure coolant injection system.

EVALUATION

The team inspected the installed configuration of the low pressure coolant injection loops including the type of fittings, size of lines, and location and elevation of equipment. This configuration was compared to that in design documents including piping drawings and flow diagrams. No substantial discrepancies were found between the installed system and these controlled Stone & Webster design documents.

The team inspected various design documents for conformance to requirements of the design basis. Calculations and equipment purchase specifications were evaluated for conformance to the General Electric supplied system design requirements.

A significant deficiency was found with respect to the Stone & Webster decision to eliminate the resistance orifice from each low pressure coolant injection loop. An error was found in the calculational method used as the basis of the decision that directly resulted in the conclusion not to install the orifices. Also, the calculation was performed in a manner that was conservative for total dynamic head but nonconservative for limiting runout flow. It is expected that, when the error is corrected, the system will exceed the runout limitation for the low pressure coolant injection mode. This will result in a higher net positive suction head required for the pump which will cause cavitation if it exceeds the net positive suction head available (see Deficiency D2.3-1).

The team reviewed the low pressure coolant injection system for its ability to withstand a single passive failure without jeopardizing adequate core cooling during long term post LOCA operation. The team identified a potential common mode failure that could affect the three low pressure coolant loops and other emergency core cooling systems which take suction from the suppression pool. A single passive failure in the low pressure coolant injection pump suction lines could result in draining the suppression pool water below the minimum acceptable level. In addition, flooding of emergency core cooling system equipment could result. The impact of this deficiency is that design changes may be needed to prevent excessive loss of suppression pool inventory and flooding of emergency core cooling system equipment (see Deficiencies D2.3-2 and D2.3-3).

The team inspection included the fill systems which are designed to maintain the emergency core cooling systems in a condition of readiness. Our inspection revealed areas where either the design or related documentation could be improved:

The design flow rate of the fill pumps was based on a calculation which was voided by Stone & Webster (see Deficiency D2.3-4).

The alarm for an abnormal condition of the discharge line combines the two conditions of not being filled and over pressurization into one alarm without providing some means to distinguish which condition exists (see Observations O2.3-1 and O2.3-2).

The installed low pressure core spray system fill pump which also serves loop A of the residual heat removal system has a total developed head at design flow substantially under that which the calculation determined to be necessary (see Deficiency D2.3-5).

The team inspected preoperational test procedures for the low pressure coolant injection mode of operation. This inspection revealed two concerns. First, generic General Electric documents rather than Stone & Webster design documents directly applicable to River Bend were not used as the basis for developing test procedures. This resulted in a test procedure to verify the sizing of a device that is not installed in the system (see Deficiency D2.3-6). Second, the method for conversion of net positive suction head available at test conditions to design basis conditions had a twofold problem: a) the method for adjusting the measured value is technically incorrect, and b) the interpretation of the design basis conditions was not complete. This portion of the preoperational test has added importance because it will be the basis for determining whether or not a Stone & Webster requested exception to the General Electric design basis is valid (see Deficiency D2.3-7).

The team reviewed the Stone & Webster procedures for purchasing engineered equipment. The team found sound procedures in place for integrating equipment purchases into the design process (see Detail A2.3-1). The team also reviewed how Stone and Webster incorporates manufacturer's information, such as certified pump curves, into the design process. It was noted that the vendor certified pump performance curves for the residual heat removal pumps on River Bend Units 1 and 2 were different and that this information could have been better distinguished to minimize the potential for inadvertant mixing of design data (see Observation O2.3-3).

CONCLUSION

The findings discussed above indicate weaknesses in the design process in the following areas: a) understanding of General Electric supplied design basis, b) coordination or transfer of information between steps in the design process, c) use of governing design documents, and d) documented basis for design decisions. Although these problems were found, the fact that the low pressure coolant injection mode for the residual heat removal system was substantially pre-engineered by General Electric reduced their impact on the design. Because of the significance and number of deficiencies as discussed in this section and others, the team was concerned that these weaknesses are systematic. Appendices A-1 and A-2 discuss this overall concern and the team's recommendation for further design review.

The team found capable and motivated personnel within all the contacted organizatons. Their ability to answer questions, provide data and produce documents allowed a thorough inspection and indicated a project strength. The team was able to review all the steps of the design process. This included the original design basis, calculations, production drawings, equipment purchase specifications, vendor certified data, installation and preoperational test procedures.

RELATED FORMS:
A2.3-1 (Detail) Design Process and Equipment Purchase
D2.3-1 (Deficiency) RHR Pump Runout Calculation
D2.3-2 (Deficiency) Failure to Prepare a Calculation that Meets the Stated Objective
D2.3-3 (Deficiency) Failure to Consider Passive Piping Failure in Emergency Core Cooling System Suction Lines Post-LOCA
D2.3-4 (Deficiency) Lack of Documented Basis for Sizing Class 2 Equipment
D2.3-5 (Deficiency) Insufficient Fill Pump TDH
D2.3-6 (Deficiency) Preoperational Tests for Confirming Compliance with System Design Basis
D2.3-7 (Deficiency) Preoperational Test to Verify LPCI NPSHa (212 degrees F)
O2.3-1 (Observation) RHR Discharge Line Abnormal Condition Alarm
O2.3-2 (Observation) ECCS Subsystem Fill Pump Runout
O2.3-3 (Observation) Control of Design Basis Information

2.4 Automatic Depressurization System Design

OBJECTIVE

The objective of this portion of the inspection was to evaluate the adequacy and control of design information for the automatic depressurization system.

EVALUATION

The team reviewed the basic automatic depressurization design information contained in the FSAR, design specifications issued by the nuclear steam supplier, and selected design calculations and outputs prepared by Stone & Webster.

Sections 5.2, 6.3 and 9.3 of the River Bend FSAR describe the design and operation of the automatic depressurization system including the pneumatic supply to the automatic depressurization system valves.

Three design specification documents issued by General Electric were selected for a detailed review. These specifications have been incorporated into the design freeze as agreed upon by General Electric, Stone & Webster, and Gulf States Utilities for the nuclear boiler system. The team reviewed these specifications to determine whether the actual automatic depressurization system design was consistent with the interface needs of the nuclear steam supply system.

The major design calculation evaluated for the automatic depressurization system pneumatic supply was calculation PN-255, Main Steam Safety Relief Valve Pneumatic System Piping Design Verification. The team reviewed this calculation for adequacy with respect to design of the pneumatic supply to the automatic depressurization system valves.

The design outputs selected for evaluation were flow sketches, logic sketches, setpoint data sheets and design output to groups such as inservice inspection, startup testing, plant technical specifications and plant operations.

Our review found two deficiencies in the assumptions made by Stone & Webster to calculate the adequacy of the safety grade air supply to the safety/relief valves. First, the assumption for safety/relief valve operation was nonconservative in that the low setpoint valve may operate more frequently than assumed in the analysis. Second, the safety grade air compressor actually purchased will provide a lower pressure than that assumed in the analysis. Both of these calculational deficiencies are of minor significance because the compressors purchased are adequate to supply long term air requirements (see Deficiencies D2.4-1 and D2.4-2).

Our review indicated that Stone & Webster did not adhere to two General Electric design specifications that Stone & Webster had agreed to meet in the design freeze documentation. The automatic depressurization pneumatic supply design deviated from the General Electric specifications incorporated into the design freeze; however, Stone & Webster had not informed General Electric to obtain resolution. First, a safety grade air supply of 150 psig minimum was specified and a supply of approximately 105 psig was provided. Second, the automatic depressurization system accumulator check valves were specified to have "bubble tight" shutoff; however, these valves were purchased without this requirement and they were not required to be tested for "bubble tight" shutoff (see Deficiencies D2.4-3 and D2.4-4).

The safety grade automatic depressurization system pneumatic supply pressure is monitored by instrumentation on each supply system header outside the drywell. This instrumentation is classified as nonsafety related. Since this instrumentation is used to determine availability of the safety grade automatic depressurization system pneumatic supply, it should be classified as type D instrumentation in accordance with Regulatory Guide 1.97 (see Deficiency D2.4-5).

We found that the design calculation was incorrect in concluding that the safety grade air compressors alone are adequate as the pneumatic supply for the automatic depressurization system and safety/relief valves. This calculation did not identify that the automatic depressurization system accumulator supply is needed as a short term pneumatic supply in case one compressor becomes inoperable due to the single failure of a diesel generator. Accordingly, the calculation did not establish required operational and testing criteria for the automatic depressurization system accumulators, including associated instrumentation and check valves (see Deficiency D2.4-6).

The impact of these deficiencies on design is not expected to result in significant hardware changes. The overall automatic depressurization system configuration, sizing, and electrical power sources conform to design specifications and functional requirements in the plant safety analysis. Stone & Webster should verify adequate pneumatic supply to operate the automatic depressurization system valves in accordance with the plant safety analysis and should establish operating instrumenta-tion and testing requirements for the automatic depressurization system pneumatic supply.

SUMMARY/CONCLUSION

In summary, our review found the automatic depressurization system design correct and in conformance to the FSAR licensing commitments except for several deficiencies in the design of the automatic depressurization system pneumatic supply. We found that design of the pneumatic supply was not fully in conformance with the nuclear steam system supplier's specifications agreed upon in the design freeze. The calculational deficiencies were relatively minor errors. More significant errors were found in that the design process did not identify and verify functional requirements for the automatic depressurization system accumulators and associated instrumentation and check valves. These errors might have been resolved during preparation of the preoperational tests, plant technical specifications, or the inservice inspection program, but this is not clear. As a result, the pneumatic system design was not fully consistent with functional requirements in the plant safety analysis. Otherwise, the overall automatic depressurization system design was correct and consistent with licensing requirements and nuclear steam supply steam interface requirements.

RELATED FORMS:
D2.4-1 (Deficiency) ADS Pneumatic Sizing
D2.4-2 (Deficiency) ADS Pneumatic Supply Adequacy
D2.4-3 (Deficiency) ADS Pneumatic Supply Specifications
D2.4-4 (Deficiency) ADS Accumulator Check Valve Specifications
D2.4-5 (Deficiency) ADS Pneumatic Supply Instrumentation
D2.4-6 (Deficiency) SRV Pneumatic Supply Adequacy

2.5 Line Break Analysis

OBJECTIVE

The objective of this portion of the inspection was to evaluate the adequacy and control of high energy line break and moderate energy line crack analyses.

EVALUATION

Section 3.6 of the River Bend FSAR describes the design analysis and physical protection provided against dynamic effects associated with the postulated rupture of piping. This FSAR section describes the design for protection against postulated piping failure both inside and outside containment including all high and moderate energy piping systems. The inspection team had planned to review in detail the effects on the low pressure coolant injection system from one high energy line break location and one moderate energy crack location. However, since little moderate energy work had yet been done, the team focused its review on high energy line break analyses (see Detail A2.5-1). The team further limited its review to only the jet impingement effects.

The break location selected was break 30 of main steam line loop B because this break is located in the drywell and a steam jet with the reactor pressure vessel as its energy reservoir impinges two ASME Class 1 low pressure coolant injection lines.

The team reviewed Engineering Mechanics Division's jet impingement target analyses performed to identify the jet load on each individual target in the jet path. The team found very detailed and lengthy calculations. The analyses were organized in a logical manner and were, in general, prepared in accordance with Engineering Mechanics Division's calculation procedures.

Some minor administrative errors were found, such as transcription mistakes, a revision to a page that was not identified in the list of revised pages, and a misleading objective statement. These types of errors were relatively infrequent and, considering the magnitude of the calculations, were of no significance. The team also observed three deficiencies that require more extensive corrective action. The first involves the failure to consistently identify in the jet impingement target analyses the sources of design input used within the design analyses. This deficiency appears to be systematic, because of the number of examples found in a very limited sample size (see Deficiency D2.5-1). The second involves an error in recording the results of a pipe target calculation into the Jet Impingement/Pipe Rupture Data File. Although the pipe target calculation had the correct value for jet impingement load on an ASME Class 1 line, the computer data file had a lower value than calculated (see Deficiency D2.5-2). The third involves a failure to keep the structural target calculation current such that it reflects the as constructed condition. Specifically, the team used the targets identified by Engineering Mechanics Division's calculations and conducted a field walkdown of break location 30 of main steam line loop B. The field configuration of structural components differed from that used in Engineering Mechanics Division's jet impingement target analyses. The team concluded that the analysis methodology employed by Engineering Mechanics Division can result in correct structural steel target identification provided the calculations are revised when significant changes are made to structural steel drawings. It was noted that this particular deficiency is related to the first in that it is a difficult and tedious task to identify what portion of an analysis requires revision when the analysis does not consistently identify the references used. It appears that the inconsistency between the calculated structural targets and the actual field configuration are systematic (see Deficiency D2.5-3).

The identification of pipe whip and jet impingement targets within Engineering Mechanics Division are limited to structural targets, piping and pipe supports (generally large bore pipe), electrical cable trays, control instrumentation and equipment. Other potential targets such as electrical conduit, small bore piping and instrumentation tubing are to be identified by field walkdown prior to building turnover. Stone & Webster's staff also indicated that during the planned walkdowns they will have assistance from the Construction Systems Associates, Inc. three-dimensional space model. Based upon the team's walkdown of break location 30 of main steam line loop B, the team is concerned about the Stone & Webster ability to identify these targets (see Observation O2.5-1).

Once the jet impingement targets have been identified by Engineering Mechanics Division, this design information is transmitted to the High Energy Line Break Coordinator in the Power Division. The High Energy Line Break Coordinator is responsible for transmitting a copy of the target identification packages to Electrical, Power (Building Services), and Structural Divisions. For each high energy line break location, each Division is responsible for evaluating the acceptability of a target/break interaction using the guidance of High Energy Line Break Evaluation Procedure. Since the High Energy Line Break Coordinator is in the Power Division, he is responsible for evaluating the acceptabiltiy of target/break interactions associated with mechanical equipment and instrument targets. The inspection team examined the results of the High Energy Line Break Coordinator's evaluation of the pipe and pipe support jet impingement target/break interactions to evaluate the quality of the work performed to date and to assess the adequacy of the guidance of the High Energy Line Break Evaluation Procedure.

In general the team found an evaluation effort that was not as sophisticated and as well directed as Engineering Mechanics Division's efforts in identifying the target/break interactions. This observation can in part be attributed to the fact that little, if any, activity was taking place prior to January 1984 to evaluate these interactions. The High Energy Line Break Evaluation Procedure was not formally documented until January 1984, and interviews with Stone & Webster staff indicated that essentially all of the activity prior to that time was in Engineering Mechanics Division. As described previously, the Engineering Mechanics Division was primarily involved in identifying the break location, designing pipe restraints and identifying the pipe whip and jet impingement targets. Since January 1984 two revisions have been incorporated in the High Energy Line Break Evaluation Procedure.

The team reviewed the initial issue of the High Energy Line Break Procedure (subsequent revisions were issued after the inspection cutoff date of February 17, 1984). The team found that some of the FSAR commitments were not properly translated into the procedure. This deficiency was identified by Stone & Webster staff and corrected prior to the inspection team's arrival on the River Bend Site (see Deficiency D2.5-4).

The team's review of Power Division's evaluation of piping and pipe support jet impingement targets identified two deficiencies. The first deficiency is associated with errors and inconsistencies in their evaluations (see Deficency D2.5-5). The inspection team's assessment of these errors and inconsistencies is that the High Energy Line Break Evaluation Procedure is deficient in the following areas:

It does not provide sufficient guidance to individuals performing evaluations to correctly categorize unacceptable target/break interactions into groups such as required for safe shutdown, containment isolation, environment.

It does not identify what criteria the High Energy Line Break Coordinator will use to perform a preliminary review to determine whether other acceptable means are available to achieve a safe shutdown for each pipe break.

It does not require that evaluations performed in accordance with the procedure be treated as design analyses such that the results are checked and reviewed.

In the second deficiency the inspection team found that design input into the high energy line evaluations was not being identified. Specifically, the team found that the date/revision number of applicable system flow diagrams used to determine the safe shutdown configurations were not being recorded (see Deficiency D2.5-6).

Both of these deficiencies are considered systematic weakness of the evaluations currently being performed.

The team also examined the flow of high energy line break design information between the Engineering Mechanics and Power Divisions and found that the information was treated in an uncontrolled manner. The incidents of informal transmittal of design information between these divisions represents a significant deficiency in the design control process associated with the analyses of high energy line breaks (see Deficiency D2.5-7).

SUMMARY/CONCLUSION

In summary the team observed a fairly extensive design analysis effort to identify target/break interactions. In general the team believes the effort was thorough and complete with two exceptions. These are the documentation of references used in the calculations and the inconsistency between the calculated structural targets and the as constructed condition. With respect to evaluation of the identified target/break interactions, which is work that was essentially just beginning, the team observed the implementation of a procedure which is deficient.

The team concluded that improvements should be made in the high energy line break evaluation procedure and in its implementation to assure that the River Bend design is adequate with respect to high energy line breaks. With respect to moderate energy line cracks insufficient information was available to make a conclusion.

RELATED FORMS:
A2.5-1 (Detail) Moderate Energy Line Cracks
D2.5-1 (Deficiency) Failure to Assure That Sources of Design Input Are Identified in Design Analysis
D2.5-2 (Deficiency) Incorrect Jet Impingement Loading on ASME Class 1 Low Pressure Coolant Injection Line
D2.5-3 (Deficiency) Field Configuration of Structural Component Differ From That Used In Engineering Mechanics Division's Jet Impingement Target Calculations
D2.5-4 (Deficiency) Failure To Translate FSAR Commitments Into High Energy Line Break Evaluation Procedure
D2.5-5 (Deficiency) Errors and Inconsistencies In HELB Evaluations
D2.5-6 (Deficiency) Design Control of Input to High Energy Line Break Evaluation
D2.5-7 (Deficiency) Failure to Control the Flow of High Energy Line Break Design Information Between Internal Design Groups
O2.5-1 (Observation) HELB Targets Identified by Field Walkdown

2.6 General Electric Information

OBJECTIVE

The objective of this review was to evaluate the information exchange between General Electric and Stone & Webster pertaining to the design of the low pressure coolant injection and automatic depressurization systems and performance of the plant safety analyses.

EVALUATION

We reviewed the General Electric program to control the design process in general and for the low pressure coolant injection and automatic depressurization systems specifically. We reviewed selected General Electric Design Specifications and Data Sheets for the low pressure coolant injection and automatic depressurization systems. We found that General Electric had these design documents identified and controlled for the River Bend Project. We found that the lead systems engineers were experienced and knowledgeable in the design of the low pressure coolant injection and automatic depressurization systems.

We reviewed functional qualification testing for the low pressure coolant injection pumps. We found that tests had been performed for these pumps and results were documented in a test report. We reviewed the manufacturer's certified pump curves for the low pressure coolant injection pumps and checked for consistency with plant safety analyses and process diagrams including data provided for the pump runout mode. We found consistency in the information provided by the pump manufacturer to General Electric and by General Electric to Stone & Webster. We also found consistency in the information General Electric plans to use for the final plant safety analysis.

We reviewed selected design changes and plant features unique to River Bend Station. We reviewed two design changes in progress at General Electric for the nuclear boiler system. We found the design change process to be well defined and controlled for the changes selected.

We reviewed the General Electric methodology for establishing pump runout and NPSH requirements for both the pump manufacturer and the process flow diagram applicable to River Bend Station. We found the General Electric methodology to be sound and consistent.

For the automatic depressurization system we reviewed the base General Electric design specification and selected data sheets in detail. We reviewed the General Electric methodology, assumptions, and intent of these specifications. We compared this information with that used for the plant safety analyses performed by General Electric. We found that the General Electric specifications were valid and consistent. Although the specifications for the automatic depressurization system pneumatic supply were divided among several documents and were somewhat difficult to follow, they were technically correct.

We checked to determine if General Electric had been responsive to IE Bulletin 80-01, "Operability of ADS Valve Pneumatic Supply." We found that General Electric was aware of this bulletin and had design provisions responsive to concerns discussed in the bulletin.

We reviewed selected variances to the automatic depressurization system design specifications for River Bend station. We found General Electric knowledgeable of the variances and able to technically justify the variances granted. We reviewed environmental qualifications of the safety relief valves and found that General Electric has a program to address this issue.

For the emergency core cooling system analysis we found that General Electric has not completed the River Bend specific analysis. We reviewed the approach being taken by General Electric for this analysis and the inputs to be used for the low pressure coolant injection and automatic depressurization systems. We reviewed selected areas for consistency with the Emergency Procedure Guidelines provided by General Electric. We found no problems with the planned General Electric approach in the areas reviewed.

SUMMARY/CONCLUSION

As discussed above, the design information we reviewed was adequate and consistent indicating a controlled design process. We found that the specifications for the automatic depressurization system pneumatic supply could be written more concisely but were technically correct.

RELATED FORMS:
None

2.7 Lessons Learned

The objective of this portion of the inspection was to evaluate the need for design modifications to River Bend Station based upon lessons learned from other construction sites or operating plants.

EVALUATION

We evaluated lessons learned by inspecting the Stone & Webster and Gulf States Utilities programs for response to NRC Bulletins and Information Notices. We selected NRC Bulletins and Information Notices having either general application to the River Bend Station design process or specific application to the automatic depressuration and low pressure coolant injection system designs.

We found that Stone & Webster has established a problem reporting system for the review of construction and operating plant experience and the resolution of problems which may affect Stone & Webster projects. NRC Bulletins and Information Notices, as well as other sources, such as Institute of Nuclear Power Operation Reports, are reviewed by Stone & Webster in Boston, and potential problem areas are identified in Interim Problem Reports issued by Engineering Assurance to the applicable projects for resolution. We found that Gulf States Utilities has established a program independent of the Stone & Webster program for review and resolution of lessons learned. We found that both the Stone & Webster and Gulf States Utilities programs were responsive to the Bulletins and Information Notices selected except for two cases which did not indicate any systematic weakness; IE Information Notice 83-26, "Failure of Safety Relief Valve Discharge Line Vacuum Breakers" (see Deficiency D2.7-1), and IE Bulletin 80-01, "Operability of ADS Valve Pneumatic Supply" (see Deficiency D2.7-2).

For IE Information Notice 83-26, we found that an Interim Problem Report had been prepared by Stone & Webster Engineering Assurance but that the requested response for the River Bend Project had not been provided. Stone & Webster and Gulf States Utilities had addressed the issue of safety relief valve discharge line vacuum breaker performance during the design process, but had not responded directly to failure data in Information Notice 83-26. After the inspection cutoff date, both Gulf States Utilities and Stone & Webster evaluated part of this failure data (GPE valve failures but not Anderson Greenwood valve failures) and identified features of their valve design (Velan valves) which reduce the likelihood of failure. Stone & Webster and Gulf States Utilities were not fully responsive to IE Information Notice 83-26. They had not provided detailed analysis to demonstrate that the Velan vacuum breaker design and application are adequate for River Bend Station.

For IE Bulletin 80-01, we found that Stone & Webster Engineering Assurance issued an Interim Problem Report to the River Bend Project for information only and requested no response. Stone & Webster Power Division had no record of actions taken on this bulletin for the River Bend Project.

The NRC had not issued IE Bulletin 80-01 to plants under construction, but Gulf States Utilities had obtained a copy of it from an information service. Gulf States Utilities had reviewed the bulletin but had taken no action on it. This issue was discussed with the NRC Resident Inspector - Operations for River Bend Station who had also reviewed the Gulf States Utilities response to IE Bulletins and Information Notices. He stated that all NRC Bulletins will be examined for applicability and appropriate resolution, whether they were sent to Gulf States Utilities or not.

Both Stone & Webster and Gulf States Utilities will have to evaluate IE Bulletin 80-01 and ensure appropriate leak testing and check valve application for the automatic depressuration system accumulators.

SUMMARY/CONCLUSION

In summary, we found established programs at Gulf States Utilities and Stone and Webster for ensuring that lessons learned are factored into the River Bend design. We consider the two deficiencies to indicate minor isolated weaknesses rather than an indication of systematic program breakdown. We conclude that the Gulf States Utilities and Stone & Webster programs are adequate and that recent changes in the Gulf States Utilities program will result in future improvements.

RELATED FORMS:
D2.7-1 (Deficiency) Interim Problem Report
D2.7-2 (Deficiency) IE Bulletin 80-01

2.8 High Pressure Core Spray System Design

OBJECTIVE

Although the high pressure core spray system was not identified as one of the systems to be inspected, the design of the pump suction was reviewed due to its uniqueness. The objective of the inspection was to determine if adequate NPSH available is maintained during the transfer from the condensate storage tank to the suppression pool.

EVALUATION

The preferred source of water is the condensate storage tank which is not Seismic Category 1. The backup source is the suppression pool which is Seismic Category 1. The transfer to the suppression pool is automatic and initiated upon loss of condensate storage tank static head in the suction line of the high pressure core spray pump. This is accomplished by opening valve VF015 and closing valve VF001. The sequenceing of these valves was reviewed and found to be correct. The NPSH available during the transfer was also reviewed and no problems were found. The calculation performed to determine the NPSH available during the transfer was conservative and met the stated objectives.

CONCLUSION

This limited inspection of the high pressure core spray system did not reveal any significant concerns.

RELATED FORMS:
None

3.0 MECHANICAL COMPONENTS

The objective of this portion of the inspection was to evaluate the design process in the mechanical components area with emphasis on the preparation, communication, use, and control of design information used in component evaluations. This review included the licensee, Gulf States Utilities and the architect-engineer, Stone & Webster Engineering Company.

3.1 Design Information

Stone & Webster controls the analysis, design, procurement, fabrication and erection of the piping, supports and equipment for the River Bend Station residual heat removal system in accordance with applicable subsections of the FSAR, and by a series of specifications, procedures and drawings.

Stone & Webster Power Division initiates the analysis and design cycle for ASME III large bore pipe by preparing FSK-series schematic drawings which depict the major piping and equipment for the residual heat removal system. These FSK-series drawings are derived from the residual heat removal system definition provided by General Electric. The Power Division prepares EP-series drawings derived from the FSK-series which detail the geometry of the ASME III pipe and the locations of valves and other equipment. Piping 2" and smaller is shown separately on small bore isometric drawings. The Power Division is also responsible for procurement of major equipment such as valves, pumps and heat exchangers. The Power Division specifies the governing loads and load combinations to be employed in the stress analysis of the ASME III pipe detailed on the EP-series pipe drawings. The Power Division also specifies equipment geometry and mass distribution, maximum allowable valve accelerations and maximum allowable equipment nozzle loads.

The Stone & Webster Engineering Mechanics Division performs an initial stress analysis of the piping detailed on the EP-series drawings. The Power Division then prepares an issue-for-construction set of EP-series pipe drawings. Engineering Mechanics Division prepares an issue-for-construction set of BZ-series pipe support drawings.

The Stone & Webster River Bend Station Site Engineering group prepares the drawings used to fabricate and erect large bore pipe and supports. The Site Engineering Group prepares ASME control drawings for large bore pipe from the issue-for-construction EP-series pipe drawings prepared by the Power Division and from the piping fabrication drawings prepared by B. F. Shaw. Site Engineering Group also prepares ASME control drawings for large bore pipe supports from the issue-for-construction BZ-series drawings prepared by Engineering Mechanics Division. Site Engineering Group performs an as-built verification of ASME III pipe and supports to note any out-of-tolerance deviations with respect to the ASME control drawings. This information is then provided to Engineering Mechanics Division which performs any additional stress analysis required to complete the as-built verification (see Detail A3.1-1).

RELATED FORMS:
A3.1-1 (Detail) Drawing Control
A3.1-2 (Detail) References for Section 3

3.2 Personnel and Guidance

This section discusses the training, available guidance and assignments for engineering personnel contributing to the mechanical components area of the River Bend Project.

EVALUATION

Training, guidance and assignments of selected personnel from the Engineering Mechanics Division were reviewed. The procedures and guidance manuals available to division personnel were reviewed as part of the review of design and analysis of specific components. Review of the assignment of personnel was limited to a sample of individuals whose work products were reviewed as part of this inspection.

Training was provided relative to administrative and quality assurance procedures. Records are kept on data files for this training and can be sorted by individual or training course. Review of records for selected personnel showed extensive participation by some in such courses. Formal technical training courses are provided for selected problem areas of component design or evaluation such as component supports.

Procedures provide guidance and requirements for training of personnel for assignment to specific functions. Such training requirements are primarily in the quality assurance area. Those personnel assignments to River Bend Project which were reviewed satisfy these requirements for specific training.

Classroom training of entry level personnel appears to be intensive for their first year. Records showed intensive participation in training programs in 1983 for new staff. Records also showed that some staff who have been with Stone & Webster Engineering Corporation for two or three years have not had this same intensive classroom training. On the job training is provided by senior level personnel guiding specific work assignments (see Detail A3.2-1).

Senior personnel assigned to the River Bend Project appear to be well qualified as to formal training and in-house training. In-house training appeared to be less intensive among personnel with four to five years experience.

Procedures are provided in Engineering Mechanics Division manuals which specify requirements for engineering quality assurance, technical guidance and procedures to assure that interface requirements are defined and satisfied. These appear to be comprehensive, but detailed guidance on review of vendor documents was not found.

Document reviews are discussed in Detail A3.2-2.

RELATED FORMS:
A3.2-1 (Detail) Stone & Webster Training and Assignments
A3.2-2 (Detail) Stone & Webster Technical Procedures

3.3 Piping Stress Inputs

OBJECTIVE

The objective of this review was to evaluate the input information used to perform pipe stress analysis for the residual heat removal system in the low pressure coolant injection mode and for the automatic depressurization system safety/relief valve discharge lines. The review was directed at the process of assuring that all necessary information is controlled, transmitted, recognized, considered, and incorporated in the analysis with recognition of feedback requirements and the need for verification of preliminary information.

EVALUATION

Nine stress analysis packages were selected for review:

(1) Six residual heat removal system stress packages used for the low pressure coolant injection mode.
(2) Two of sixteen safety/relief valve discharge lines extending from the upper anchor to the quencher base. The lines with the shortest and the longest paths were selected.
(3) The main steam Loop C which included part of the safety/relief valve discharge lines from six of the safety/relief valve outlet flanges to the first anchor.

The team reviewed the following: input data, control of the source of data, transmittal records of that data, assumptions used in the analysis, the record of commitment to validate assumptions, evaluation of the resulting stresses and loads and deflections for compliance with limits and licensing commitments, Stone & Webster procedural requirements, and the transmittal of output information. Input information included acceleration response spectra and equipment characteristics including weight, size, flexural characteristics, and interface requirements.

At the time that these nine analyses were performed, the process of providing input to stress analysis was informal and relied on the systems engineer and the responsible engineer to recognize the stress engineer's need for data and to transmit all such data. No response was required if all limits were met. This resulted in no need for subsequent consideration by the systems engineer to determine if inputs were adequately understood by the pipe stress engineer.

These informal procedures appear to serve adequately with two observed exceptions. In one case, the limit on allowable movement of the ball joints which should have been considered by the stress analysis engineer was not specified in the transmittal. This limit also was affected by the amount of misalignment of the joint allowed during erection of the piping. The pipe stress analysis group should have been informed of the overall limit and the amount of misalignment which was permitted (see Deficiency D3.3-1). The analyst had the overall limit of +/- 7.5° rotation and should have independently verified that this limit was not exceeded. In the second case, there was no official transmittal of limits on the loading from piping to be applied to the quenchers at the bottom of the safety/relief valve discharge lines. This second item did not result in any problem since the pipe stress analysis engineer recognized the need for limits on the pipe loadings and handled the problem adequately (see Detail A3.3-1).

The pipe support group was relied on to provide information to the pipe stress engineer on the mass of pipe supports to be considered as part of the pipe system mass in the dynamic analysis. This added mass was generally overlooked in this process (see Deficiency D3.4-6 and Unresolved Item U3.4-1). Improved procedures were developed before the start of this inspection. These recognize the need to assure communication between the responsible engineer for the equipment and the pipe stress engineer.

Documented recognition of the need for the review of stated assumptions and any need for verification were generally adequate but the inspection revealed some instances of inadequate recognition by stress engineers and reviewers of the need to flag areas where verification of assumptions is needed. Examples were found where the need for special consideration of piping stress inputs was not recognized and components were not adequately modeled or considered for evaluation.

The weight of the flow meter in the residual heat removal piping was stated as an assumption for an input to the stress analysis. There was no indication that this assumption should be verified as required by procedures (see Deficiency D3.3-2).

It was not recognized that there was a limit on the available movement of the ball joints. The results of the pipe stress calculations were not evaluated to determine if the predicted movement of these joints were within the limits (see Deficiency D3.3-1). The weight, length and stiffness factors for the ball joints were assumed as an input to the analysis. There was no indication that the assumptions for weight and length should be verified. The stiffness against rotation was assumed to be the same for the three major axes which is not theoretically correct. There was no indication that stiffness was to be confirmed by soundly based or test data (see Deficiency D3.4-2).

Limits on loadings to be imposed on the nozzles of the safety/relief valves were transmitted to the pipe stress analysis group. However, the results of the pipe stress calculations were not evaluated to determine if the limits had been exceeded (see Deficiency D3.3-3).

The inadequacies relating to the ball joints and the safety/relief valve nozzle loadings are considered to be potentially significant. Initial efforts for resolution conducted during the inspection showed that allowable limits may be exceeded.

Input of valve weights was adequate and, as required by Stone & Webster Procedures, transmittal of valve accelerations was adequate (see Detail A3.6-2) except for one minor case in which analytical results from two different valves were interchanged in the transmittal (see Deficiency D3.3-4).

SUMMARY/CONCLUSION

In summary the input information for the review sample was adequately handled for normal design features. However, substantial problems were found where special consideration was required. Significant deficiencies were found in the handling of input data for the ball joints and the safety/relief valve nozzles. These deficiencies can be attributed in part to procedures which existed at the time the calculations were performed but which have recently been improved. In the team's judgement, newer procedures provide improved safeguards against systematic recurrence of such deficiencies during the as built verification.

RELATED FORMS:
A3.1-2 (Detail) References for Section 3
A3.3-1 (Detail) Quencher Inputs and Interface
D3.3-1 (Deficiency) Control of Ball Joint Rotation
D3.3-2 (Deficiency) Flow Meter Weight
D3.3-3 (Deficiency) Safety Relief Valves Nozzle Loads
D3.3-4 (Deficiency) Transmittal of Valve Acceleration Data

3.4 <u>Piping Stress Procedures</u>

OBJECTIVE

The objective of this review was to evaluate the modeling procedures employed by the Stone & Webster Engineering Mechanics Division to stress analyze the River Bend Station residual heat removal system.

EVALUATION

As noted in Table 3.2-1 of the River Bend Station FSAR, the Residual Heat Removal System is classified as a Seismic Category I, Quality Assurance Category B piping system, Safety Class 1 within and Safety Class 2 beyond the outermost isolation valves. This classification is equivalent to ASME Section III Code Classes 1 and 2 (see also FSAR Tables 3.2-2, 3). Design requirements for Safety Class 1 and 2 piping and supports are tabulated in FSAR Tables 3.2-4, 5 for the normal, upset, emergency and faulted design condition categories.

Seismic Category I ASME Code Class 1, 2 and 3 pipe and pipe supports for the Stone & Webster balance of plant scope of work are analyzed and designed in accordance with the requirements of the 1974 Edition of ASME Section III, with exceptions as noted in FSAR Subsection 3.9.1.4.2A. That subsection also commits to comply with the functional capability requirements detailed in NEDO-21985. Load combinations and stress limits for the stress analysis of ASME Code Class 1, 2 and 3 pipe are given in FSAR Tables 3.9A-2, 3. Load conditions for pipe supports are defined in FSAR Table 3.9A-13.

FSAR Table 3.9A-4 mandates computer analysis for all Class 1 pipe having a nominal pipe size greater than 1" and all Class 2, 3 pipe having a nominal pipe size greater than 6". Engineering Mechanics Division uses the computer code NUPIPE for this work.

Engineering Mechanics Division pipe stress engineers perform pipe stress analysis in accordance with a series of in-house technical procedures in addition to project specific technical specifications such as the piping design, pipe support, and snubber design specifications.

Loop B of the low pressure coolant injection mode of the residual heat removal system was specifically evaluated for this review. The piping geometry for this subsystem is detailed on the EP-series drawings originally issued for stress analysis. Support types, locations and restraint orientations are detailed on individual BZ-series drawings. Support locations and designations are also shown for information only on EZ-series drawings, which are a composite of the EP-series piping drawings and the BZ-series pipe support drawings. The standard line designation table provides additional information concerning pipe wall and insulation thicknesses and operating and design pressures and temperatures. Pipe material properties are defined by the pipe classes noted on the EP-series drawings and the corresponding material specifications detailed in the piping specification.

Governing loads and load combinations for pipe stress analysis as initially defined in the FSAR are elaborated and updated by Engineering Mechanics Division. The Stone & Webster Power Division derives and transmits all load data to Engineering Mechanics Division with the exception of amplified response spectra and seismic anchor displacements, which the Structural Division provides.

Loop B of the low pressure coolant injection mode of the residual heat removal system was broken down into six AX-series subsystems for stress analysis. These subsystems are decoupled by establishing subsystem boundaries at anchor points and equipment nozzles. Each AX-series stress package was sampled to assess the pipe stress analysis modeling procedures employed by Engineering Mechanics Division with respect to input data, in-house technical procedures and specifications, and FSAR commitments (see Detail A3.4-1).

Line E and Line H of the safety/relief valve discharge lines and main steam line C were reviewed as part of the automatic depressurization system. In general the Stone & Webster procedures were followed for customary design features. Two exceptions were noted for unique features. Modeling of General Electric furnished valves in the analysis of main steam line C was not in accordance with Stone & Webster internal procedures (see Deficiency D3.4-1). Modeling of the ball joints in the safety relief valves discharge lines branching from main stream line C contained significant errors in geometry and the flexural characteristics of the ball joints were not treated in enough detail to yield acceptable results (see Deficiency D3.4-2).

The team identified the following additional deficiencies during the course of this review: a dimensional discrepancy in a pipe stress package (see Deficiency D3.4-3); a discrepancy between the as-built and calculated support stiffness for pipe restraints attached to structural steel (see Deficiency D3.4-4); a pipe stress analysis using unconservative damping values for input seismic response spectra (see Deficiency D3.4-5); failure to implement a Stone & Webster procedure to evaluate added mass for trapeze hangers (see Deficiency D3.4-6); and failure to address functionality criteria (see Deficiency D3.4-7). The team identified a concern regarding handling of added mass for trapeze hangers in the Nuclear Steam Supply System scope of work (see Unresolved Item U3.4-1).

Two deficiencies related to document control were also noted during this inspection. The first deficiency involved a violation of the time limits allowed to process outstanding Engineering and Design Coordination Reports and Nonconformance and Dispositions (see Deficiency D3.4-8). The second deficiency involved a failure to control interoffice correspondence containing technical information (see Deficiency D3.4-9).

SUMMARY/CONCLUSION

Many of these deficiencies are considered minor to moderate in significance, and can be addressed by Engineering Mechanics Division during the final stress analysis review/calculation that will be performed to verify the design adequacy of the as-built piping configuration. However, the deficiencies in the modeling of the ball joints require prompt reanalysis. Details of proper geometry and expert review of how the non-linear characteristics are to be treated in the analysis should be incorporated in order to determine the impact of these deficiencies.

In summary the team confirms the general adequacy of the pipe stress analysis modeling procedures employed by the Stone & Webster Engineering Mechanics Division.

RELATED FORMS:
A3.1-2 (Detail) References for Section 3
A3.4-1 (Detail) Stone & Webster Modeling Procedures
D3.4-1 (Deficiency) Valve Modeling
D3.4-2 (Deficiency) Ball Joint Modeling
D3.4-3 (Deficiency) Dimensional Discrepancy
D3.4-4 (Deficiency) Pipe Support Stiffness
D3.4-5 (Deficiency) Percent Critical Damping
D3.4-6 (Deficiency) Added Mass for Trapeze Hangers (BOP)
D3.4-7 (Deficiency) Pipe Functionality Criteria
D3.4-8 (Deficiency) Time Limits on Drawing Revisions
D3.4-9 (Deficiency) Interoffice Correspondence Control
U3.4-1 (Unresolved Item) Added Mass for Trapeze Hangers

3.5 Pipe Supports

OBJECTIVE

The objective of this review was to evaluate the modeling procedures employed by the Stone & Webster Engineering Mechanics Division to analyze and design the pipe supports for the River Bend Station residual heat removal system.

EVALUATION

The pipe supports for Loop B of the low pressure coolant injection mode of the residual heat removal system were selected for this evaluation. As detailed in Section 3.4, Piping Stress Procedures, Loop B of the low pressure coolant injection mode of the residual heat removal system is an ASME Class 1, 2 piping subsystem. Design requirements for Safety Class 1, 2 piping and supports are tabulated in FSAR tables 3.2-4 and 5 for the normal, upset, emergency and faulted design condition categories. Design, fabrication and erection of pipe restraints and anchors is governed by Stone & Webster specifications for the design and fabrication of power plant piping supports and field fabriction and erection of pipe supports. Design and fabrication of mechanical snubbers is separately controlled by Stone & Webster specification. ASME Boiler and Pressure Vessel Code, Section III, subsection NF jurisdictional boundaries for River Bend Station are defined in FSAR Subsection 3.9.3.4.1.2A and elaborated in a Stone & Webster specification.

The formal pipe support design effort proceeds from the support reactions prepared by the initial pipe stress analysis. This analysis is based upon a preliminary definition of the support types, locations and orientations. Stone & Webster is responsible for the engineering and design of all ASME III Code Class 1, 2, 3 and certain Class 4 (B31.1) pipe supports, with the exception of component standard supports.

Supports and snubbers are designed to dual stress and stiffness requirements. The initial pipe support may be designed solely to stiffness criteria, with load qualification to be performed as a part of the final as-built verification program. The as-built program for design verification of ASME III piping and pipe supports is detailed in Stone & Webster Engineering and Design Coordination Report C-13, which is to be incorporated into the Stone & Webster specification for field fabrication of piping supports.

An area of related interest to the team was the Stone & Webster implementation of FSAR Subsection 3.7.3.4.1A. This subsection specifies relative mass and frequency criteria to be employed to determine whether equipment is uncoupled dynamically from its supports. Equipment and supporting structure meeting these criteria enable the rational application of amplified response spectra, which presupposes no dynamic coupling.

SUMMARY/CONCLUSION

The team confirms the adequacy of the modeling procedures employed by the Stone & Webster Engineering Mechanics Division to design pipe supports. The team identified the following deficiencies during the course of this review: a missing key plan dimension on a pipe support control drawing (see Deficiency D3.5-1); a span between adjacent supports for a run of small-bore seismic pipe which exceeded the maximum allowable span length (see Deficiency D3.5-2); and failure to implement FSAR Subsection 3.7.3.4.1A, which addresses coupling effects to be considered in the analysis of supported components (see Deficiency D3.5-3). Although some of these deficiencies are considered to be significant, they can be addressed by Engineering Mechanics Division during the final stress analysis cycle performed to verify the design adequacy of the as-built support configuration.

RELATED FORMS:
A3.1-2 (Detail) References for Section 3
D3.5-1 (Deficiency) Incomplete Pipe Support Location Plan
D3.5-2 (Deficiency) Small-Bore Seismic Piping Maximum Support Spans
D3.5-3 (Deficiency) Dynamic Coupling

3.6 Mechanical Equipment

OBJECTIVE

The objective of this portion of the inspection was to evaluate the design adequacy of specific Seismic Category 1 components and the control, review, and approval performed by Stone & Webster Engineering of both vendor and in house equipment design analysis reports.

EVALUATION

Four pieces of mechanical equipment were selected for detailed review. One of these was in the General Electric scope of supply, Pump C002B in residual heat removal Loop B. The remaining three were in the Stone & Webster scope of supply: (1) Motor operated valve F042B in the residual heat removal Loop B, (2) Ball joints in the safety/relief valve discharge lines, and (3) Vacuum breaker valves F078 and F037 in the safety/relief related valve discharge lines. The team reviewed the procedures for establishing and specifying requirements for equipment design and qualification and for reviewing vendor reports and the controls on information transfer.

Procedures for establishing seismic qualification requirements were reviewed. The term seismic qualification is used to include qualification for all dynamic loadings, including the hydrodynamic loadings from the safety/relief valve openings. The equipment qualification program was not sufficiently developed to allow detailed review for components subjected to hydrodynamic loads. Because qualification of mechanical equipment was an open item in the NRC Safety Evaluation Report and will be subject to a later audit, the existing reports were not reviewed in depth.

The design reports from the manufacturers for all four pieces of equipment were reviewed by Stone & Webster or General Electric and no reports were disapproved by them. The procedures appear to be adequate for equipment design for pressure boundary integrity and strength. The review found few exceptions. There was a minor inconsistency in the specification of the effective ASME Code date for the vacuum breaker valve (see Deficiency D3.6-1), and there were minor omissions of the use of the corrosion allowance in the pump design (see Detail A3.6-1).

The procedure for establishing seismic qualification requirements appears adequate for equipment requiring seismic and dynamic qualification (see Detail A3.6-2). Certain equipment, e.g. the vacuum breaker valves and ball joints, may be judged as not requiring seismic qualification, but may require functional qualification. Questions related to operability performance of the vacuum breaker valves had developed prior to this inspection but were resolved to the satisfaction of Gulf States and Stone & Webster (see Detail A3.6-3). The ball joint specification requires a substantial functional performance qualification effort but does not require seismic qualification. The qualification reports on the ball joints were provided by the supplier and were reviewed and approved by Stone & Webster. Significant deficiencies were found in the quality of this review in that some tests did not meet specifications and there was no apparent reconciliation of the differences (see Deficiency D3.6-2).

Industry wide delays in recognizing and resolving the problem of hydrodynamic loadings in the suppression pool have caused significant problems for the seismic/dynamic qualification program of equipment inside the containment. Past practice based on procurement of equipment which had been previously qualified for seismic conditions has had to be reconsidered. Stone & Webster has recognized these problems and has had a program underway for about two years which may be able to meet commitments. It may be demonstrated that present hardware has sufficient margins over past requirements to be able to satisfy new requirements. Such demonstrations of margin require thorough understanding of equipment failure mechanisms.

In an area where Stone & Webster internal engineering practice forms the basis for requirements, the Stone & Webster program appeared to be not well controlled. Requirements appear to be established on the basis of the judgment of the staff directly involved in design evaluation and procurement. One area where they have not developed experienced judgment is the qualification and application of ball joints in piping systems. This particular problem area does not appear to have been managed satisfactorily to date. A similar area is in the initial performance requirements specified for the vacuum breaker valves which had to meet requirements established by General Electric. This now appears to be adequately handled.

SUMMARY AND CONCLUSION

Stone & Webster is making a strong effort to meet established NRC requirements for qualification of mechanical equipment. Emphasis is directed at components for which NRC qualification requirements have been established. The delays in resolving hydrodynamic loading problems in the face of a tight schedule result in some risk taking with the equipment already procured. Proposed procedures and the basis for extending previous qualification results should be critically evaluated.

RELATED FORMS:
A3.6-1 (Detail) Pump Design and Qualification
A3.6-2 (Detail) Valve Qualification Program
A3.6-3 (Detail) Vacuum Breaker Valve Design and Qualification
D3.6-1 (Deficiency) ASME Code Edition for Valves
D3.6-2 (Deficiency) Ball Joint Qualification

4.0 CIVIL AND STRUCTURAL

The objectives of this portion of the integrated design inspection were to evaluate the civil and structural engineering practices and technical execution of the design with specific emphasis on control and exchange of information within the project. The team inspected areas to determine whether:

(1) Regulatory requirements and design bases as specified in the license application have been correctly translated and satisfied as part of specifications, drawings, and procedures.

(2) Correct design information has been provided both internally and externally to the responsible design organizations.

(3) Design engineers had sufficient technical guidance and/or experience to perform assigned engineering evaluations, and

(4) Design controls applied to the original design have also been applied to design changes, including field changes

These objectives were accomplished by selecting a sample of structural elements which make up the building structures or are supporting mechanical, electrical, and instrumentation and control systems being reviewed by team members.

In addition, the review included a subcontractor, Reactor Controls, Inc. This firm performed analysis, engineering and installation of the control rod drive hydraulic system and associated piping supports.

4.1 Design Information

The objectives of this portion of the inspection were to evaluate, based on specific samples, how the basic structural design criteria taken as input from such sources as the NRC Regulations, the General Design Criteria of 10 CFR Part 50, Regulatory Guides, Branch Technical Positions and industry codes and standards have been incorporated into design documents and design and quality control procedures. FSAR commitments relating to the structural design effort with the associated basic structures and the residual heat removal system were selected for review. The involvement of Gulf States Utilities in the major design effort delegated to Stone & Webster was evaluated. Stone & Webster was the major design organization reviewed during this inspection. Interfaces and information flow between various organizations involved in the design were also examined and evaluated in order to assess the design control mechanisms as well as the final design as evidenced by construction documents and the completed physical plant.

Inspection of the interdisciplinary coordination effort (see Figure F4.1-1) disclosed that design information from various groups such as the Stone & Webster Engineering Mechanics Division, Geotechnical Division, and Power Division are transmitted to the Structural Division. This information is used for the design and proportioning of structural members. The Structural Division Staff Group generates the amplified response spectra and displacements which are forwarded to the various groups. This information is used for the purpose of pipe stress analysis and equipment qualification.

The Structural Division independently receives information regarding cable trays, conduit routings, etc. in the form of electrical drawings. From this basis they prepare cable tray support details which are routed to the Electrical Division for verification and approval. Finally, the Structural Division is responsible for the preparation of structural specifications, calculations and drawings. These are sent to fabricators and subcontractors for detailing and construction.

Interfacing between various disciplines with regard to the development of seismic loads was examined (see Detail A4.1-2). The Geotechnical Division develops various design parameters such as soil properties and the Geotechnical Design Criteria. This information is coupled with other design parameters from various divisions including the Structural Division. The Structural Division Staff Group performs the seismic analysis and distributes the information to various divisions. The amplified response spectra and displacements are transmitted to the Engineering Mechanics Division which performs pipe stress and equipment qualification analyses. The forces, bending moments and displacements are forwarded to the Structural Division Project Group which uses this information to perform the structural analysis and design. Results of the seismic analysis are also forwarded to General Electric for the seismic qualification of the nuclear steam supply system.

The team also evaluated the processing of hydrodynamic loads. In this case General Electric developed the loads which are described in the "Containment Loads Report". This information together with the soil properties, pertinent equipment information from the Engineering Mechanics and Power Divisions, as well as structural steel and concrete drawings, is transmitted to the Structural Division Staff Group which performs the dynamic analysis. Distribution of the results is identical to that for seismic loads as described above.

The basic document used in the Structural Division for the design of the foundation mat and the structural concrete fill in the composite section of the containment was the ASME Boiler and Pressure Vessel Code, Section III, Division 2. For other reinforced concrete structures, the ACI building code, ACI 318-71, was used. For steel structures, the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, was used. These commitments are contained in Sections 3.8.3.2 and 3.8.4.2 of the FSAR.

Design information developed by Stone & Webster from the commitments in the FSAR was reviewed by the team. The basic documents addressing the work by the Structural Division and the Geotechnical Division were the Structural Design Criteria and the Geotechnical Design Criteria. The requirements of both criteria documents were implemented in the design.

During the project work a reorganization took place in which the Structural Mechanics portion of the Engineering Mechanics Division was transferred to the Structural Division. The Structural Design Criteria cover only the responsibilities of the structural portion of this newly formed group. The structural design work had been completed prior to the reorganization.

The team concluded that the criteria documents reflected the FSAR commitments and were implemented in the design with few exceptions.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
A4.1-2 (Detail) Stone & Webster Interdisciplinary Flow

4.2 Personnel and Guidance

This section describes the qualifications and guidance information related to Stone and Webster personnel.

The Structural Group for the River Bend Project at Cherry Hill consists of forty five engineers and draftsmen who are divided into four subgroups. Sixteen persons are working in the area of design of the Reactor Building, fourteen on Category I structures other than containment, eleven in the design and drafting section and three as material specialists and handling special problems. The Lead Structural Engineer reports to the Assistant Project Engineer for project matters. Several assistants known as Principal Engineers are responsible for various technical areas and report to the Lead Structural Engineer. Reporting to the Principal Engineers are Group Leaders with well defined specific responsibilities. The Lead Structural Engineer is a professional engineer with seventeen years of professional experience, with eleven of those years on the River Bend Project.

The team interviewed ten members of the Stone & Webster staff at the Cherry Hill office and two at the plant site. All of the persons contacted, with one exception, have professional engineering licenses. Their average length of engineering experience is fifteen years, of which ten years is nuclear. Most of the engineers interviewed have advanced degrees from various universities. In general, the structural engineers with important responsibilities were assigned to River Bend Project in the early stages of the project.

The Geotechnical Division for the River Bend Project at Cherry Hill consists of a Lead Geotechnical Engineer who is a professional engineer and has ten years of professional experience with eight years on the River Bend Project. Inasmuch as geotechnical activities have nearly ceased, the division at this time consists of the Lead Geotechnical Engineer with assistance from the former Lead Geotechnical Engineer.

At the River Bend site the structural group is assigned field responsibilities. The site engineers communicate with the Cherry Hill office extensively.

Gulf States Utilities maintains a resident engineer at the Cherry Hill Operations Center. His responsibility is to coordinate the design efforts with Gulf States Utilities. He has a professional engineering license and has been in the engineering profession for fourteen years, of which five is nuclear.

Numerous guidelines, reports and procedures are available for use by the Stone & Webster River Bend Project. Some of these were reviewed by the team. A few examples are the Design Procedure for Category I Conduit Systems for River Bend Project, Seismic Design of Conduit Support Systems, and Technical Guideline No. STG 19.4.0. The Stone & Webster corporate procedure for performing calculations is Engineering Assurance Procedure 5.3. The Structural Division has expanded upon Engineering Assurance Procedure 5.3 and has implemented Structural Technical Procedure 11.5. These procedures were reviewed by the team during the inspection. Specific findings with their implementation are addressed later in this section.

Stone & Webster engineering personnel attend certain periodic training courses. Some of these training courses are: Engineering & Design Process Overview; Engineering Assurance Procedure 2.10 Licensing Document Changes; Quality Assurance Program - Engineering & Design Overview; Engineering Assurance Procedure 3.13 Control Unapproved Documents; Engineering Assurance Procedure 6.1 Document Control; Engineering

Assurance Procedure 15.1/15.2 Nonconformance & Disposition Reports; Engineering Assurance Procedure 16.1 Problem Reports; ASME III Engineering Overview; and, Engineering Assurance Procedure 6.3 Engineer and Design Coordination Reports. While the team did not review the content of the courses, the establishment of such a program is beneficial in the team's judgment.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.3 Seismic Analysis

OBJECTIVE

The objective of this evaluation was to determine whether the seismic analyses of the Reactor Building and the Auxiliary Building were performed in accordance with the commitments in the FSAR.

EVALUATION

The evaluation extended from the establishment of the mathematical models to the development of the amplified response spectra curves. The review of the Reactor Building seismic analysis included two calculations. The first calculation consisted of the mass and stiffness properties. The second calculation was performed to analyze the effect of the concrete fill between the shield building and the steel vessel. For the Auxiliary Building only one set of calculation exists. This calculation was also reviewed by the team.

The artificial earthquake motions (2 horizontal and 1 vertical) developed enveloped the provisions of Regulatory Guide 1.60.

The development of the mathematical model for the Reactor Building, including the mass and stiffness properties was reviewed (see Detail A4.3-1). The structural drawings were compared to the model that was used in the analysis. The review of the mass calculations showed one minor discrepancy in that the input and output data were not contained in the calculation (see Deficiency D4.3-1). In the stiffness properties one member had a slightly different stiffness than the actual design (see Deficiency D4.3-2). None of these items was considered significant with respect to design control. Overall, the mass and stiffness properties reflected the actual design parameters and configurations.

The Auxiliary Building seismic model consists of a lumped mass model with soil springs attached to it. The development of the mass and stiffness properties included manual and computer calculations.

To calculate the masses, dead load and heavy equipment were considered. A live load of 25 psf was also taken. Center of mass and rigidity for each mass point were calculated and incorporated into the model. Spot checks showed that the mass and stiffness properties reflected the actual design and construction.

The subgrade was represented by horizontal, vertical, torsional and rocking springs. These elastic springs were computed for three different soil shear moduli; 12, 14 and 24 ksi. For the Reactor Building, the structural design and the development of the amplified response spectra curves were done using the envelope of these three soil cases. In the Auxiliary Building structural design, the same enveloping was performed.

The amplified response spectra, though, was developed using only one soil spring with a shear modulus of 18 ksi. Such a procedure is consistent with the FSAR requirements. The soil damping was limited to ten percent critical for both Operating Basis Earthquake and Safe Shutdown Earthquake, except in the Auxiliary Building a more conservative value of five percent critical damping was used for rocking and torsion.

The seismic analysis was performed for both the Operating Basis Earthquake and the Safe Shutdown Earthquake. Structural damping values used for each component agreed with Regulatory Guide 1.61. The model responses were combined in accordance with Regulatory Guide 1.92. The effects of three earthquake motions (2 horizontal and 1 vertical) were considered in obtaining the final response.

Amplified response spectra curves were developed for both buildings (see Detail 4.3-2). Torsional effects may be significant when the center of mass does not coincide with the center of rotation of the structure. The horizontal component of torsional motion is most pronounced at the periphery of the structure. Similarly, since during a seismic event a structure would rock about its horizontal axis, the vertical component of rocking is most acute at the extreme section of the structure. The horizontal and vertical amplified response spectra did not include the components of torsion and rocking (see Deficiency D4.3-3).

The vertical response at the midspan of a flexible floor system would be greater than at the supports. For the River Bend Project, amplified response spectra for vertical were developed with no consideration of the flexibility of the floor systems. This should be addressed; however, due to the site conditions, it is unlikely to have any effect on the results of the analyses (see Deficiency D4.3-4).

The interface of the different organizations within Stone & Webster which develop and use amplified response spectra curves was evaluated. The results show that such interface is well controlled and the possibility of not using the specified amplified response spectra curves is remote.

SUMMARY/CONCLUSION

The review showed that the artificial earthquake motions developed adhere to the provisions of Regulatory Guide 1.60. The mass and stiffness properties are consistent with the structural design. The amplified response spectra curves developed include a 50% increase in the amplitude of the ground acceleration which makes the amplified response spectra conservative. Although this is true, the horizontal and vertical components of rocking and torsion were not accounted for. The contribution of such modes to the overall amplified response spectra should be evaluated. Overall, the methods used in the seismic analysis of the Reactor and Auxiliary Building were appropriate and the analyses were well controlled.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
A4.3-1 (Detail) Mass and Stiffness Properties of Reactor Building
A4.3-2 (Detail) Development of Amplified Response Spectra Curves
D4.3-1 (Deficiency) Calculation of Lumped Masses
D4.3-2 (Deficiency) Concrete Strength of Drywell
D4.3-3 (Deficiency) Effects of Torsional and Rocking Modes
D4.3-4 (Deficiency) Vertical Frequency of Floor Systems

4.4 Design of Reactor Building Internal Structures

OBJECTIVE

The objective of this portion of the inspection was to examine adequacy of the analysis, consistency between the analysis and engineering and construction drawings and between the analysis and the commitments stated in the FSAR. The internal structures selected for this inspection were the drywell, the primary shield wall, the weir wall, and the reactor pedestal.

EVALUATION

(a) Drywell

The drywell was assumed to be an axisymmetric thin shell. The liner plate was assumed not to contribute to the strength or stiffness of the drywell wall. The top slab was assumed to be cracked and the wall was assumed to be cracked or uncracked, depending on the direction and the loading condition considered.

The drywell was analyzed using Stone & Webster computer program SHELL 1. The member properties were developed based on an assumed degree of cracking in the horizontal and vertical directions. The results of the SHELL 1 analysis were used as an input to another computer program called NEWSECT which converts the forces in steel and concrete calculated by the SHELL 1 into the corresponding stresses. These are then compared with the stresses permitted by the code (see Detail A4.4-1). The standard governing the design was the ASME Boiler and Pressure Vessel Code, Section III, Division 2. The calculations reviewed contained the necessary information to assure traceability of design parameters such as temperature, analytical model, seismic loads, etc. This information was found in the attachments accompanying the calculations and the team judged this to be a convenient way of filing the important parameters of design. We were told that copies of these documents are filed in the central document system. We made a test search for some documents and found that in general, documents issued prior to 1980 were not available in the central files. Documents issued later have been properly filed and were available on a short notice (see Detail A4.4-2).

During review of the drywell calculation we found that there was a discrepancy between the calculation criteria and the drawings in that the concrete strength used in the calculations was 3000 psi instead of 4000 psi as stated on the drawings and in the Structural Design Criteria Section 5.1.3.2. We considered this to be an inconsistency but since the lower value was used in the design calculations the difference is on the conservative side (see Deficiency D4.4-1).

(b) Primary Shield Wall

The primary shield wall is constructed of two steel cylinders joined and stiffened by horizontal and vertical steel web plates. The steel stiffeners are mostly 1-1/2 inch thick. All plates are joined together by continuous full penetration welds. The space between the steel cylinders is filled with nonstructural high density grout. The primary shield wall was analyzed using a finite element, two dimensional model and the STRUDL II computer code. Both rectangular and triangular elements have been used depending on the stress conditions. The steel plates were the only structural material considered in the analysis.

We found evidence that some sections of the calculations were not reviewed for long periods of time (see Observation O4.4-1). There was no listing of pages reviewed to indicate the reviewer when there was more than one reviewer (see Deficiency 4.4-8). This practice opens an opportunity for an error in the calculations such that some pages could remain unreviewed.

While reviewing the shop drawings we noted that a drawing reference to another drawing for a typical wall section was incorrect (see Deficiency D4.4-3).

We reviewed the calculations and noted that the section capacity factor listed in the summary sheets was inconsistent with the calculations and with the regulatory position of the Standard Review Plan. Further inspection of the calculations disclosed that this error was not carried through to the analysis and therefore had no impact on the analysis (see Deficiency D4.4-4).

We reviewed the stresses resulting from the analysis and found that they are within those permitted by the AISC specification.

(c) Weir Wall

The weir wall is a concentric wall 2' 1" in thick and 21' 3" high with a diameter of 59' 10". Its function is to prevent the suppression pool water from entering the interior of the drywell.

The weir wall was analyzed as a thin cylindrical shell using an inhouse Stone & Webster computer program SHELL 1. In the design the structure was subjected to the loads resulting from water and steam pressures, temperature gradients and shears and moments resulting from mat rotations and displacements.

We reviewed the design calculations, the computer input and output of the SHELL 1 program. The design check, which used Stone & Webster computer program NEWSECT, was also reviewed.

During the review a few minor mistakes were found. These mistakes did not affect the design (see Deficiency D4.4-5). We reviewed the load combination equations and found them to be in agreement with the applicant's commitments in Section 3.8.4.3.2 of the FSAR.

(d) Reactor Pedestal

The reactor pedestal supports the primary shield wall and the reactor pressure vessel. It is a reinforced concrete cylindrical structure. The analysis is contained in calculations. The following computer programs were used in the analysis.

F Series, Stone & Webster Computer Program ST-251, Version 0.0, level 01, "Fourier Coefficient" by A. J. Hsi,

STRUDL II, Stone & Webster Program ST-015 Version 10, Level 06, "ICES STRUDL II"

SHELL 1, Stone & Webster Computer Program ST-142, Version 01, Level 06 "Thin Shell of Revolution Under Arbitrary Loadings" by S. L. Coy

NEWSECT, Stone & Webster Computer Program ST-246, Version 00, Level 01 "Reinforced Concrete Section Stress Analysis," by I. H. Tai (see Detail A4.4-3).

The discontinuity moments and shears at the mat pedestal junction were obtained from the mat analysis and used as a bottom boundary condition in the analysis of the pedestal. Forces at the base of the primary shield wall and reactor pressure vessel were used as the top boundary conditions in the analysis of the pedestal. The first analysis was performed by the SHELL 1 code ignoring all of the openings using a cracked and uncracked section. The check for adequacy of the design was done by the NEWSECT computer code. The stresses on the shear bars were checked by hand calculations. The STRUDL II computer code was used for the analysis of the forces and moments around control rod drive openings and the stresses were checked using the NEWSECT code.

We reviewed the load combinations for which the pedestal was analyzed. SHELL 1 computer runs were performed on the four equations after preliminary elimination of some of the load combinations. On the basis of the investigation two equations were selected as the ones producing the highest forces. One of these two, by comparison, was chosen as governing the design. We reviewed the results of the analysis and found that they are within the code allowables.

SUMMARY/CONCLUSION

Our review disclosed that the input information, the assumptions used in the analysis and the methods of the analysis have been based on good engineering practice and on sound theoretical background. The resulting stresses met those which are allowed by the appropriate design codes. The team concluded that the deficiencies found during the inspection of the selected interior structures were not of the type to have adverse effects on the outcome of the analysis.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
A4.4-1 (Detail) Consistency of Computer Programs
A4.4-2 (Detail) Traceability of Design Documents
A4.4-3 (Detail) Verification of Computer Codes
D4.4-1 (Deficiency) Discrepancy Between Calculations and Structural Criteria
D4.4-2 (Deficiency) Identification of Reviewed Pages
D4.4-3 (Deficiency) Mistake in Cross-Reference in Drawings
D4.4-4 (Deficiency) Incorrect Section Capacity Factor
D4.4-5 (Deficiency) Mistake in Review of Weir Wall Calculations
O4.4-1 (Observation) Timeliness of Verification of Design Calculations

4.5 Analysis of Steel Containment

OBJECTIVE

This evaluation was performed to determine whether the analysis of the steel containment was performed in accordance with the FSAR commitments.

EVALUATION

The evaluation included a review of Stone & Webster calculations as well as vendor submittals. The analysis of the steel containment was performed by manual and computer calculations. The manual computations were done in order to prepare input to a computer program (SHELL 1) which analyzed the containment vessel. In some cases, stresses have been computed manually and added to those observed from the SHELL 1 output in order to calculate the final stresses as required by the appropriate load combination equations.

All the input and output of the computer analysis was reviewed. Spot checks showed that stresses obtained from the computer analysis were reasonable. The governing load combination for the design of the containment vessel included the internal pressure and the local loads due to attachments. The overall stresses in the containment vessel were low, with the exception of attachment loads.

All load conditions which effect the analysis of the vessel were reviewed. The mathematical model used for the computer program agreed with the actual design. As noted in the calculation, confirmation is still required for the safety relief valve loads that affect the containment vessel.

The containment vessel was reinforced around the penetrations. W.J. Woolley supplied the major penetrations and performed the finite element stress analysis which the team reviewed. Stress concentrations due to local loads were accounted for.

The buckling analysis of the vessel was performed in accordance with ASME Code Case N-284. The maximum compressive stresses obtained from the vessel analysis were used to calculate the safety factors against buckling. These calculations showed that the safety factors were above 3.0 for normal operating conditions and above 2.0 when the design basis accident was considered. These factors satisfied the NRC Staff's interim criteria.

SUMMARY/CONCLUSION

The design organizations had performed an extensive review of the analysis of the steel containment. The review showed that the stresses in the containment vessel are within the limits of the ASME Boiler and Pressure Vessel Code. Detailed overall and local analysis of the vessel was performed using computers and mathematical models. The overall analysis was performed by Stone & Webster and local analysis was done by the vendor. The work performed by the vendor was reviewed by Stone and Webster responsible engineers. The analysis of the steel containment meets the commitments in the FSAR and the requirements of applicable codes.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.6 Reactor Building Shield Structure

OBJECTIVE

The objective was to determine whether or not the loadings were accurately obtained and used in the analysis and that the analysis and design were consistent with the commitments in the FSAR.

EVALUATION

Two sets of calculations were reviewed, one being the original calculations and the other supplementary calculations. The original calculations were used as the basis for designing the reinforcing steel. The supplementary calculations were used to confirm certain load combinations.

The structural analysis of the shield structure was performed using the computer program SHELL 1. This program is capable of structural analyses of axisymmetric structures subject to either axisymmetric loads or asymmetric loads which can be represented by a Fourier series expansion. Prior to the placement of concrete in the annulus between the containment and the shield structure, the pressures resulting from safety relief valve actuation and LOCA affected the shield structure. The analyses were performed using SHELL 1. Care was taken to ensure analytical compatibility at the interface of the shield structure and the mat.

An analysis was performed to determine reinforcing bar requirements in the concrete placed in the annulus which extends from the top of mat at Elevation 70' up to Elevation 95'. The design is such that the concrete effectively ties together the containment vessel and the reinforced concrete shield structure. The result of this is a composite structure which reacts to loading differently from the individual structures. Revisions to the calculations dated May 29, 1984 indicate that no shear reinforcing was in fact required in the shield structure wall.

The review of the calculations resulted in finding two deficiencies, one technical and the other administrative. The technical deficiency (see Deficiency D4.6-1) concerned the anchorage of the diagonal shear reinforcing bars, the "Z" bars. The design method used for the original design departed from ACI 318-71; however, the need for such shear reinforcing was eliminated by subsequent placement of the concrete as described above and revised transient analyses resulting in lower temperature gradients. The administrative deficiency (see Deficiency D4.6-2) concerned traceability of checkers and reviewers on the calculation cover sheets. No design errors were found which could be attributed to this deficiency, however, potential confusion could exist in determining whether all calculations were properly checked and reviewed.

SUMMARY/CONCLUSION

Although there were two deficiencies noted in the design of the shield structure, they are not considered to affect the as-built condition of the structure. In the controlled condition of the shield structure, no problems in the analysis and design were found which affect the integrity of the structures.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
D4.6-1 (Deficiency) Inadequate Anchorage of Radial Shear Reinforcing
D4.6-2 (Deficiency) Calculation Procedures

4.7 Reactor Building Mat

OBJECTIVE

The objective of the evaluation was to review whether FSAR commitments were met, whether interfaces with the major superstructures (weir wall, the primary shield structure, drywell wall and the steel containment vessel) were consistently handled, and whether the analytical methods represented the structure, supporting soil, and the anticipated loadings.

EVALUATION

The evaluation included a review of the values selected for the soil shear modulii, the appropriateness of computer programs used in the analysis, the structural stiffness of the mat, loadings, methods of calculating stresses, and certain reinforcing bar details.

The analysis of the mat was quite complex in that three different computer programs were used for different loadings; namely, MAT 6, SHELL 1, and GHOSH. All three programs are in-house Stone & Webster programs. MAT 6 is capable of modeling the axisymmetric structures and the soil and is limited to axisymmetric loadings such as dead load and static pressures. The soil is mathematically represented as an elastic continuum loaded on the surface by a rigid plate. SHELL 1 is also restricted to axisymmetric structures but solutions can be obtained for axisymmetric loads which can be represented by Fourier expansions. The soil stiffness was represented by means of Winkler springs (linear vertical springs). For the safety relief valve and LOCA hydrodynamic loads (which are asymmetric and vary with time) the GHOSH program was used which can solve for axisymmetric structures. The soil was represented by finite element rings, however, the analysis was simplified by calculating compliance functions which represent the soil stiffness and damping as it affects the structural response.

The Geotechnical Division recommended a soil shear modulus equal to 18 ksi for static and seismic loads. The analyses were also performed for soil shear modulii of 12 ksi and 24 ksi. For the high frequency safety relief valve and LOCA hydrodynamic loads a soil shear modulus equal to 27 ksi was used. The team found this acceptable.

A review of the results of the soil pressures obtained using the MAT 6 computer program showed approximate uniformity except at the perimeter where the soil pressure was about four times greater than the average. We noted that these results do not reflect practical conditions, although the computer program is based upon mathematical conditions in which pressures at the edge of the circular mat become infinite. The team decided that for the Reactor Building mat, the calculated soil pressures probably produce conservative bending moments and shears.

A check was made to determine if the soil stiffness representations were consistent between the MAT 6 and SHELL 1 analyses. It was found that there was good agreement between the representation.

Another interface involved the affect of superstructures represented in all three analyses. It was obvious that great care was taken to assure consistency of results in the analyses of the mat as well as superstructures. The moments and shears introduced into the mat from the superstructures were conservatively calculated by assuming full vertical cracks and partial horizontal cracking.

In the course of using the GHOSH program, an error concerning the stiffness of triangular elements was discovered by Stone & Webster. This was reported to Stone & Webster - Boston where the programs are controlled and a 10 CFR 50.55e filing was made to NRC prior to this inspection. The team reviewed this error and considered that appropriate corrective action had been taken.

The results of loadings obtained from the three computer programs were entered into the required load combinations. A program NEWSECT was used to calculate the bending stresses in both the concrete and the reinforcing bars. While the program considers that the concrete will crack under tension, as a conservatism no credit was taken for the reduction in moments resulting from thermal differentials and gradients.

The loadings used in the mat represented hydrodynamic loads before the reductions in magnitude based upon tests performed at the Caorso nuclear plant and the concrete fix.

SUMMARY/CONCLUSION

In spite of the complicated procedures, the analysis and design were performed in a consistent manner. It is apparent that there are many conservatisms. A final analysis in all probability will merely document the amount of conservatisms. It is expected that the embedment requirements discussed in Deficiency D4.7-1 will be satisfactorily resolved.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
D4.7-1 (Deficiency) Inadequate Anchorage of Vertical Reinforcing Bars Into Mat

4.8 Safety Relief Valve and LOCA Hydrodynamic Loading

OBJECTIVE

The loading resulting from the actuation of safety relief valves during normal operation as well as a LOCA in combination with other hydrodynamic loads were reviewed. The objective of the evaluation was to determine whether the recommendations and guidelines developed by General Electric and those contained in NUREG-0978 were properly implemented.

EVALUATION

Various documents were reviewed, such as the Gulf States Utilities Report of Safety Relief Valve Loads For River Bend Units 1&2, the General Electric CLR 22A4365, NUREG 0802, NUREG 0978, and Stone & Webster Calculation No. S54.3500, Floor Grating Design.

The scope of the evaluation included examination of the application of criteria in developing loadings, structural modifications, and analyses.

The main criteria documents were General Electric CLR 22A4365 which includes general reductions in loads based upon Caorso testing, GESSAR, NUREG 0802 which accepts GESSAR for safety relief valve loadings, and NUREG 0978.

Prior to the "concrete fix" it was found that the potential amplified accelerations in equipment and piping attached to the steel containment vessel were extremely high. Nevertheless, the structures were designed in most cases to meet these loadings. The "concrete fix" consisted of placement of concrete in the annulus between the steel containment vessel and the outer shield structure. The concrete was placed from the top of mat to 25' above the mat. The main objective of the "concrete fix" was the reduction of vibrations of the containment vessel resulting from the hydrodynamic loadings. The containment vessel is a low mass, high frequency structure which proved to be very sensitive to this loading. The placement of concrete as described reduced the accelerations.

The draft NUREG 0978 changed the duration of the initial spike of the froth impact from 100 milliseconds to 50 milliseconds. Stone & Webster plans to take exception to this and is preparing submittals to the NRC.

The GHOSH computer program was used to generate amplified response spectra and to perform the structural analysis of the structures.

SUMMARY/CONCLUSION

In as much as the "concrete fix" is a recent change and the effect upon the structures will be to reduce loads, reanalysis of the structures has not been undertaken. This is considered reasonable because the structures have been designed and built for the most part to much greater hydrodynamic loading. On the other hand, the amplified response spectra were needed for equipment and piping qualification. This work was expedited and completed prior to the IDI inspection.

The applicable loading criteria were properly input in the design of such structures as the foundation mat, the shield structure and the grating.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.9 Jet Impingement

OBJECTIVE

The objective of the evaluation was to determine whether the loads from jet impingement as calculated by Engineering Mechanics Division were transmitted to the Structural Division and that the loads were correctly used in the design of structural members.

EVALUATION

Four classes of structural components were evaluated; the reinforced concrete drywell wall, structural steel beams, floor grating, and electrical cable tray supports.

The Structural Design Criteria specifies that a dynamic load factor of 2.0 shall be applied to the pressures supplied by Engineering Mechanics Division. This is considered appropriately conservative in that a dynamic load factor of 2.0 assumes that the peak pressure of jet impingement is instantanously applied. The jet impingement load summary contained in Calculation No. 201.120-105 was used in the structural analysis of the drywell. These values correctly reflected the data transmitted by Engineering Mechanics Division in Design Engineering Mechanics 1033, which later was superceded by Design Engineering Mechanics 1370. These documents indicated the magnitude of pressures on identified targets and gave the locations by specifying elevations and azimuths and by delineating the area subjected to the pressures from jet impingement. The pressures were correctly introduced into the design in all of the designs of structural components reviewed.

The structural steel beams were designed for jet impingement in Stone & Webster Calculation No. 554.700, while the floor grating was checked for jet impingement loads (see Detail A4.9-1).

SUMMARY/CONCLUSION

In summary the jet impingement loads were considered in accordance with the design objectives.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
A4.9-1 (Detail) Floor Grating

4.10 I&E Bulletin 80-11, Block Walls

OBJECTIVE

The objective was to determine whether masonry block walls were used as anchorage for various supports for piping, cable trays, conduits, equipment, etc.

EVALUATION

The evaluation consisted of review of applicable documents and a visual check of support anchorage.

The use of masonry block walls on River Bend has been minimized. They were used only where walls had to be removable.

The Pipe Support Specification for River Bend requires that pipe supports shall not be attached to masonary walls in Quality Assurance Category I and II buildings. The Structural Design Criteria contains the statement in paragraph 6.3.1, Removable Concrete Masonry Block Walls, that "Nuclear safety-related equipment or piping shall not be supported by the masonry block walls."

The Structural Division has responsibility for the design of supports for electrical conduit and cable trays and therefore has direct control of the location of these supports. They must also approve all support anchorages and, therefore, exercise another means of control.

During the field walk-downs no instance of anchorage of supports to block walls was observed.

SUMMARY/CONCLUSION

The team concluded that masonry block walls were not used as supports for equipment or piping.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.11 Analysis and Design of Auxiliary Building

OBJECTIVE

This review's purpose was to confirm whether the design objectives as committed to in the FSAR were met and that the design was performed in accordance with applicable codes and regulatory guides.

EVALUATION

The Auxiliary Building is a Seismic Category I reinforced concrete structure with some steel framing. It is supported by a reinforced concrete foundation mat. Several calculations were reviewed for a complete evaluation.

The foundation mat analysis was performed by the use of the ICES STRUDL finite element program. Manual computations were used to prepare the input to this program. The analysis and design of the mat is described in Detail A4.11-1. The review of the design of the mat showed that the pump shaft casing moments, although calculated, were inadvertently not considered (see Deficiency D4.11-1). Also, the design of the mat for shear did not adhere to the requirements of the ACI Code 318-71 (see Deficiency D4.11-2). A check of the calculations and the drawings for the mat showed that reinforcing was provided as required by the design calculations. The design of the pump shaft housings which extend from the bottom to the top of the mat were reviewed and found satisfactory.

The team reviewed calculations for certain floor slabs. The analysis and design of these slabs was performed manually. The review showed that loads as described in the FSAR were analyzed properly and the designs were in accordance with code requirements. The structural drawings were checked. The team found that the specified reinforcement for flexural and shear in the calculations were provided. One particular area of interest was the design of the slab in the main steam tunnel. The initial design required a concrete strength of 3000 psi at 28 days. Due to field requirements this strength was changed in certain areas to either 4000 psi or 5000 psi at 60 days, depending on the needs. A check of the structural drawings and the test results confirmed these values were met.

Steel framing calculations at certain elevations were also reviewed. The steel beams and girders were designed to take the dead load of concrete and other piping and cable tray loads. A minimum concentrated load of 17 kips was placed at the midspan for the beam design. This load was increased in areas where heavy piping existed. All calculations for analysis and design of this structural steel were performed manually. Since most of the piping and cable tray loads were estimated all these beams have to be verified. Stone & Webster has a program for such load verification. Structural drawings were checked and found to reflect the design requirements.

The design of the structural supports for the residual heat removal heat exchanger supports was performed with vertical loads and stiffness requirements provided by the Engineering Mechanics Division. A review of the calculation showed that the design objectives were met and the structural drawings were in accordance with the design requirements.

SUMMARY/CONCLUSION

The design and analysis of the foundation mat was performed by various personnel. This led to the omission of some loadings on the mat. It seemed to the team that more care was given to flexure than to shear in the design of the mat. Although the shear forces were evaluated, the design did not adhere to the code requirements. A reanalysis of the mat for shear should be performed to check its structural adequacy.

The design of the concrete floor slabs and the steel framing were adequate and within the FSAR requirements. The construction was consistent with the design objectives.

The items reviewed in the analysis and design of the Auxiliary Building generally met the FSAR commitments.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
A4.11-1 (Detail) Analysis and Design of Mat
D4.11-1 (Deficiency) Pump Shaft Casing Moments
D4.11-2 (Deficiency) Mat Shear Design

4.12 Structural Supports for Hydraulic Control Units

OBJECTIVE

The objective of this review was to evaluate the interfacing in design, fabrication and installation of structural supports for the hydraulic control units.

EVALUATION

The hydraulic control units are located between the drywell wall and the steel containment wall at elevation 114' 0". The floor consists mostly of steel grating, with exception of the portion occupied by the hydraulic control units. This portion is reinforced concrete slab. About 90 percent of the total annular area between the drywell and steel containment consists of grating. The entire floor is supported by a complex system of steel beams.

This floor had been selected for inspection because of the complex interfacing associated with the hydraulic control units which have been manufactured by General Electric. The control rod drive piping between the reactor and the hydraulic control units were engineered, manufactured and installed by Reactor Controls Inc. The structural supports were designed and constructed by Stone & Webster.

The control rod drive system design information and interface requirements were transmitted by General Electric to Stone & Webster to provide input for design of the structural supports. Stone & Webster issued the specification for design and analysis of the piping and its supports to meet the requirements of General Electric and transmitted this information to their subcontractor, Reactor Controls Inc. The information provided to Reactor Controls Inc. also contained the seismic loads (response spectra), pool swell loads (such as froth) and safety relief valve loads. Reactor Controls Inc. developed the procedure for water hammer analysis, pipe whip and pipe rupture analysis and the analysis of control rod drive hydraulic system piping and supports. This procedure has been reviewed and approved by Stone & Webster. The loads on supports resulting from the control rod drive analysis performed by Reactor Controls Inc. have been transmitted to Stone & Webster for the design of structural supports. Finally, Stone and Webster forwarded the amplified response spectra of the supports to General Electric for hydraulic control unit qualification.

The analysis and design verification of the structural steel framework was done using the computer program STRUDL. The computer program has the provisions of the AISC 1969 edition built in and performs the necessary calculations. As a result of the analysis the computer prints out whether or not the structural section which is being designed passed the code check.

While reviewing the associated drawings we noticed that Nelson studs have been welded to the webs of the structural beams. The Stone & Webster staff could not produce any calculations to demonstrate that the studs were necessary and if so, that they were of sufficient strength. Therefore, the adequacy of the Nelson studs could not be assessed (see Deficiency D4.12-1).

The design procedure for load verification indicated that the load combinations stated in the FSAR have been reduced to one which was considered to be the critical combination. Since some load combinations have different section capacity factors while the allowable stresses vary depending on the loading condition, an elimination of various load combinations without quantitative comparison might lead to an error (see Deficiency D4.12-2).

While reviewing structural drawings pertinent to the subject floor we noticed that in some cases slabs were supported on an angle whose one leg was welded to the web of the beam. No calculations were found which demonstrated that the angle is sufficient to support the slab and that the shear in the slab at the point of support is not excessive (see Deficiency D4.12-3).

SUMMARY/CONCLUSION

In summary, the team concluded that the design of structural supports for the hydraulic control units are controlled by adequate procedures. For the sample reviewed the procedures were generally followed. The team concluded that the deficiencies found in this portion of the inspection will not have a significant adverse effect on the structures.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
D4.12-1 (Deficiency) Lack of Calculations for Nelson Studs
D4.12-2 (Deficiency) Qualitative Elimination of Load Combinations
D4.12-3 (Deficiency) Lack of Calculations for Support Angles and Shear in the Concrete Slab

4.13 Verification of Structural Design of Pipe Supports

OBJECTIVE

The objective of this portion of the inspection was to examine the coordination between the design of the mechanical components and the structural design of pipe supports and to verify that the selected sample represents an adequate design.

EVALUATION

Nine pipe support drawings were selected for the inspection. Four of the pipe supports were located in the Reactor Building and five in the Auxiliary Building. The inspector verified (with one exception which was not accessible) that the locations of the supports were according to the plans on the basis of the information provided on the drawings. We also verified portions of the design of the structural members affected by the supports. Six calculation packages for which the structural design has been verified were provided by the Stone & Webster office. The review of these packages indicated that the supporting members have been analyzed using the GT STRUDL computer code and the design check was done using the AISC 1969 standard. All of the packages examined passed the code check.

SUMMARY/CONCLUSION

Based on the samples selected for review, the team concluded that the proper interfaces between disciplines had been utilized and that control of design had been exercised. No discrepancies or deficiencies in design or location of the pipe supports were found.

The team found samples of designs in this area to be adequate.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.14 Analysis & Design of Mechanical Component Supports

OBJECTIVE

The objective of this section was to determine whether the supports for mechanical components were properly designed.

EVALUATION
The residual heat removal pump and the residual heat removal heat exchangers were selected for evaluation. The anchor bolts for the residual heat removal pump were designed by Stone & Webster. The analysis and design of the residual heat removal heat exchanger supports were also performed by Stone & Webster.

The residual heat removal pumps and motors were supplied by General Electric. Calculations were performed by Stone & Webster to design the anchor bolts for the system. The forces that act on the bolts were determined. These included the nozzle loads provided by General Electric. The concrete pad around the anchor bolts was designed to carry the shear that was transferred to the top of the Auxiliary Building mat. Some reinforcing was provided in the concrete pads. The plate washers used with the bolts were designed to accommodate the loads imposed on them. The pad details were later revised during the construction stage to include a 2" sole plate for leveling the equipment. A check of the structural drawings showed that the design was reflected as required by the calculations.

The residual heat removal heat exchangers were supplied by General Electric. Stone & Webster has performed calculations to analyze and design the support system for these heat exchangers. Mathematical models were prepared for the STRUDL computer program to perform a stress analysis. The support systems of the residual heat removal heat exchangers were analyzed and designed by the Engineering Mechanics Division of Stone & Webster. They also provided the Structural Division with loads for design of the structural supports for the heat exchanger support system.

Two mathematical models were prepared to reflect the actual design conditions. The upper and lower supports, as well as the structural supports, were included in the models. The actual construction, as observed during the site visit, was reflected in the mathematical models.

The basic design criteria for the upper and lower supports of the heat exchangers were the stiffness requirements imposed by General Electric. All horizontal loads were taken by the upper and lower supports. The vertical loads were transferred to the structural supports by the hanger members. For two of the heat exchangers a structural column replaced one of the hanger members. Computer analysis was performed for various load combinations. Deflections and stresses were checked and found to be within the design limits.

SUMMARY/CONCLUSION

The two mechanical component supports reviewed were analyzed and designed in accordance with the commitments of the FSAR. The structural drawings, as well as the as-built conditions, reflect the actual design.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.15 Support of Electrical Components

OBJECTIVE

The objectives of the evaluation were to evaluate the appropriateness of the seismic testing of the batteries and the structural capability of the battery rack and to review the methods and means of supporting electrical cables.

EVALUATION

The evaluation consisted of a review of the design and analysis of the test setup, the design of the racks, and inspection of the site installation.

The test set-up consisted of 3 cells which weigh 996 lbs. The racks consisted of stringers which directly support and confine the batteries and braced framing which ties into the floor. The stringers consist of unistruts, while the braced framing consists of structural steel angles, tubes, and flat plates. Although the test setup consisted of 3 cells and the actual rack can accommodate 60 cells, proportionality between stiffness and mass was maintained. The structural and dynamic analysis of the test rack and the actual rack were performed by independent organizations: Ralph C. Dumack, P.E. and Associates and Allstates Design and Development Co. Inc., respectively. The structural members of the racks were identical and both analyses indicate that the fundamental natural frequencies for vertical and orthogonal horizontal directions were above 33 Hz. This is well above the frequency for which the installation may be considered rigid. Another difference between the test and actual racks was that the base of the test rack was anchored by means of anchor bolts while the actual rack base is welded to embedded plate strips. This was not considered to be significant.

The team observed that several of the base welds were missing. A check of design changes revealed that grout holes in the embedded plates prevented the welding. The deletion of these welds was reviewed and approved according to procedures.

A walk down of key cable trays and a review of calculations concerning the generic design of cable tray supports was conducted. Special designs required due to field conditions were also examined.

SUMMARY/CONCLUSION

The team concluded that the seismic testing of the batteries was appropriate. With the exception of an isolated procedural error which might affect the calculated loads on one of the cable tray supports, the procedure for supporting electrical cable appeared to function well. The team concluded that the electrical cable trays are sufficiently anchored.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
D4.15-1 (Deficiency) Checking Procedure Violations, Cable Tray Supports

4.16 Inspection of Reactor Controls, Inc.

OBJECTIVE

The objectives of this review were to examine the Reactor Controls, Inc. quality assurance procedures, the technical personnel qualifications, to review the methods of analysis of the control rod drive piping and the associated supports and to evaluate Stone and Webster performance in defining and monitoring the engineering services and construction provided by Reactor Controls, Inc.

DESIGN INFORMATION

Reactor Controls, Inc. is divided into two major groups consistent with their functional responsibilities. The project group is responsible for the engineering and design while the technical support group consists of staff specialists who provide technical assistance in specific areas. Furthermore, the company is divided into three organizational categories: the president of the company, who is concerned with corporate business, the headquarters, which is concerned with analysis and design; and the site group, which is responsible for construction.

The major portion of the effort under the contract with Reactor Controls, Inc. is the design, fabrication, and installation of about 35,000 feet of small diameter piping between the control rod drives and the hydraulic control units. Engineering services and construction were procured to satisfy a Stone & Webster specification. Significant changes to the scope of the contract were made in addenda to the specification. Major changes were also made to provide information to Reactor Controls on loading requirements. The basic information regarding parameters of design was transmitted from Stone & Webster by means of Addendum 1 to the specification. The development of the specifications pertaining to the Reactor Controls, Inc. over a period of years is described in Detail A4.16-1.

The major milestone in the analysis procedure is the interaction between the piping group and the support group in an effort to develop the stiffnesses of the piping system and piping supports. After the piping adequacy is verified the pipe reactions are transmitted to the support group for verification of the adequacy of the supports. This information, together with the loads resulting from the pipe rupture analysis, is transmitted to Stone & Webster for analysis of the embedment plates and structural supports.

PERSONNEL AND GUIDANCE

In order to assess the professional qualifications of the Reactor Controls Inc., six members of the Reactor Controls, Inc. staff were interviewed. Their experience in the field of nuclear engineering varied from one year to fifteen years. Their experience on the River Bend project is rather limited, the maximum length of time spent by one person being two years. None of the engineers interviewed had a professional engineer's license. The two lead analysts interviewed and one analyst have master degrees in engineering. The academic level of the remaining four members of the staff was bachelors' degree in civil or mechanical engineering.

At the site we interviewed the key personnel responsible for quality control. The site group consists of 284 employees, 240 being craft employees. There are 95 welders and 28 foremen. The remaining 117 employees are apprentices and laborers.

Reactor Controls, Inc. holds the following ASME certificates:

(a) N-1299, NPT, Class 1, 2, and 3, vessel parts and appurtenances, piping subassemblies, component supports and penetrations, and

(b) N-1300, NA, Class 1, 2, 3 and CS, installation of components, parts, appurtenances, piping subassemblies and components supports.

The N certificate is held by Stone & Webster. The Quality Control inspectors perform an inspection upon receiving material at the site and final inspection after the construction has been completed. The type of inspection varies from visual to radiographic according to the requirements of the Process Requirements Sheet. This is the document listing the appropriate requirements, such as those for welded joints.

The inspection included review of the Quality Assurance/Quality Control program and procedures. The main documents defining the quality assurance program of Reactor Controls, Inc. are the "Quality Assurance Manual" (the Manual) Second Edition (which describes the policy of the company and has been approved by Stone & Webster, the ASME Code Committee, and the Hartford Steam and Boiler Company) and its companion "Quality Assurance Instruction Book", which implements the provisions of the Manual. The Manual contains a matrix which shows coverage of criteria for 10 CFR 50, Appendix B and Paragraphs NCA 4134.0 through 4134.18 of the ASME Code "Boiler and Pressure Vessel Section III, Division 1.

The inspection included review of quality assurance surveys of vendors, vendors' audits and internal audits. Quality assurance surveys are conducted at approximately three year intervals and internal and external audits on an annual basis. On the basis of the surveys an Approved Vendors List is formed from which prospective subcontractors are selected when needed.

For the purpose of this inspection a sample of three quality assurance surveys and three internal and external reports were selected. No deficiencies were found in this area.

Documentation required from Reactor Controls, Inc. as part of the contract includes an outline of the stress analysis for the piping and supports. Revision 4 of that document is the present approved evaluation requirement. No significant problems were found with Revision 4 of the proposal for stress analysis. It should be noted, however, that Reactor Controls, Inc. followed the Stone & Webster specification on required translational stiffness of pipe supports. This required stiffness is based on the size of the pipe to be supported with the intent to maintain natural frequencies above the peak frequencies of response spectra loadings. No allowance is made for how many pipes of that size might be supported by one support. In the control rod drive piping there can be 72 pipes on one support. Unresolved Item U4.16-1 discusses this aspect of design.

EVALUATION

Analysis of the insert and withdrawal piping was performed by dividing the piping systems into two independent piping structures:

(a) a piping system inside the drywell, i.e., piping running from the control rod drive housing to the drywell wall penetration, and

(b) a piping system outside the drywell, i.e., piping running from the drywell wall penetration to the hydraulic control units.

This division of piping into two independent piping models was made on the basis of the assumption of a rigid anchor at the drywell wall as specified in the design. Each independent piping structure was modeled as a multi-degree-of-freedom, finite element system, consisting of straight and curved beam elements using a lumped mass formulation. The TPIPE Computer Code, Version 5.0 was used for the analysis. The effects of the mass of friction clamp and connecting hardware attached to piping were included in the piping models.

Both static and dynamic analyses were performed. Static analyses were performed for gravity, thermal, seismic and thermal anchor movement, and zero period acceleration for both upset and emergency loads. Jet impingement loads were applied to the piping by a static equivalent method. In performing this static equivalent analysis for jet impingement a dynamic amplification factor of 2.0 was applied.

Dynamic analysis techniques were used to determine the system's response to seismic, structural response loads and water hammer loads. These techniques were based on the outline developed by Reactor Controls, Inc. and were reviewed and approved by Stone & Webster. These techniques use either response spectrum or time history analysis methods depending upon input loading characteristics. In performing response spectrum analysis, a cutoff frequency of 150 Hertz was used in modal extraction. Individual maximum responses for each direction were combined in accordance with Regulatory Guide 1.92. The resulting peak responses for each of three directions were combined by square-root-of-the-sum-of-squares method. These results were then screened against the results from zero period acceleration analysis of individual load cases. The zero period acceleration value was picked up from the input spectrum at the cutoff frequency of 150 Hz.

Review of the Pipe Rupture and Pipe Whip Evaluation and the Verification Descriptions of Computer Programs used for the River Bend Project disclosed that the program PIPERUP, which has been included in the evaluation, was not listed in the verification. Our inquiries regarding the apparent omission of the program from the verification revealed that it had been decided not to use the PIPERUP computer code. Inclusion of the PIPERUP in the evaluation might lead to its inadvertent use. To avoid this it should be removed. During the course of the inspection we were informed that the PIPERUP program was removed from the pipe rupture and pipe whip evaluation (see Deficiency D4.16-1).

While reviewing the mathematical model used for dynamic analysis of the control rod drive piping we noted a dimensional discrepancy. This discrepancy was between the model and information provided in the "Hydraulic Control Unit Riser Isometrics" which contained the basic information used for the analysis. Furthermore, comparison of the dates of the two documents disclosed that the mathematical model was prepared before the design analysis outline was finally approved, which indicates that the analysis was made on the basis of preliminary information. The design document did not contain evidence that it was updated (see Deficiency D4.16-3).

We reviewed the analysis of the pipe supports. The seismic category I supports were analyzed by statically applying pipe reaction loads and support dead weight using the various loading combinations. Dynamic loads were evaluated using the response spectrum method. Friction forces were included at applicable supports where axial movement of the pipe is allowed. Friction factors of 0.54 for thermal and dead loads and 0.0 for seismic loads were used for sliding steel surfaces. Loads due to jet impingement were included in the piping analysis and were applied uniformly to the portions of the supports identified as targets. Pipe support loads were determined from piping analysis by adding the normal operating load and the absolute value of inertial loads. The analysis was performed using the EASE2 computer program using unit loads in three directions, by finite element method and postprocessing computer programs E2A17 and EWELD. Piping and support loads were combined using normal, upset, emergency and faulted combinations. The allowable stresses used in the analysis were in accordance with the ASME Boiler and Pressure Vessel Code Subsection NF Appendix XVII, 1977 Edition with Addenda through Summer 1979.

Review of the calculations for pipe supports disclosed that the analysis was carried using a mathematical model in which the piping was decoupled from their supports. We discussed this matter with Reactor Controls, Inc. and stated that such a model would be against generally accepted engineering practice and the commitments stated in the FSAR Section 3.7.2.1.1. The Reactor Controls Inc. staff provided us with their justification for decoupling between piping and supports in which they stated a number of items have been described which compensate for the omission of piping in analysis of the supports. Due to the complexity of the problem, the inaccuracy of the analysis caused by the decoupling of piping from their supports raises the question whether the analysis is acceptably conservative. In view of the above the team concluded that confirmatory analysis should be performed considering a coupled model of the piping and supports in order to verify that the original analysis is acceptable (see Unresolved Item U4.16-1).

Our inspection of the document control program disclosed that the calculations lack proper controls and documentation. These calculations, containing computer input data, are essential to understanding the analysis and traceability of the computer inputs. The controls for this vital information violate the requirements of the Quality Assurance Instructions (see Deficiency D4.16-2).

SUMMARY/CONCLUSION

In summary, the quality assurance program is considered adequate. We found evidence that it is properly implemented, although we found that internal document control should be improved.

The methods of analysis were generally good. We found that additional analysis should be performed in order to obtain greater confidence regarding the conservatism of the mathematical model for piping. The quality control operation at the site appears to be in accordance with acceptable standards. No deficiencies were found in this area.

The performance by Stone & Webster in defining the effort to be performed with their specification is generally acceptable. Delays in defining loading conditions are consistent with the generic problems associated with hydrodynamic loads in boiling water reactor suppression pools.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4
A4.16-1 (Detail) Stone & Webster Specification for Reactor Controls Inc. Contract

D4.16-1 (Deficiency) Unverified Program PIPERUP
D4.16-2 (Deficiency) Document Control of Reactor Controls Inc.
D4.16-3 (Deficiency) Inconsistency Between Mathematical Model & Criteria
U4.16-1 (Unresolved Item) Decoupling of Control Rod Drive Piping and Supports

4.17 As-built Conditions

OBJECTIVE

The objective of this review was to determine whether the as-built conditions matched the design requirements and whether field changes were incorporated in the design documents. Also, some systems were checked to determine whether the mathematical models used in analysis conformed to the as-built conditions.

EVALUATION

Design changes can be initiated by Stone & Webster personnel at the site or at the project offices in Cherry Hill. When a design change is initiated an Engineering and Design Coordination Report is issued. These reports must be incorporated into the structural drawings to reflect the as-built conditions.

To determine whether the Engineering and Design Coordination Reports were incorporated properly in the structural drawings, samples were selected for verification during the site visit. Stone & Webster computer log showed each drawing with a particular revision and the Engineering and Design Coordination Reports issued for the revision. Two drawings as referenced in this section were reviewed. This review showed that these drawings were revised and all information was properly incorporated.

One of the major Engineering and Design Coordination Reports in Drawing 12210-EC-67A-4 was the change of concrete strength in the main steam tunnel. The required compressive strength of concrete was changed to higher values as requested by the field. A check of the test reports showed that such strengths were obtained and that the as-poured concrete matched the design requirements.

To confirm the application of design parameters at the construction site, certain anchor bolts for various equipment were checked. The team selected the following equipment for review: Automatic depressurization system accumulators, Residual Heat Removal Pump No. 1E12*PC002B, and the chilled water tanks, pumps and motors. The chilled water tanks, pumps and motors are in the control building. The other equipment is in the auxiliary building. The appropriate vendor drawings were checked against the Stone & Webster drawings for anchor bolt schedules. Both drawings were consistent. The number of bolts and sizes supplied in the actual construction matched the drawings. A field survey by the team showed that one of the chilled water tanks (1HVK*TK1A) had two anchor bolts with insufficient projection. A Nonconformance & Disposition Report had been prepared by Stone & Webster personnel at the site, showing that the problem was detected and satisfactorily resolved.

The support system of the residual heat removal heat exchangers was also inspected to determine whether the site conditions matched the analytical models used. This check revealed that the mathematical models reflected the actual structure.

SUMMARY/CONCLUSION

Site investigation and review of documents indicated that changes at the site were well controlled. Such changes are incorporated into the structural drawings to reflect the updated as-built conditions. Inspection of certain equipment anchor bolts and supports indicated that design requirements were met and the actual constructed conditions reflect the analytical models used.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

4.18 Geotechnical Engineering

OBJECTIVE

The objective of the evaluation was to determine if the design concept of excavating in situ soil and replacing with compacted structural fill to support the Category I structures had been implemented satisfactorily.

EVALUATION

Selected documents pertaining to the engineering, design and construction of the plant were reviewed (see Detail A4.1-1). Data included relevant sections of the FSAR, specific hydrologic and foundation design parameters, geotechnical design criteria and construction specifications covering the placement and compaction of structural backfill. The following areas were addressed:

a) Hydrologic studies
b) Foundation investigations
c) Engineering properties of foundation soils
d) Foundation design concept
e) Excavation and dewatering
f) Structural backfill
g) Liquefaction potential
h) Response of soil to dynamic loading
i) Earth pressure considerations
j) Subsurface movements

SUMMARY/CONCLUSION

The team agreed with the methodology used to derive the significant parameters influencing the foundation engineering design of the River Bend Station. It is apparent that the implementation of the foundation design concept was successful. Movements of the base of the excavation due to heave of the underlying soils and subsequent settlement due to placement and compaction of structural fill were anticipated and generally confirmed by instrumentation readings. Application of load from the structures resulted in an additional incremental settlement of approximately 2" and it is concluded from the data presented that settlement had essentially levelled off resulting in a maximum residual heave of approximately 2". Long-term settlement over the extended 40 year life of the plant is expected to be negligible.

RELATED FORMS:
A4.1-1 (Detail) References for Section 4

5.0 ELECTRIC POWER SYSTEMS

The objective of the inspection in the electric power discipline was to perform a technical review of selected electrical components of the Residual Heat Removal System and other relevant portions of the power distribution system to verify:

That the regulatory requirements, licensing commitments, and design bases specified in the Final Safety Analysis Report are correctly translated into the plant design and incorporated in the design documents, such as specifications, and drawings.

That correct design information has been provided to responsible design organizations.

That design documents are correlated and are consistent with one another.

That design changes, including field changes, are conducted in a controlled manner.

The adequacy of the design to accomplish design basis safety functions under normal and accident conditions.

The conformance of the as-built system or component to the design criteria and design documents.

The team also reviewed organizations, responsibilities, and work functions and evaluated the interfaces and methods of communications between these organizations.

5.1 Design Information

This section discusses electrical design responsibilities and functions.

Prior to 1982, Gulf States Utilities Company Project Engineering Department was located at Beaumont, Texas. Project Engineering electrical engineers monitored the Stone & Webster and General Electric electrical design of the River Bend Station and coordinated activities with Gulf States site engineers and the Stone & Webster Cherry Hill Office. Project Engineering electrical engineers participated in conceptual design, and provided review and approval of the following items: electrical specifications, bid recommendations, changes to procurement and erection specifications, design changes which impact the Final Safety Analysis Report, and nonconformances. The electrical engineers also reviewed potentially reportable deficiencies, requested certain design changes, and performed detailed analyses of selected design changes. In 1982, Gulf States Utilities formed two separate engineering groups for the River Bend Project. The Nuclear Plant Engineering Department, located in Beaumont, Texas, was formed with staff members of major disciplines from Project Engineering. Project Engineering is now located at the site.

The Nuclear Plant Engineering Department provides technical support for Start-up and Project Engineering for the River Bend Station. Nuclear Plant Engineering Department electrical engineers coordinate activities with Project Engineering on site, manage the equipment qualification efforts of Stone & Webster and General Electric, participate in Boiling Water Reactor Owners Group activities, monitor control room design and modification efforts, provide technical support for licensing, and provide technical support in the electrical area. Gulf States Nuclear Plant Engineering Department electrical engineers, for example, recently reviewed elementary diagrams for 21 General Electric systems and developed change packages to incorporate corrections and River

Bend-specific information. Other activities include support in developing technical specifications and review of start-up procedures.

The Gulf States Utilities Project Engineering Group has electrical engineers on the staff who monitor the Stone & Webster and General Electric electrical design, design modifications, and electrical installation. The electrical engineers review and approve procurement specifications and specification revisions, prepare change documents, monitor the electrical installation specification, and review and approve Stone & Webster and General Electric commercial proposals and adjustments. Electrical engineers from the Gulf States start-up group coordinate control room design modifications and develop change packages.

The Electrical and Controls Group at the Stone & Webster Cherry Hill Operation Center has the prime design responsibility for the electrical and instrumentation and controls portion of the plant. Prior to January 1984, the electrical group and the controls group were independent and their responsibilities fell clearly within discipline lines. The electrical group was responsible for the electrical power area, electrical equipment specifications, calculations, one-line diagrams, wiring and physical design, General Electric interface, the main control room, and the electrical sections for the Final Safety Analysis Report. In January 1984 a combined Electrical/Controls Group was formed to facilitate support to the field. The current Electrical/Controls organization is separated into a field support group and a technical support group with electrical and controls engineers in each group. The field support group is responsible for: elementary and analog wiring diagrams, vendor elementaries, loop diagrams, the main control room, system diagram reviews, vendor drawing review, interconnection wiring and cabling design, panel arrangements, bills of material, raceway and physical design, and field modifications. The field support group is also responsible for two design squads. The physical design squad produces cable tray and conduit installation drawings, seismic support design, and equipment installation details. The wiring design squad produces wiring diagrams, elementaries, and cable tabulation information. The technical support group has responsibility for: logic diagrams, one-line diagrams, calculations, Final Safety Analysis Report sections, studies and analyses, equipment qualification interface, specifications, instrument design and setpoints, and the plant computer. The Stone & Webster Boston office performs design reviews in specific electrical areas as required.

The Stone & Webster Site Engineering Group Electrical Discipline administers and maintains the electrical installation specification and assures that the electrical installer provides electrical installation in accordance with the design drawings. Other activities include procurement of field purchased non-engineered equipment, resolution of field problems such as cable installation and termination problems, development of work change packages, and review of drawings for constructability and interfaces. Production engineering, design, and drawings are not accomplished by the Site Engineering Group Electrical Discipline; this group essentially resolves problems impacting construction.

The team reviewed the Stone & Webster Electrical/Control group's technical correspondence files concerning the main control room to determine the level of detail of communication between Stone & Webster, General Electric, and Gulf States Utilities. The Stone & Webster responsible engineer for the main control room readily obtained File 242.414 from Project Document Control covering the period from 1974 to present. This included all related correspondence such as telecons, interoffice correspondence, meeting notes, and letters which reflected the level of detail concerning communication between Stone & Webster, Gulf States Utilities, and General Electric Company. The team reviewed correspondence covering: engineering change memoranda concerning line codes for field cabling; controlboard device cross-reference lists; control room design freeze; Agastat type relays; annunciator alarm numbering; arrangement of controls and instruments on balance of plant panels; control room cabling interface; panel wiring; test procedures; and environmental qualification. We found that technical communications were detailed and that they reflected effective transfer of information by Stone & Webster to Gulf States and General Electric on all aspects of the control room design. The correspondence also reflected guidance, technical input, and review of the control room design by Gulf States.

The team reviewed the methods and design inputs used by Stone & Webster engineers for development of the electrical design. The control system engineer develops the elementaries and analog diagrams using the following design inputs: process system diagrams, the Final Safety Analysis Report, one-line diagrams, related correspondence, applicable change packages, and control loop and logic diagrams. The wiring squad uses the elementaries and analog diagrams as design input for development of interconnection diagrams, cable block diagrams, termination drawings, cable tabulation and routing list, motor and load list, and equipment bills of material. The physical squad uses the elementaries, analog diagrams and cable block diagrams as design input for development of the conduit installation and tray arrangement drawings and details.

5.2 Personnel and Guidance

This section summarizes the staffing and guidance information reviewed in the electrical area.

The team reviewed the background and experience of the Gulf States Nuclear Plant Engineering Department electrical group. The supervising electrical engineer has a BSEE Degree, a professional engineering license, and 11 years engineering experience in the nuclear area. He has been with Gulf States for less than one year. The supervising electrical engineer is responsible for the technical work of seven engineers. All of these engineers have advanced degrees and two have professional engineering licenses. They have an average of ten years professional experience with seven years of experience on nuclear projects and two years of experience on the River Bend Project.

During the 1972-1983 period, when the bulk of the electrical design effort was under development, Gulf States Project Engineering Department electrical engineers monitored the design effort. The staffing for the 1972-1979 period was approximately three engineers per year. For the 1980-1983 period, an average of five engineers were assigned to the River Bend Project; four of the five engineers were contract type employees. All of these engineers had advanced degrees and the majority were registered professional engineers. They had an average of eleven years of professional experience.

The team also reviewed the qualifications of the Stone & Webster Cherry Hill Office electrical engineers within the River Bend Electrial/Controls group. The lead electrical engineer has a BSEE degree and has 12 years engineering experience, which includes 4.5 years with Stone & Webster and 2 years on the River Bend Project. The lead electrical engineer supervises the field support group which includes both electrical and controls engineers, and the wiring and physical design squads. Eighteen electrical engineers are assigned to the River Bend Electrical/Controls group in either the field support or technical support areas. Fifteen of the eighteen engineers have advanced degrees and seven are registered professional engineers. They have an average of 15 years professional experience with six years on nuclear projects and 3.5 years on the River Bend project.

The team reviewed the qualifications of the Stone & Webster Site Engineering Group Electrical Discipline. The lead electrical engineer has a BSEE degree and 6 years engineering experience, including 3.5 years with Stone & Webster on the River Bend Project. The lead electrical engineer directly supervises 32 engineering and support personnel, 11 of which are electrical engineers. Eight of the eleven electrical engineers have advanced degrees. None of the electrical engineers in this group are registered professional engineers. The eleven electrical engineers have an average of 14 years professional experience with 5 years on nuclear projects and 1.5 years on the River Bend project.

Guidance to Stone & Webster electrical engineers for the preparation of elementary diagrams and analog wiring diagrams is provided by various project guidelines and standards. Control Group Guidelines define requirements for balance of plant electrical drawings associated with main control room circuits such as power supply, signal resistor, and control circuit drawings. Administrative Guidelines and Engineering Assurance Procedures provide guidance on preparation, review, and approval of elementary diagrams. Guidelines also provide instructions for proper interfacing of control engineering and control design efforts. Technical Guidelines require engineers to comply with electrical separation requirements provided in Electrical Independence Design Criteria 12210-240.200. In addition to project guidelines, numerous elementary diagram development instruction drawings are provided to ensure technical accuracy, completeness and consistency of electrical drawings; these instruction drawings address electrical aspects such as wire number identification, switchgear details, motor operated valve circuits, isolator circuits and analog wiring diagrams.

Formal training on use of Stone & Webster's Engineering Assurance Procedures is provided to all engineers by the Stone & Webster Boston Staff. The presentations on the Engineering Assurance Procedures are specifically tailored to the River Bend procedures. Informal training is essentially on-the-job training conducted by lead engineers and supervisors.

The team had no further questions regarding personnel or guidance in the electric power systems area of the River Bend design.

5.3 Motor Operated Valves

OBJECTIVE

The objective of this portion of the inspection was to review the equipment environmental qualification assessment program with respect to safety-related motor operated valve actuators.

EVALUATION

The team reviewed Stone & Webster's efforts with respect to environmental qualification of safety-related electrical equipment. We focused our review on activities associated with Limitorque motor operated valve actuators. In particular, we reviewed the qualification documentation and Stone & Webster's methodology for assessing environmental qualification of Limitorque actuators for Low Pressure Coolant Injection Valves 1E12*F042A and B (see Detail A5.3-1).

SUMMARY/CONCLUSION

We observed that Stone & Webster was actively engaged in the equipment environmental qualification program for River Bend. We found that Specification 228.212 data sheet incorrectly specified the location of motor operated valves 1E12*F042A and B as "Auxiliary Building, outside Containment." These valves are actually located inside containment, and the existing Class B motor insulation and non-metallic actuator parts are unqualified for this application. If these valves (loop A and B low pressure coolant injection valves) were to fail to open during a postulated accident, the residual heat removal pumps would be unable to deliver the required flow to the reactor vessel. Specification 228.212 has been reviewed and design verified several times since 1974 (9 addendums and 1 revision). Although this error was found during environmental qualification reviews, we concluded that the specification review process and the design review process for Specification 228.212 were deficient because the initial reviews and amendments failed to uncover this critical data sheet error (see Detail A5.3-1).

We determined that numerous other balance of plant and General Electric-supplied motor operated valve actuators have been reviewed and found to be deficient with respect to qualification by the Stone & Webster Equipment Environmental Qualification Groups. We concluded from our evaluation of Stone & Webster's review activities with respect to Limitorque motor operated valve actuators that the initial environmental qualification efforts on the project were weak and inadequate; however the recent review process was well organized and controlled. Resolutions and actions were initiated to resolve the numerous identified problems in this area.

We also found that the Stone & Webster design has unqualified Limitorque motor actuator space heater loads supplied by Class 1E power distribution panels in violation of IEEE-Std 384-1974 and Regulatory Guide 1.75 Rev. 2 provisions for maintaining electrical and physical independence of redundant safety-related equipment. These loads should have either been connected to non Class 1E power or analyzed to demonstrate that their failure would not affect the Class 1E supply (see Deficiency D5.3-1).

RELATED FORMS:
A5.3-1 (Detail) Motor Operated Valves
D5.3-1 (Deficiency) Unqualified Motor Operated Valve Space Heaters

5.4. Cable Design and Analysis

OBJECTIVE

The objective of this portion of the inspection was to evaluate a sample of cable analyses performed by Stone & Webster to determine whether cables were properly sized and installed to limit the temperature rise of conductors to within the rating of the cable for normal and overload conditions.

EVALUATION

The electrical design and associated calculations were evaluated for branch circuit power cabling from motor operated valve 1E12*F042B to the respective motor control center 1EHS*MCC2K. Design cabling, calculation methods, cable criteria, project engineering standards, breaker coordination and manufacturer's data were reviewed (see Detail A5.4-1).

SUMMARY/CONCLUSION

One documentation discrepancy and one error were found. They did not affect the overall validity of the calculations or the design (see Deficiency D5.4-1). However, we noted that these types of errors, which are considered to be obvious, were not detected by two independent design reviews. The team concluded that the calculation and analyses were performed in a competent manner and the results were judged to be correct. The technical data that were reviewed confirmed the major assumptions used in the analyses with the exception of discrepancies which are discussed in Deficiency D5.4-1. We had no further questions in this area.

RELATED FORMS:
A5.4-1 (Detail) Cable Design and Analysis
D5.4-1 (Deficiency) Inadequate Calculation Assumptions

5.5 Cable Routing

OBJECTIVE

The objectives of this portion of the inspection were to evaluate the drawings, documentation, and methods used by the electrical designers for cable routing and cable data tabulation, and to evaluate the installed cabling in the field in comparison to the design documents.

EVALUATION

The team randomly selected power cabling for a motor operated valve located inside containment and reviewed all associated electrical drawings and documents which show the design cable route for conduit and cable trays between the motor operated valve and its respective motor control center. The team compared the design documentation for the selected cable to the actual installed configuration. We reviewed all documents and drawings which established the cabling design. This included:

Cable block, one-line power distribution, wiring interconnection, and elementary diagrams;

Conduit installation, cable tray identification and tray arrangement drawings;

Electrical separation criteria; computer developed cable and raceway reports for routing, tray fill, tie points; and

Cable identification; and cable pull tickets and raceway tickets.

We then conducted a complete field walkdown of the subject cabling and raceways (see Detail A5.5-1).

SUMMARY/CONCLUSION

We found one deficiency in that the cable was not installed in a specific tray as required by the design (See Deficiency D5.5-1). The team believes that this error is an isolated construction deficiency. We determined that a field developed routing change to the design conduit routing tie point at the cable tray was performed in a documented and controlled manner. The team observed that Stone & Webster is not providing as-built (as-installed) Conduit Installation drawings. Since conduit is essentially field routed, there can be large deviations between the installed conduit path and the design path between tie points. The team was therefore concerned that a reasonable representation of the actual path for safety-related conduit is not presented on Stone & Webster design drawings. The team recommends that the conduit installation drawings be revised each time field run conduit installation deviates from the design drawings by a predetermined value (see Unresolved U5.5-1).

We found that the numerous design documents and drawings were consistent with one another and in agreement with the installed configuration where required. In general, we found this area to be in good order. The team was impressed with the accuracy and consistency that Stone & Webster maintained among the numerous related design documents.

RELATED FORMS:
A5.5-1 (Detail) Cable Routing
D5.5-1 (Deficiency) Cable Installation Routing Error
U5.5-1 (Unresolved) As-Built Conduit Installation drawings

5.6 Containment Electrical Penetrations and Fault Currents

OBJECTIVE

The objective of this portion of the inspection was to evaluate a sample of fault current protection and to evaluate the ability of containment electrical penetration assemblies to withstand the maximum postulated fault current which could occur.

EVALUATION

The team reviewed the electrical design and associated calculations for branch circuits which feed through electrical containment penetration assemblies to evaluate the short circuit capability. We selected a class 1E motor operated valve circuit and followed the circuit through the calculation and design documents. Electrical interconnection methods, breaker coordination, fault currents, test reports, manufacturers data, cable data, and calculation methods were reviewed (see Detail A5.6-1).

SUMMARY/CONCLUSION

We reviewed calculation E-107, performed for short circuit determination. We found that the 480V load center transformer rating and impedance assumptions were incorrect, therefore the results of the calculation are currently in error. We learned that Stone & Webster intended to revise calculation E-107 to account for receipt of actual vendor data. However, Stone & Webster subsequently found that several problems existed in the computer program used to develop the calculation results. A review of the vendor test data indicated that the penetration test program short circuit current compensated for errors in the computer program; however, a verification of the calculation is needed.

Stone & Webster started calculation E-162 in an attempt to replace calculation E-107, but later decided to recalculate penetration fault currents by hand calculation. We observed that the Engineering Assurance Problem Tracking System continues to identify that calculation E-107 requires recalculation and verification. The team was impressed by this tracking scheme. We reviewed the present activities of the electrical technical support group with respect to containment electrical penetration fault current calculations. We found that the Stone & Webster sample thermal capacity and protection coordination curves clearly show that the fault current verses time conditions are well within the circuit breaker trip rating and the circuit breaker will trip earlier than the maximum time for which the penetrations are qualified. The curve also shows that the fuse provides back-up protection as required.

We concluded that the design is consistent with Section 8.3.1.1.4.3 of the Final Safety Analysis Report which requires the design to withstand the maximum short circuit current, and requires primary and back-up protection to be provided. We also determined during our inspection that Stone & Webster was recalculating fault current.

RELATED FORMS:
A5.6-1 (Detail) Containment Electrical Penetrations and Fault Currents

5.7 Motor Control Centers

OBJECTIVE

The team reviewed the design to determine the application of thermal overload protection for class 1E motor operated valves and to determine conformance to Final Safety Analysis Report commitments and regulatory criteria.

EVALUATION

We reviewed calculation E-164-3, sizing and selection of thermal overload heaters for class 1E motor operated valves fed from motor control centers. As a basis for the review, we selected Low Pressure Coolant Injection valve 1E12*F042B which is fed from cubicle 3D of motor control center 1EHS*MCC2K. We also reviewed elementary diagrams, Final Safety Analysis Report requirements, vendor drawings, electrical load lists, engineering guidance on thermal overload trip settings, and vendor data. We also reviewed site activities with respect to procurement, installation, and testing of these devices (see Detail A5.7-1).

SUMMARY/CONCLUSION

We found that calculation E-164-3 was performed and documented in a competent manner and was consistent with the requirements of Engineering Assurance Procedure EAP 5.3. We also found that the calculation was in agreement with Stone & Webster engineering memoranda in this technical area. The calculation was also in agreement with all vendor data. The team was impressed with Stone & Webster's knowledge of the issues and their detailed study of this area. We found the design to be in compliance with Regulatory Guide 1.106 and Final Safety Analysis Report commitments. We also found the procurement, installation, and testing of the subject device to be in good order. We concluded, based on our selective review, that this area was in good order.

RELATED FORMS:
A5.7-1 (Detail) Motor Control Centers

5.8 Power Generation Control Complex Design Changes

OBJECTIVE

The objectives of this portion of the inspection were to evaluate design change activities conducted by Stone & Webster, Gulf States Utilities, and General Electric personnel for the Power Generation Control Complex (PGCC) to determine the nature of those activities and the degree of conformance to design requirements and project procedures.

EVALUATION

The main control room is comprised of benchboard and vertical type panels, floor sections, interconnecting cabling, and field termination cabinets designated as the Power Generation Control Complex. Due to the complexity of the Power Generation Control Complex design and numerous interfaces involved, we focused our efforts on several randomly selected safety-related design changes to determine whether the process was conducted in a controlled manner. We reviewed design change documentation on Field Deviation Disposition Requests, Work Package Change Notices, Change Request Forms, Boundary Inspection Program Support Plans, Engineering and Design Coordination Report document tracking systems and associated electrical drawings (see Detail A5.8-1).

SUMMARY/CONCLUSION

We reviewed Field Deviation Disposition Request LDI-925 which was developed by Gulf States Utilities personnel. We found that Stone & Webster inadvertently missed incorporation of changes on elementary wiring diagrams in two cases (see Deficiency D5.8-1). This item had no affect on design and is considered to be minor. We reviewed Work Package Change Notice WPCN-786, Change Request Forms CF-0300 and CF-5068, Field Deviation Disposition Request LDI-1443, and associated Boundary Inspection Program support plans and Engineering and Design Coordination Report's. We found that all documents were in good order with the exception of a minor error on an analog Wiring diagram (see Deficiency D5.8-2). We then reviewed Change Request Form CF-0288, Field Deviation Disposition Request LDI-1331 and Engineering and Design Coordination Report C-51164. We found several errors on sketches and markup drawings attached to Field Deviation Disposition Request LDI-1331 (see Deficiency D5.8-3). We determined that these errors had no apparent impact on design.

During our site visit we reviewed the Power Generation Control Complex design modification activities of the General Electric Site Engineering Group. We reviewed the General Electric tracking scheme for changes to production design base documents. We checked the tracking scheme with respect to Field Deviation Disposition Requests LDI-1331 and 1443. We found the General Electric Engineering Information System computer data base accurately tracked assigned Engineering Change Notices against each design drawing. We also reviewed Change Request Form CF-0263 and found the revised drawings to be in good order.

In summary we found three deficiencies which were drawing and documentation type errors. These items had no effect on design. These errors are considered to be minor and infrequent based on our extensive review of drawings and documents during this inspection.

We concluded that the Power Generation Control Complex design change process was well documented and controlled.

RELATED FORMS:
A5.8-1 (Detail) Power Generation Control Complex Design Changes
D5.8-1 (Deficiency) Drawing Error
D5.8-2 (Deficiency) Drawing Error
D5.8-3 (Deficiency) Field Deviation Disposition Request Error

5.9 Electrical Separation

OBJECTIVE

The objective of this portion of the inspection was to examine the design provisions for electrical separation and physical independence of class 1E circuits.

EVALUATION

We reviewed and evaluated the design criteria and Final Safety Analysis Report commitments with respect to electrical separation, and randomly selected portions of the design to determine whether the criteria were correctly incorporated into the design documents and the field installation (see Detail A5.9-1).

SUMMARY/CONCLUSION

We determined that the Stone & Webster design criteria and the Electrical Installation Specification provide separation requirements pertinent to the design, installation and inspection activities. We found these documents to be complete and detailed. We determined that these documents appeared to be consistent with Final Safety Analysis Report requirements and criteria presented in IEEE -384 (1974). We reviewed non-class 1E loads supplied from class 1E buses and found that the required LOCA trip features were incorporated into the design. We randomly selected main control room termination cabinets and found that all cables were routed to their respective divisional bays as required by the separation criteria. We determined that Stone & Webster conducted a separation review program of cabling terminated at equipment; we reviewed several cases and found no deviation from the results of the Stone & Webster review.

We concluded, based on our limited review, that the design appears to be consistent with Final Safety Analysis Report requirements for physical separation and electrical isolation of circuits and raceways, thereby preserving electrical independence of redundant safety-related electrical systems. We also concluded that a controlled process is in place to resolve identified problems.

RELATED FORMS:
A5.9-1 (Detail) Electrical Separation

5.10 Terminal Blocks Inside Containment

OBJECTIVE

The objective of this portion of the inspection was to review terminal blocks within junction boxes located inside containment and to evaluate the environmental qualification of the terminal blocks.

EVALUATION

There are valid technical concerns regarding the ability of terminal blocks to provide sufficient insulation characteristics during and after exposure to accident environments. These concerns generally focus on the potential for increased leakage current, introduction of electrical noise on low level instrumentation signals, and temporary, intermittent or permanent electrical breakdown. These potential failure modes can result in loss of control circuits due to blown fuses or circuit breaker trips, or spurious and erroneous instrumentation signals.

Eleven General Electric supplied local instrument racks, located in-containment, have safety related equipment wired to divisional rack mounted junction boxes which contain terminal blocks. We therefore reviewed the documentation to determine whether the terminal blocks were qualified to perform their design basis safety function (i.e. provide sufficient insulation).

SUMMARY/CONCLUSION

We found that although General Electric had performed various tests on terminal blocks (GE type CR151B), the documentation was inadequate and incomplete to support qualification for use in the containment (see Deficiency D5.10-1). We also found that Stone & Webster did not provide technical details to the electrical installer to meet General Electric's stringent sealing requirements for conduit entry on terminal boxes mounted on in-containment General Electric instrument racks. We found installed conduit which violated General Electric sealing requirements (see Deficiency D5.10-2). We concluded that General Electric's difficult sealing requirements were not emphasized to Stone & Webster in a timely manner. However, the requirements were included in General Electric documents. We also concluded that Stone & Webster did not provide adequate attention to this critical interface.

During the inspection, Gulf States and Stone & Webster made the decision to bypass these in-containment terminal blocks thereby eliminating our initial concerns over sealing of terminal boxes and qualification of terminal blocks.

We found inconsistencies within General Electric documents concerning equipment classification for transmitter E31-N092. This is considered a minor documentation discrepancy (see Deficiency D5.10-3).

RELATED FORMS:
D5.10-1 (Deficiency) Terminal Block Qualification
D5.10-2 (Deficiency) Conduit Sealing
D5.10-3 (Deficiency) Documents Inconsistent

5.11 D.C. Power System

OBJECTIVE

The objective of this portion of the inspection was to review the calculations performed by Stone & Webster to determine if the 125 Volt Class 1E batteries were properly sized to perform their safety function and were properly installed following approved installation procedures and quality assurance requirements.

EVALUATION

The team selected 125 Volt Class 1E batteries IEN*BAT01A and IEN*BAT01B, which supply dc power to all safety-related dc components before, during and after a design basis event. These batteries are located in the control building in their respective divisional battery rooms. We reviewed applicable calculations E-149 and E-150, the welding procedure and the quality assurance inspection plan, and Stone & Webster Engineering Assurance procedure EAP 5.3.

Calculations E-149 and E-150 were performed to verify the 125Volt battery size, assumptions, input-document references, and the necessary analysis to justify the selection of Gould size NCX-2100 for the Class 1E battery supply. The calculation method principally followed IEEE 485 methods and associated work sheet. The calculation format and procedure followed Stone & Webster Engineering Assurance procedure EAP5.3. It was observed that the manufacturer's data reference was not included in the input document list of the calculation. The procedure requires listing of all documents related to or supporting the calculation (see Deficiency D5.11-1).

The team noted that the amperes per positive plate values used in the work sheet to calculate the number of positive plates (cell size) had no documented basis. The engineer stated that he received these values by telephone from Gould. However, no telephone memorandum was identified and no record of this data from the manufacturer could be located. This data is the governing factor in determining the number of positive plates in a cell (cell size) (see Deficiency D5.11-1).

The team inspected the installation of these two batteries. We found that welding of the battery racks was performed according to Stone & Webster welding specification and sill detail. We checked the installer's report and the quality assurance inspection plan for the welds. We found that the weld checking was visually performed in accordance with the quality assurance inspection plan, as allowed by American Welding Society Standard AWSD-11 for the structural welding. The team found two spots in each rack where the vertical section of the rack angle was not welded to the painted sill. These were spots where the sill grout holes are under the rack angle. The team noted that two changes provided exemption to the requirement for welding above grout holes.

SUMMARY/CONCLUSION

Two minor discrepancies were found. In one, the Engineering Assurance Procedure was not followed in that a reference for important data used in the calculation was not listed. In the other deficiency, undocumented data was used to size the battery which may have affected the validity of the calculation. However, the revision of the calculation that was performed by Stone & Webster in response to our review indicates that the battery size was not affected. The installation was found to have followed procedures. The team concluded that the seismic analysis of the battery rack provided sufficient assurance of the adequacy of the rack design and the seismic capability of the battery installation.

RELATED FORMS:
A5.11-1 (Detail) References for Section 5.11
D5.11-1 (Deficiency) Use of Undocumented Data

5.12 Diesel Generator Lube Oil Pump Motor

OBJECTIVE

The objective of this portion of the inspection was to determine whether Class 1E diesel generator auxiliary pump motors were correctly classified for their application by the supplier (Delaval) and Stone & Webster.

EVALUATION

The safety classification and the power supply to the ASME Code III and seismic Category I lubricating oil circulating pump motor (before and after pump) and lube oil heater were evaluated. We reviewed the requirements of the FSAR and guidelines of the Standard Review Plan and NRC NUREG/CR-0660. The team also reviewed the diesel generator specification, Delaval piping schematic, and the position letter and meeting notes between Stone & Webster and Delaval (see Unresolved Item U5.12-1).

SUMMARY/CONCLUSION

This item requires further evaluation by Gulf States Utilities of Delaval's position. If necessary in light of developing programs, modifications can be provided to assure that the prelube circulating oil pump motor will not fail, or if it fails, that there shall be reliable indication of its failure.

RELATED FORMS:

U5.12-1 (Unresolved Item) Safety Classification of the Diesel Generator Pre-Lube Oil Pump

6.0 INSTRUMENTATION AND CONTROL

The objective of this portion of the inspection was to review the instrumentation and control aspects of the Residual Heat Removal and Automatic Depressurization Systems in the Nuclear Steam Supply System scope and the Standby Service Water System in the Balance-of-Plant scope. During the inspection, a portion of the High Pressure Core Spray System involving cooling of its diesel generator was added to the review.

Design information reviewed included both general and system specific design criteria, functional requirements, piping and instrument diagrams, flow diagrams, logic diagrams, and related detailed design documentation. The review concentrated on the control of design information for these particular systems among Gulf States Utilities, Stone and Webster, and General Electric. The scope of the review extended from the reactor vendor's design input through the installed instrumentation and control equipment at the River Bend Station.

6.1 Design Information

This section summarizes the instrumentation and control design process among Gulf States Utilities, Stone & Webster, and General Electric Company.

The team conducted reviews with the Gulf States Utilities Site Engineering, Startup, and Operations Groups; Stone & Webster Site Engineering, Electrical/Controls Technical Support, Electrical/Controls Field Support, Power (i.e. Mechanical), Building Services, Nuclear Analysis, and Equipment Qualification Groups, and the General Electric Control and Instrumentation Group.

The team was frequently referred to the FSAR for design criteria and design basis information relating to instrumentation and control for safety-related Balance-of-Plant systems. This Stone & Webster practice for the River Bend project differs from typical nuclear industry methods. In particular, many of the design bases needed to review and assess the adequacy of these systems could only be inferred by examination of the design.

Instrumentation and control requirements and recommendations provided by General Electric were traced through to Balance-of-Plant design output documents produced by Stone & Webster. Similarly, design requirements for the Standby Service Water System were traced from the River Bend FSAR through the Stone and Webster design output documentation. Implementation of these various requirements was reviewed during a walk-down of each of these systems at the River Bend Station.

A variety of design documents produced by General Electric and Stone & Webster, such as the master parts list, system design specifications, design criteria documents, piping and instrument diagrams, flow diagrams, functional control diagrams, logic diagrams, loop diagrams, elementary diagrams, component procurement specifications, environmental design criteria, instrument data sheets, and setpoint calculations were reviewed during the inspection.

The means used to establish and maintain instrumentation and control interface requirements with other disciplines in each inspected organization, and the use of instrumentation and control design outputs, both within the organization and by external organizations, were reviewed. These included Stone & Webster design control procedures used by the Electrical/Controls Technical Support group. The team examined exchange and control of design output information at interfaces with the Power group (flow diagrams), Building Services (Heating, Ventilation, and Air Conditioning), Nuclear Analysis (Failure Modes and Effects Analysis, setpoint calculations), Electrical/Controls Field Support group (schematic diagrams, panel arrangements), and with the Environmental Qualification group. Instrumentation and control technical interfaces with the Stone and Webster Site Engineering group and with Gulf States Utilities Site Engineering group were also reviewed. The means used to downgrade the classification of a component and to downgrade particular requirements placed on a component were reviewed.

6.2 Personnel and Guidance

(a) Gulf States Utilities Personnel

A sample of instrumentation and control activities performed by key individuals assigned to the Gulf States Utilities Site Engineering Group at the River Bend Station was reviewed during the inspection. The Group Manager has Mechanical and Nuclear Engineering degrees, a Professional Engineer license, and 16 years of nuclear experience. Seven of the eight Gulf States Utilities employees in this organization have engineering degrees, and their average experience level is approximately 5 years. The team had no further questions in this area.

(b) Stone & Webster Personnel

A large number of personnel involved in instrumentation and control activities at both the Cherry Hill Operations Center and at the River Bend Site were contacted during this inspection. The Electrical/Controls Technical Support Group in Cherry Hill comprises approximately 20 individuals with virtually all having technical degrees. The Group Lead Engineer has BSEE and MSEE degrees, a Professional Engineer license, and approximately 10 years of nuclear experience including 8 years on the River Bend project. Three principal engineers in this group have comparable educational and experience backgrounds with approximately 6 years on the River Bend or NMP-2 Boiling Water Reactor projects. Key individuals responsible for instrumentation and control activities performed by the Site Engineering Group have extensive and relevant Boiling Water Reactor and Pressurized Water Reactor experience. The team had no further questions regarding these Stone & Webster groups.

Stone & Webster design procedures used at Cherry Hill and at the River Bend site appeared to be complete and comprehensive. Their use in accomplishing design work at both locations was evident in all of the documents reviewed. The team had no further questions regarding Stone & Webster design procedures or their use on the project.

6.3 General Electric

(a) Objective

The objective of this portion of the inspection was to evaluate the adequacy of control and instrumentation (C&I) design, implementation of the design control procedure, and compliance to the FSAR commitments for the Low Pressure Coolant Injection mode of Residual Heat Removal System and the Automatic Depressurization System.

(b) Design Information

General Electric is the designer and supplier of Nuclear Steam Supply System and the control room to the River Bend generating station. General Electric is also responsible for the installation of the Power Generation Control Complex and controls the interface between the control room and field cables designed by Stone & Webster. General Electric maintains a site engineering office to resolve the field problems and to forward items for in-depth review and design changes to their Nuclear Energy Division head office at San Jose, California. The reactor instrumentation and protection design group in Controls and Instrumentation Equipment Systems Engineering is responsible for elementary diagrams, panel design, connection diagrams and cable lists. The elementary diagrams are the base line product of the controls and instrumentation group. These are based on the system process diagrams, P&IDs, system design data sheets and system specifications. These design documents are prepared by the lead system engineers in the Nuclear Power Systems Engineering Department. Equipment qualification is handled by a separate group reporting to the Nuclear Controls and Instrumentation Product Design Operations Manager.

(c) Personnel

There are two controls and instrumentation engineers at the head office who develop elementary diagrams for the Residual Heat Removal System and Automatic Depressurization System. The connection diagrams, cable lists and panel design are handled by one engineer for the River Bend project. An overall controls and instrumentation design and site interface is carried out by one engineer assigned to this project. Site engineering support was provided by a five engineer team until 1983. Most of the field deviations, redesign and engineering changes were referred to the head office during this period. General Electric established an on-site engineering group beginning in 1984. This group expedited field activities, including closing most of the field change requests, control room interfaces and changes, and provided test, start up, and engineering support.

(d) Evaluation

The team reviewed the design record file for the Automatic Depressurization System and Residual Heat Removal System. This review included Residual Heat Removal System pump motor specification; Automatic Depressurization System and Residual Heat Removal System design specifications, design specification data sheets, Master Parts List, functional control diagrams, P&ID, and the design bases for General Electric's scope of the River Bend project. The team examined if both systems are designed to meet the General Electric scope of the design bases and if the FSAR commitments to regulatory guides and standards are adequately addressed in General Electric documents. The team noted that General Electric does not include Regulatory Guide 1.97 "Post Accident Monitoring," in the design bases for River Bend controls and instrumentation system design. The licensee and Stone & Webster have identified the components for meeting Regulatory Guide 1.97 commitment to General Electric. General Electric has included these components in their phase 3 qualification program. The team noted a documentation discrepancy in that the design basis document indicated many IEEE standards and regulatory guides which are included in the FSAR are not to be applied to the River Bend project until an Engineering Change Authorization or Engineering Change Notice directs implementation. No Engineering Change Authorization or Engineering Change Notice was identified by General Electric. This document was issued in October 1981 and was applicable at the time of our inspection (see Deficiency D6.3-1).

The team also reviewed the process of an independent review by a change control board consisting of off-project experts. The items chosen for this review were Emergency Core Cooling System/Low Pressure Coolant injection valve interlock and Automatic Depressurization System logic modifications designed and proposed by General Electric in response to NUREG-0737 (see Detail A6.3-1).

The team reviewed the elementaries and the connection diagrams of the two systems on sample basis. The connection diagrams follow the wire list, electrical device list or instrument data sheet, and the standard assembly drawing (component layout on the panel). During this review emphasis was on the compliance with Regulatory Guide 1.75 and General Electric design specification MPL #A62-4050 for physical and electrical independence of the electrical equipment and relay logics circuitry. The team noted a minor error on one of the Residual Heat Removal System elementaries that required correction on the drawing without any hardware change (see Deficiency D6.3-2).

The team reviewed the purchase specification of Residual Heat Removal System pump motors, the general purchase specification of Rosemont transmitters and qualification documents of Crosby Automatic Depressurization System valves and Rosemont transmitters. The motor specification included motor data sheet (General Electric, generic), pump data from the pump manufacturer (Byron Jackson), and the qualification parameters. Qualification of Crosby Automatic Depressurization System valves is to be completed under GE's Phase 3 qualification program in compliance to NUREG-0588. While reviewing the qualification documents for the Rosemont transmitters, the team noted that the power supply to the conductivity indicator transmitter was downgraded from a Class 1E supply to a non-Class 1E supply. The analysis, Engineering Change Authorization and Engineering Change Notice were checked to establish the validity of the change. The change was found to be correct and the change process was controlled (see Detail A6.3-1).

The team reviewed the process of establishing trip setpoints of Rosemont transmitters. The Automatic Depressurization System valve trip and Residual Heat Removal/Automatic Depressurization System permissive trip setpoints were checked in the design specification data sheet. The team found an Engineering Change Authorization in process for providing changes in the data sheet. These changes account for the Rosemont trip units set points, accuracy and ranges. Values are determined by the lead system engineer and are provided in the data sheet. The team checked the instrument ranges at the site with the data sheet for a Residual Heat Removal System instrument rack and found them within the actual ranges. Pre-operational testing of the system requires instrument calibration. Therefore these instruments will be calibrated and set for the data sheet requirements.

The team reviewed various field changes at the site. The team noted that certain instruments from a Residual Heat Removal System instrument rack had to be relocated due to excessive elevation difference between the root valve, high point vent and the instruments. The team reviewed the necessary design change interface documents and the actual installation and found them in order. During this review the team noted that the General Electric essentiality classification of a pressure transmitter was changed from an active to a passive safety function while the power supply remained as Class 1E. We found this change to be documented and controlled (see Detail A6.3-1).

The team reviewed General Electric elementary diagram 828E534AA and Stone and Webster sketch ESK-5-RH501 to determine if the manual override capability of the Engineered Safety Features Actuation System meets the requirement of IEEE 279.

The drawings indicate that the Engineered Safety Features Actuation System can be overridden by a manual control switch at the main control board at the component level (not at the system level). Motor and valve starting by Engineered Safety Features Actuation System signal can be prevented by the manual override switch. The team determined that the system design complies with the requirement of IEEE 279.

The team reviewed General Electric Emergency Core Cooling System logic 24Vdc power supply for its divisional association and qualification. The Division A supply is associated with the Low Pressure Core Spray System and is derived from the Division 1 battery/inverter 120Vac power supply. The Division B dc power supply is associated with a portion of Residual Heat Removal System and is derived from Division 2 battery/inverter, 120Vac supply. The transformer/rectifier circuitry packages are located in the control room and were procured as commercial grade units from General Electric or SOLA. The General Electric equipment qualification division has an in-house program to qualify these components for usage in safety grade systems (see Detail A6.3-1).

(e) Summary/Conclusion

On the basis of the sample included in the inspection, the design process at General Electric's San Jose facility and the design change and installation process by their River Bend site establishment appeared to be controlled in the instrumentation and control area. General Electric's documentation procedures, specifications and drawings were detailed and understandable. The sample review indicated that General Electric design is in compliance with the basic criteria specified in the FSAR for River Bend project in the area of instrumentation and control. The two documentation discrepancies noted in the course of inspection did not affect any component or require correction by design change. The team concluded that General Electric's scope of instrumentation and control system design for the River Bend Station was controlled.

(f) References

References for section 6.3 are listed on Detail A6.3-2.

RELATED FORMS:
A6.3-1 (Detail) Design Record Review, Specification and Qualification of Equipment, Field Design Changes, 24Vdc Power Supply Qualification
A6.3-2 (Detail) Reference List for Section 6.3
D6.3-1 (Deficiency) Design Bases Inconsistency with the FSAR
D6.3-2 (Deficiency) Incorrect Information on GE Drawing

6.4 Functional and Detailed Design Requirements for the Balance-of-Plant Protection System and ESF Actuation System

OBJECTIVE

For safety-related Balance-of-Plant and Engineered Safety Features (ESF) systems, the documentation of appropriate functional and detailed design requirements is an important element for achieving compliance with industry standards, codes, and regulatory requirements. Documentation of these requirements provides implementation guidance to the designer, and facilitates design review activities in assessing the design relative to its design basis requirements. Therefore, the methods and practices used by Stone and Webster to document and use such functional and detailed design requirements were examined.

EVALUATION

The Electrical Independence Design Criteria document, as well as system specific flow diagrams, logic diagrams, loop diagrams, analog wiring diagrams, and elementary diagrams were reviewed for documented functional and detailed design requirements. In addition to the FSAR functional descriptions and licensing commitments, certain procurement documents were also reviewed for the translation of such requirements to component vendors. A number of discussions were held with Stone & Webster and Gulf States Utilities personnel on this topic.

The Division 3 High Pressure Core Spray diesel generator cooling flow path from the Standby Service Water System contains four motor-operated valves that are manually controlled from the main control room. Since inadvertent closure of these valves could quickly cause the loss of one Emergency Core Cooling System train, the absence of automatic valve opening upon receipt of a LOCA signal was investigated. Standby Service Water System flow diagrams, logic diagrams, and elementary diagrams were reviewed. General Electric criteria for the High Pressure Core Spray System and the Emergency Core Cooling System network were also reviewed. A number of individuals at Stone & Webster and General Electric were contacted to determine the basis for use of manual control for positioning of these cooling valves. The operator's displays on panel 1H13*P601 were reviewed both on drawings and during the plant walk-down.

SUMMARY/CONCLUSION

Stone & Webster uses the River Bend FSAR as the primary means to fulfill the IEEE Std. 279-1971 Section 3 requirement for design basis documentation of functional and detailed design requirements. Normal industry practice is to provide a documented design basis in the form of system design specifications, system descriptions, or other types of controlled design documents. The River Bend FSAR is not a controlled design document, and it does not provide the complete design basis in an explicit manner. Hence, Stone & Webster has neither controlled nor consolidated the set of specific performance requirements to be satisfied by the design as required by IEEE Std. 279-1971.

In the Stone & Webster design, motor-operated valves 1SWP*MOV77A and B and 1SWP*MOV506A and B are manually controlled from control room panel 1H13*P601. An accident signal is not used to provide automatic opening of the valves should they be inadvertently closed, nor are additional administrative controls provided in the design to augment the operator's actions so as to preclude this possibility. Since operator action would be required in approximately two minutes following a postulated LOCA in the event the valves have been left inadvertently closed, the basis for Stone and Webster acceptance of operator administrative control was sought. The team was unable to identify or locate pertinent background information addressing this question.

The team also observed that the High Pressure Core Spray Division 3 independence requirement relative to other plant systems stated in General Electric criteria documents has not been satisfied in the River Bend design. A reduction in predicted reliability of the Emergency Core Cooling Network is anticipated because of the River Bend system's dependence on other on-site class 1E AC power sources (see Observation O6.4-1).

RELATED FORMS:
D6.4-1 (Deficiency) Standby Service Water Manual Valves for HPCS D/G Cooling
D6.4-2 (Deficiency) Logic Sketch Error in Valve Assignments
O6.4-1 (Observation) HPCS Division 3 Dependency Upon ESF Divisions 1 and 2

6.5 Balance-of-Plant Protective Action System

OBJECTIVE

The architect-engineer is provided with redundant Engineered Safety Feature Actuation Signals (i.e. identified as Engineered Safety Features Actuation System or LOCA signals) indicative of an accident situation so that emergency diesel generators, pumps, valves, coolers, and filter units may either be started or placed in a state of readiness to respond to accident mitigation needs. Hence, the technical interface between the Nuclear Steam Supply protection system designed by General Electric and the Balance-of-Plant protective action system designed by Stone & Webster was inspected for design process control.

EVALUATION

Starting with the redundant LOCA signals provided by General Electric from relays 1E12-K110A and B, Stone & Webster logic diagrams and elementary diagrams were reviewed through their relay actuation circuits to the actuated equipment. Particular emphasis was placed on automatic initiation of diesel generators 1EGS*E1A and B used to provide emergency on-site power to equipment in separation Divisions 1 and 2. Although primary emphasis was placed on the automatic initiation circuits, the review included power source connections, manual initiation circuits, and related operator displays and controls. Beyond the NSSS-Balance of Plant Engineered Safety Feature Actuation Signal interface discussed in subsequent paragraphs, Stone and Webster design documents were examined for the Standby Service Water System and several related systems.

SUMMARY/CONCLUSION

Consistency and accuracy were observed in flow diagrams, loop diagrams, logic diagrams, analog wiring diagrams, and elementary diagrams from sensors through to the actuated equipment. Three random errors were observed involving a resistance temperature detector classification designation, a resistance temperature detector procurement specification temperature limit, and an AC power assignment to manual valves in the High Pressure Core Spray diesel generator cooling loops.

A significant technical error was noted in the Stone & Webster design at the NSSS-Balance of Plant interface in that redundant, but interruptible, Class 1E AC power sources were chosen for the relay contact multiplication circuits used to actuate emergency power sources and loads. For this interface, the only acceptable power sources are either the redundant Class 1E DC station batteries or the redundant Class 1E uninterruptible AC power supplies derived from the station batteries. During the inspection, Stone & Webster processed a design change to replace the original power sources with redundant Class 1E uninterruptible AC sources, which is a satisfactory resolution for this particular concern. However, the error had remained undetected through many design reviews over a ten year period, and was of concern to the team (see Deficiency D6.5-2).

RELATED FORMS:
D6.5-1 (Deficiency) RTD Classification Drafting Error
D6.5-2 (Deficiency) Standby Diesel/Generator Initiation on LOCA
O6.5-1 (Observation) Instrumentation Separation within BOP ESF Systems

6.6 Balance-of-Plant Instrument Setpoint Calculations

OBJECTIVE

During recent inspections, significant differences have been observed in architect-engineer design practices regarding safety-related Balance-of-Plant instrument setpoint calculations. Consequently, design control practices used by Stone & Webster for setpoint calculations were inspected in detail.

EVALUATION

Setpoint calculations for low pressure initiation of Standby Service Water from measurements taken at the Normal Service Water header and the Reactor Plant Component Cooling Water header were selected for examination.

SUMMARY/CONCLUSION

A comprehensive setpoint calculation methodology and procedure was established by Stone & Webster for the River Bend project in 1981 that clearly states that a setpoint calculation is the only source document for setpoint values. This Stone & Webster position was reiterated by an informal memorandum issued during the inspection. During the inspection, a number of Stone & Webster engineering documents provided safety-related setpoint values that were inconsistent with one another. To reduce the potential for error in subsequent use of these setpoint valves, the team believes that design documents listing such setpoint values should also contain a note to refer users to the setpoint source document.

One calculation did not follow the established procedure; however, after being revised during the inspection, it now conforms to the procedure and also reflects technical input information regarding radiation exposures. The other calculation has not yet been revised to reflect this same radiation exposure information; however, the present setpoint value will not be affected.

In each instance, the selected setpoint value appeared to be reasonable and consistent with system operating constraints established by the Stone and Webster Power Group. The examined calculations were comprehensive, controlled, thoroughly documented, and technically conservative with regard to assumptions and anticipated equipment performance. The Cherry Hill Operations Center is tasked to perform any needed setpoint calculation revisions on a priority basis to support pre-operational and startup testing requirements. The team had no further questions in this area.

RELATED FORMS:
D6.6-1 (Deficiency) Instrument Setpoint Documentation Inconsistencies
D6.6-2 (Deficiency) Instrument Setpoint Calculation Assumptions
D6.6-3 (Deficiency) Instrument Procurement Specification Inconsistencies

6.7 Periodic Testing to Demonstrate Standby Service Water System Availability

OBJECTIVE

In the River Bend design, the non-safety-related Normal Service Water System provides needed cooling flow to Engineered Safety Features equipment as long as off-site power remains available. The safety-related Standby Service Water System only initiates operation when either Normal Service Water or Reactor Plant Component Cooling Water header low pressure is encountered. This design approach is in contrast to other safety systems and Engineered Safety Features, such as the diesel generators, in that the operational availability of the Standby Service Water System is not confirmed immediately following a LOCA. Periodic test design provisions implemented to assure high availability of the Standby Service Water System were inspected.

EVALUATION

Using Stone & Webster design documents, the periodic test capability provided for the Standby Service Water System during normal power operation was assessed relative to the commitments to NRC Regulatory Guide 1.22 "Periodic Testing of Protection System Actuation Functions" (Safety Guide 1.22). Intended demonstration of these capabilities by means of the Standby Service Water System pre-operational test procedure currently under development by Gulf States Utilities was also reviewed.

SUMMARY/CONCLUSION

From a review of the design documents, the Standby Service Water System is provided with the capability to test pumps and valves independently during power operation in conformance with FSAR Chapter 7 and 9 commitments.

Based on discussions with Stone & Webster personnel regarding this testing capability, it appears that operation of one Standby Service Water pump concurrent with continued operation of the Normal Service Water System can be used during full power operation to more fully demonstrate Standby Service Water System availability (except for the Spent Fuel Pool Cooling heat exchanger loop which must remain isolated).

Under these circumstances, each system would contribute partial flow based on their individual pressure drop characteristics. Demonstration of partial flow from the Standby Service Water System during power operation should provide a greater assurance that the system will actually be available when needed.

Performance of this concurrent system test is not included in the pre-operational test procedure, and subsequent discussions with both Stone & Webster and Gulf States Utilities personnel have indicated that this mode of operation has only been considered for the surveillance test performed during refueling outages. As a result, the on-line demonstration of Standby Service Water System availability is, at present, an unresolved item for this inspection.

RELATED FORMS:
U6.7-1 (Unresolved Item) Periodic Test of the Standby Service Water System

6.8 Control Systems

OBJECTIVE

Functional requirements for Balance-of-Plant control systems are delineated on Stone and Webster logic diagrams. The team reviewed the particular control system design practices used by Stone & Webster. Emphasis was placed on the identification of potential adverse interactions between these control systems and the plant safety systems.

EVALUATION

Control system aspects for the Standby Service Water, Normal Service Water, and Reactor Plant Component Cooling Water Systems were reviewed. Controls related to the Standby Cooling Tower Fans, Standby Service Water Pump House Room fans, Cooling Tower Switchgear Room temperature instrumentation, and the Yard Structures Ventilation System instrumentation were also reviewed. Implementation of a selected portion of these functional requirements on detailed electrical circuit schematics was then reviewed.

SUMMARY/CONCLUSION

Based on this drawing review, the team concluded that design control of these control system aspects were comparable with those used for the Balance-of-Plant safety systems. Implementation details satisfied the documented functional requirements in each situation examined. No adverse interactions between the control systems and the plant safety systems were identified. Appropriate consideration of control system requirements needed by the Stone & Webster Power group was noted between the operating characteristics of tornado dampers and controls for particular ventilation fans. The team had no further questions in this area.

RELATED FORMS:
None

6.9 Balance-of-Plant Indication and Annunciation Systems

OBJECTIVE

Main control room display indications and annunciation alarms provided by the River Bend instrumentation and control design for the selected systems were reviewed during this inspection. Particular emphasis was placed on accident monitoring instrumentation and indication of bypassed or inoperable status of safety-related systems relative to commitments to NRC Regulatory Guides 1.97 and 1.47 respectively.

EVALUATION

Measurements of Engineered Safety Feature Supporting System cooling flow and temperature, needed to monitor heat transfer to the ultimate heat sink, were selected to assess Stone & Webster design conformance with NRC Regulatory Guide 1.97 commitments. Other system status measurements, such as emergency AC and DC bus voltage and bus current and pneumatic supply pressure, were also investigated.

SUMMARY/CONCLUSION

Following publication of NRC Regulatory Guide 1.97 revision 2 in late 1980, certain system status monitoring instruments (i.e. equipment for Type D variables subject to Qualification Category 2 requirements) have been subject to additional requirements beyond the commercial grade equipment ordinarily used in non-safety-related instrumentation applications. Stone & Webster makes only two designations; namely, Quality Assurance Category I (i.e. safety-related including Class 1E) or Quality Assurance Category II (i.e. non-safety-related). Requirements for some of these Type D variables fall in between these well-established Class 1E and non-Class 1E designations.

Even though a FSAR commitment has been made with regard to the variables to be monitored, the designation of which instrument satisfies a particular accident monitoring variable has not been made. Relevant accident monitoring instrumentation design information is provided in mid-1982 letters between Stone & Webster and Gulf States Utilities. Accident monitoring instrumentation design requirements are not depicted on controlled River Bend design documents.

RELATED FORMS:
D6.9-1 (Deficiency) Balance-of-Plant Accident Monitoring Variables

6.10 Balance-of-Plant Instrumentation Installation

OBJECTIVE

The purpose of this activity was to inspect a selected portion of the installed instrumentation for: (1) conformance with FSAR commitments and specialized technical requirements contained on design drawings; (2) provisions for physical separation and electrical isolation devices where required; (3) the physical location and arrangement of the instrumentation in relation to operating and maintenance needs, and (4) the absence of potential hazards in their immediate vicinity.

EVALUATION

Low header pressure instrument tubing and transmitters in the Normal Service Water and Reactor Plant Component Cooling Water Systems used to initiate operation of the Standby Service Water System were inspected. Standby Service Water pumps, valves, motor control centers, and instrument taps for Class 1E flow measurements were also inspected. High Pressure Core Spray diesel generator cooling loop motor-operated valves were inspected for their individual electric power separation division assignments. Main control room operator controls, displays, alarms, instrument loop components, and related relay panels for the Standby Service Water System and its High Pressure Core Spray diesel generator cooling loop were examined.

SUMMARY/CONCLUSION

The inspected equipment and instrumentation conforms to the Stone & Webster design drawings with the exception of one instrument rack, 1JPB*RAK1, provided by Mercury of Norwood, Massachusetts. As this error had been detected and corrected by the Site Engineering Group, the team had no further questions in this area.

Physical separation and electrical isolation aspects appeared satisfactory, and accessibility for instrumentation maintenance and repair activities was adequate.

The location of Standby Service Water instruments appeared free of potential hazards. Hence, the team had no further questions regarding installation of the inspected instrumentation.

RELATED FORMS:
O6.10-1 (Observation) Reversed Instrument Positions on Mercury (Norwood) Rack

6.11 Balance-of-Plant Instrument Environmental Qualification

OBJECTIVE

Consistency and technical adequacy among the Stone & Webster design documents, instrument procurement specifications and their related technical data sheets, and the environmental design criteria being used for the Environmental Qualification Program were examined.

EVALUATION

A limited sample of instruments and equipment related to the Standby Service Water System and its High Pressure Core Spray diesel generator cooling loops was selected for this comparison. A recent Gulf States Utilities letter authorizing Stone and Webster use of unissued environmental design criteria as an interim measure was reviewed and discussed with responsible Gulf States Utilities personnel.

SUMMARY/CONCLUSION

The current state of environmental qualification activities for the River Bend Station is typical of most nuclear plants under construction, and the limited sample inspected indicates that appropriate requirements are being placed on instrumentation and control equipment. Several aspects are of concern since they could impact the overall technical quality of this work:

(a) Stone & Webster has been directed by Gulf States Utilities to use an unissued version of the Environmental Design Criteria document to expedite the accomplishment of environmental qualification work. The existence of errors in abnormal temperature values in the current document was the stated reason for this action. The team was concerned that normally well-controlled design work by various technical disciplines is being performed in conjunction with an uncontrolled environmental design criteria document. A preferred approach is to correct and reissue the document for use in these controlled design activities;

(b) In one instance, incorrect normal ambient temperature and humidity values were used in enveloping a number of components in various plant locations; and

(c) The capability to designate component safety functions is provided in the Environmental Design Criteria document; however, no entries were observed in these columns during the inspection. Knowledge of these safety functions is an essential element for an effective environmental qualification program, such as for the important-to-safety accident monitoring instrumentation.

RELATED FORMS:
D6.11-1 (Deficiency) EQ Analysis Error in Normal Environment Enveloping

6.12 Balance-of-Plant Instrumentation Pre-operational Testing

OBJECTIVE

The Stone & Webster design philosophy does not actually confirm the operational availability of the Standby Service Water System prior to an actual need. Hence, pre-operational test procedures that would confirm those design provisions needed to assure high availability of this system during power operation were inspected.

In addition, performance of actual in-situ instrument calibrations by Gulf States Utilities personnel was examined.

EVALUATION

A sample of instrument calibration records for Standby Service Water initiation using Normal Service Water and Reactor Plant Component Cooling Water low header pressure transmitters was reviewed. An unissued draft pre-operational test procedure for the Standby Service Water System was briefly reviewed, and its content was discussed with both Gulf States Utilities and Stone & Webster personnel. The Turbine Plant Component Cooling Water System Flush Procedure was also reviewed during the inspection.

SUMMARY/CONCLUSION

The team was unable to form definitive conclusions in the on-site testing area because of the draft status of pre-operational test documents and the limited number of available records for calibration of instruments.

The instrument calibration records inspected appeared to be well documented. The team felt uncomfortable with one calibration record for 1CCP*PT1A because it demonstrated identical transmitter output values for both increasing or decreasing pressure input conditions, whereas similar records for 1CCP*PT1B and C showed an expected amount of output hysteresis. This concern was discussed with on-site personnel (see Observation O6.12-1).

RELATED FORMS:
O6.12-1 (Observation) Instrument Calibration Hysteresis

6.13 Balance-of-Plant Instrumentation and Control Design and Field Changes

OBJECTIVE

Stone & Webster and Gulf States Utilities methods and procedures used to implement Balance-of-Plant design changes and field changes were examined.

EVALUATION

At the Cherry Hill Operations Center, a number of Stone & Webster implemented design changes were examined. Stone & Webster Site Engineering Group procedures were reviewed, and several examples of design changes implemented in the field were also examined. Discussions were held with the Gulf States Utilities on-site engineering group regarding their activities. In particular, their involvement in a new requirement for loose parts monitoring instrumentation was traced from the NRC letter through to preparation of a draft procurement specification by Stone & Webster.

SUMMARY/CONCLUSION

Stone & Webster procedures for control of revisions to issued documents were deemed satisfactory and are being used to implement design changes. Those design changes specifically examined during the inspection were generally correct and complete. In two particular instances, required changes to design documents had not been adequately implemented or controlled (see Deficiencies D6.9-1 and D6.6-2).

Site Engineering Group procedures were deemed to be as comprehensive as the Cherry Hill procedures, and in the instrumentation area, were limited to two types of engineering activity directly in support of construction. Those field initiated instrumentation and control design changes inspected appeared correct and complete. The team had no further questions regarding the Stone & Webster Site Engineering Group.

The Gulf States Utilities Site Engineering Group stated that they do not have procedures to either prepare or modify specifications or calculations. For the loose parts monitoring instrumentation issue, Gulf States Utilities documented their functional and commercial requirements for Stone & Webster's use in developing the procurement specification. Gulf States Utilities then has formal approval of the specification prior to actual procurement in accordance with Stone & Webster's procedure. The team had no further questions regarding the Gulf States Utilities Site Engineering Group or its activities.

RELATED FORMS:
D6.13-1 (Deficiency) Instrument Change Revision Notice Calculation Reference

7.1. Meetings

During the course of the inspection frequent meetings were held to discuss the preliminary inspection findings. The team leaders held meetings twice per week to brief Gulf States and Stone and Webster personnel. The entire team met with applicant, Stone and Webster, and General Electric personnel several times during the inspection to overview organizations, facilitate understanding of the design process, explain the purpose, scope and conduct of the inspection, and to discuss findings. The following describes the general purpose of the various meetings. Table 7.1 is provided as a matrix of meeting attendance.

Meeting #1. The NRC explained the purpose of the Integrated Design Inspection program at this meeting. Background material for several candidate systems was requested. A tentative schedule for the inspection was presented. The meeting was held at NRC's Region IV Office on February 24, 1984.

Meeting #2. The team entrance meeting was held at the site on April 9, 1984. Gulf States Utilities presented their organization and described their involvement in the design process.

Meeting #3. Stone and Webster Engineering Corporation presented an overview of the design process used for the River Bend Project. This meeting was held at the site on April 10, 1984. Specifics of the project organization were discussed. Interfaces with General Electric and Gulf States were also addressed.

Meeting #4. General Electric Company discussed the engineering and design for the River Bend project at the site on April 11, 1984. Organizations, design procedures and approaches, design review, change control, etc. were addressed.

Meeting #5. Stone and Webster Engineering Corporation presented an overview of their corporate and Cherry Hill Operations Center organizations and responsibilities on April 12, 1984. The majority of the Stone and Webster production engineering work for River Bend was done at Cherry Hill.

Meeting #6. On June 1, 1984 an exit meeting was held at the site to discuss the inspection findings and preliminary conclusions.

Meeting #7. A meeting was held on June 20, 1984 to discuss the Reactor Shield Building design for shear reinforcement (see Chapter 4 for details).

TABLE 7.1

Meeting Attendance

Name	Organization	Title	Meeting Attended 1	2	3	4	5	6	7
D. Allison	NRC, OIE	IDI Team Leader	X	X	X	X	X	X	X
R. Architzel	NRC, OIE	Team Leader (Asst)		X	X	X	X	X	X
L. Stanley	Zytor, Inc.	Team Member					X	X	
C. Crane	Westec Services	Team Member		X	X	X	X	X	
I. Ahmed	NRC, NRR	Team Member		X	X	X	X	X	
R. Lipinski	NRC, NRR	Team Member		X	X	X	X	X	X
G. Harstead	Harstead Engineering	Team Member		X	X	X	X	X	X
A. duBouchet	Harstead Engineering	Team Member		X	X	X	X	X	
A. Unsal	Harstead Engineering	Team Member		X	X	X	X	X	X
G. Overbeck	Westec Services	Team Member		X	X	X	X	X	
J. Nevshemal	Westec Services	Team Member		X	X	X	X	X	
W. Mills	Nuclear Energy Cslts.	Team Member		X	X	X	X	X	
W. Anderson	NRC, OIE	Team Member		X	X	X	X	X	
R. Helmick	GSU	Project Engineer		X				X	
G. Ankrum	NRC, OIE	QA Branch Chief		X	X			X	
J. Milhoan	NRC, OIE	Section Chief	X	X	X	X	X	X	X
W. Clifford	SWEC	Res. Proj. Mgr.		X				X	
F. Canuso	SWEC	Project Engr.-EA		X	X		X	X	
J. Kirkebo	SWEC	Sr. Proj. Engr.	X	X	X		X	X	
R. Normandean	SWEC	Sr. Mech. Engr.		X					
J. Booker	GSU	Mgr. Engr., NF&L	X	X				X	
J. Deddens	GSU	V.P., RB Group	X	X					
T. Crouse	GSU	Manager, QA	X	X	X			X	
D. Chamberlain	NRC, RIV	Sr. Res. Insp.	X	X	X	X		X	
E. Weinkam	NRC, NRR	Licensing Proj. Mgr.		X	X	X		X	X
J. Jaudon	NRC/RIV	Proj. Sec. Chief	X	X	X				
T. Chitester	S&W	Sr. Mech. Sys. Engr.		X				X	
W. Tucker	SWEC	Asst. to Supt. of Engr.		X	X			X	
L. England	GSU	Super. - Nuclear Lic.		X	X	X	X	X	
D. Lorfing	GSU	Nuc. Engr.-NuP.E.		X	X	X			
P. Dautel	GSU	Onsite Licensing Rep.		X		X			
R. Stout	SWEC	Sr. Ctrl. Sys. Engr.		X				X	
R. Johnson	SWEC	Sr. Instr. Engr.		X				X	
T. Shea	SWEC	Sr. Electrical Engr.		X					
C. Lambert	GSU	Supr.-Proj. Engr.		X	X		X		
T. Vears	S&W	Structural		X					
D. Barry	SWEC	Supt. of Engr.		X	X			X	
R. Burell	SWEC	Sr. Mech. Engr.		X					
J. Glazar	GSU	Director, NPE			X			X	
E. Zoch	GSU	Supr.-Nuc. Engr.			X	X			
E. Grant	GSU	Licensing Supr.			X				
W. Reed	GSU	Licensing			X			X	X
F. Reuter	GE	Sr. Proj. Mgr.			X	X			
R. Ciccarelli	GE	Sr. Proj. Mgr.			X	X		X	
L. Bohl	GE	Manager, QA & RO			X	X			
J. Bisti	SWEC	Proj. Engr. - Fld. Sppt.				X	X		

Table 7.1 (Continued)

Name	Organization	Title	Meeting Attended 1	2	3	4	5	6	7
R. Byrnes	SWEC	Engrng. Manager					X		
W. Culp	SWEC	Proj. Engr.					X		
W. Drotleff	SWEC	VP - Mgr. of Projects					X	X	
N. Motiwala	SWEC	Lead Eng'g Mech. Engr.					X	X	
W. Raughley	SWEC	Asst. Proj. Engr.					X		
T. Szabo	SWEC	Lead Nuc. Tech. Engr.					X		
J. Weller	SWEC	Lead Elec./Controls Engr.					X		
C. Fonsizca	SWEC	Asst. Mgr. - Eng'g Mech.					X		
T. Ott	SWEC	Asst. Mgr. - Electrical					X		
R. Jadeja	SWEC	Lead Struct. Engr.					X	X	X
S. Datta	SWEC	Cont. Sys. Div. - Mgr.					X		
P. Guha	SWEC	Lead Control Engr.					X		
T. Coughlin	SWEC	Mgr. - Struct. Div.					X		
R. Lykens	SWEC	Proj. QA Administration					X		
W. Curtis	SWEC	Supr. (E.A. Div.)					X		
J. Lord	SWEC	Manager, E.A.	X				X	X	
R. McMorland	SWEC	Lead Power Engr.					X	X	
M. Boothby	SWEC	Asst. Mgr. - Power Div.					X		
T. Vaughn	SWEC	Proj. QA Supervisor					X		
R. Berry	SWEC	Asst. Proj. Engr.					X		
D. Norkin	NRC, OIE	Inspection Specialist	X						
R. Farrell	NRC, RIV	Sr. Resident Inspector	X						
P. Check	NRC, RIV	Deputy Administrator	X					X	
H. Powell	GE	NSS Proj. Mgr.	X					X	
D. Denise	NRC, RIV	Project Div. Dir.	X						
E. Johnson	NRC, RIV	Branch Chief	X						
N. Grace	NRC, OIE	Division Dir. - QASIP						X	
R. Stafford	GSU	Dir.-Qlty. Services						X	
M. Walton	GSU	Tec. Asst., Proj. Engr.						X	
W. Eifert	SWEC	Chief Engr. - EA Div.						X	
D. Wilson	SWEC	Const. Engr. - Elec.						X	
P. Graham	GSU	Asst. Plant Mgr.						X	
S. Radebaugh	GSU	Preop Test Supv. - NSSS						X	
J. Hamilton	GSU	Supervisor Site Engrg.						X	
G. McGee	GE	Mgr. - River Bend Engrg.						X	
L. Test	GE	Consulting Engr.						X	
R. Spence	SWEC	Supt. - Field QC						X	
C. Ballard	GSU	Supervisor Qual. Engr.						X	
J. Zullo	SWEC	QA Engr.						X	
H. Wang	NRC, OIE	Inspection Specialist							X
H. Polk	NRC, NRR	Struc. Eng.							X
A. Copoulos	BNL-NRC Cslnt.	Struc. Consultant							X
D. Fiorello	SWEC	Struc. Engr.							X
N. Amin	SWEC	Struc. Engr.							X
M. Holley, Jr.	SWEC-Cslnt.	Struc. Consultant							X
R. Shewmaker	NRC, OIE	Sr. Struc. Engr.							X
G. Lear	NRC, NRR	Branch Chief, SGEB							X

7.2 Persons Contacted - Mechanical Systems

Name	Title	Organization
D. P. Barry	Superintendent of Site Engineering	Stone & Webster
W. T. Tucker	Assistant to Superintendent of Site Engineering	Stone & Webster
F. A. Canuso	Project Engineering Assurance Engineer	Stone & Webster
R. Ciccarelli	Senior Project Engineer	General Electric
R. J. McMorland	Lead Power Engineer	Stone & Webster
T. S. Szabo	Principal Engineer	Stone & Webster
R. S. K. Smith	Engineer	Stone & Webster
P. S. Wiesendanger	Engineer	Stone & Webster
M. L. Boothby	Assistant Manager Power Division	Stone & Webster
G. E. Hirst	Engineer	Stone & Webster
N. S. Motiwala	Lead Engineering Mechanics Division Engineer	Stone & Webster
L. Dietrich	Lead Licensing Engineer	Stone & Webster
J. Zaccaria	Designer	Stone & Webster
S. M. Feldman	Engineering Mechanics Division Mechanical Section Manager	Stone & Webster
W. Wang	Engineering Mechanics Division Mechanical Section Assistant Mgr.	Stone & Webster
N. Ly	Engineer	Stone & Webster
W. G. Culp	Project Engineer Licensing and Equipment Qualification	Stone & Webster
R. Normandeau	Senior Engineer Mechanical	Stone & Webster
A. A. DiSabatino	Assistant Manager Power Division	Stone & Webster
R. Weaver	Engineering Assurance Engineer	Stone & Webster
L. A. England	Supervisor-Nuclear Engineering	Gulf States Utilities
M. J. Shah	Engineer Structural Division	Stone & Webster
J. A. Kirkebo	Senior Project Engineer	Stone & Webster
V. Josyula	Principal Engineer ESSA Group	Stone & Webster
C. A. Fonseca	Engineering Mechanics Division Assistant Division Manager	Stone & Webster
K. S. Jadeja	Lead Structural Engineer	Stone & Webster
W. R. Curtis	Supervisor Engineering Assurance Division (Boston)	Stone & Webster
T. Bates	Assistant Superintendent of Engineering	Stone & Webster
R. A. Berry	Assistant Project Engineer	Stone & Webster
L. Ferringo	Engineer Building Services	Stone & Webster
C. Chin	Engineering Mechanics Division Mechanical (Pipe Rupture) Engineer	Stone & Webster
B. Salter	Engineer	Stone & Webster
R. Buell	Senior Mechanical Engineer Site Engineering Group	Stone & Webster
R. Howard	Principal Engineer	General Electric
B. Powell	Supervisor Project Manager	General Electric

7.2 continued.

Name	Title	Organization
S. Radebaugh	Startup Testing	General Electric
E. Ringel	Power Division	Stone & Webster
O. Foster	Principal Engineer	General Electric
F. Reuter	Manager, PSAE	General Electric
G. Romanek	Senior Engineer Plant Equipment Design	General Electric
F. Paradiso	Plant Safety Performance	General Electric
P. Freehill	Supervisor of Startup Testing	Gulf States Utilities
E. Zoch	Supervisor of Nuclear Engineering	Gulf States Utilities
R. Stout	Senior Electrical Engineer	Stone & Webster
P. Graham	Technical Staff Supervisor	Gulf States Utilities
C. Bogolin	Operations Supervisor	Gulf States Utilities
J. Booker	Manager	Gulf States Utilities
J. Glazer	Director Projects	Gulf States Utilities
W. Reid	Director Licensing	Gulf States Utilities

7.3 Persons Contacted - Mechanical Components

Name	Title	Organization
R. E. Foley	Asst. Chief Engr. Stress Reconciliation	S&W
R. E. Lintelman	(Back-up) Sr. EM Engr. SEG/EMD	S&W
R. S. Chattha	Principal Pipe Stress SEG/EMD	S&W
R. Belanger	(Back-up) Sr. EM Engr. SEG/EMD	S&W
D. M. Cowart	Chief Inspection Supervisor FQC	S&W
P. A. Miktus	Sr. Task Leader (L. B. Pipe Supports) SEG/EMD	S&W
T. Flynn	Sr. Task Leader (S. B. Pipe Supports) SEG/EMD	S&W
W. Hess	Assistant to Sr. EM Engr. SEG/EMD	S&W
R. D. Normandeau	Sr. EM Engr. SEG/EMD	S&W
C. Fonseca	Assistant Manager EMD	S&W
N. Motiwala	Lead Engineering Mechanics Engineer	S&W
R. A. Berry	Assistant Project Engineer	S&W
R. McMoreland	Lead Power Engineer	S&W
T. Szabo	Lead Nuclear Technology Engineer	S&W
R. F. Bates	Section Leader Mech. Sys. Squad	S&W
A. Chan	Manager EMD	S&W
A. C. Leonard	Principal Pipe Support Engineer	S&W
R. W. Ferguson	Sr. FQC Engineer	S&W
J. A. Green	FQC Inspection Supervisor	S&W
G. C. Pentek	Chief Const. Engr.	S&W
J. J. Gingera	As Built Coordinator EMD	S&W
T. C. Mitchell	Asst. Supt. of Eng.	S&W
D. C. Hodges	Supervisor - Quality Systems	GSU
L. C. Ballard	Supervisor - Quality Engineering	GSU
T. Y. Chang	EMD	S&W
E. Ingels	EMD	S&W
Y. Wu	Principal Pipe Stress Engineer EMD	S&W
I. Vishnevetsky	PSD	S&W
D. Bray	PSD	S&W
T. Y. Chow	Section Leader Pipe Stress EMD	S&W
K. Basu	Principal Pipe Stress Engineer EMD	S&W
T. L. Wang	EMD	S&W
R. Tate	Principal Engineer, Piping	S&W
A. Blum	Equipment Qualification Coordinator	S&W
N. Muni	Lead Equipment Qualification Engineer EMD	S&W
B. MacKellar	Project Engineer	RCI
J. Maheshwari	Engineer	RCI
R. Ciccarelli	Sr. Project Manager	GE
D. Bhargava	Equipment Qualification	S&W
S. Feldman	Section Leader, Equipment Qual.	S&W
C. Lambert	Supvr. Project Engineer	GSU

7.4 Persons Contacted - Civil and Structural

Name	Title	Organization
K. Jadeja	Lead Structural Engineer	Stone & Webster
L. Shiau	Group Leader/Structural	Stone & Webster
S. Leary	Group Leader/Structural	Stone & Webster
N. Amin	Principal Engineer/Structural	Stone & Webster
P. Bertucci	Group Leader/Structural	Stone & Webster
M.J. Shah	Assistant Division Manager	Stone & Webster
B. Ebbeson	Principal Engineer	Stone & Webster
E. Baum	Structural Engineer	Stone & Webster
M.A. Bilowich	Senior Construction Engineer	Stone & Webster
T. Vears	Senior Construction Engineer	Stone & Webster
R. Belanger	Senior Eng. Mechanics Engineer	Stone & Webster
N. Motiwala	Lead EM Engineer	Stone & Webster
R. McMoreland	Lead Power Engineer	Stone & Webster
S. Chin	Principal Engineer/EM	Stone & Webster
A. Ash	Squad Leader/Structural	Stone & Webster
D. Fiorello	Group Leader/Structural	Stone & Webster
T. Coughlin	Structural Division Manager	Stone & Webster
M. Stein	Structural Engineer	Stone & Webster
P. Conti	Lead Geotechnical Engineer	Stone & Webster
M. Masucci	Geotechnical Engineer	Stone & Webster

7.5 Persons Contacted - Electric Power

Name	Organization	Title
R. W. Helmick	GSU RBS	Project Engineer, Site Project Engineering
J. M. Glazar	GSU	Director, Nuclear Plant Engineering Beaumont
L. Schell	GSU	Electrical Engineer, Beaumont
R. G. West	GSU RBS	Electrical Engineer, Site Engineering
P. G. McGill	GSU RBS	Senior Electrical Engineer, Site Engineering
B. Kilgore	GSU RBS	Electrical Engineer, Site Area Coordinator
D. Frayer	GSU RBS	Start Up and Test Engineer
C. Lambert	GSU, Cherry Hill	Project Engineer
D. Ciccarelli	GE, San Jose	Senior Project Engineer
C. E. McGee	GE RBS	Manager, Site Engineering, PGCC
W. J. Dallas	GE RBS	Electrical Engineer, PGCC
L. K. Kelly	GE RBS	Engineering Support, PGCC
N. Luria	GE, Valley Forge	Manager, Equipment Qualification
R. Walman	GE, San Jose	Engineer
L. Test	GE, San Jose	Manager
R. Frieze	GE, San Jose	Engineer
P. Holland	GE, San Jose	Systems Engineer
T. Sigman	GE RBS	Site QC Engineer
W. Smith	GE RBS	Manager, Site Installation
J. Lord	SWEC CHOC	Manager, Engineering Assurance
P. Guha	SWEC CHOC	Lead Controls Engineer
J. Weller	SWEC CHOC	Lead Electrical Engineer
G. Doddy	SWEC CHOC	Control System Engineer
W. Raughley	SWEC CHOC	Asst. Project Manager
J. Gelston	SWEC CHOC	Electrical Engineer
W. Liverant	SWEC CHOC	Electrical Engineer
A. Gross	SWEC CHOC	Squad Leader - Wiring Design
J. Balzano	SWEC CHOC	Squad Leader - Physical Design
R. Tate	SWEC CHOC	Piping Engineer
A. Blum	SWEC CHOC	Engineer, Equipment Qualification
W. G. Culp	SWEC CHOC	Project Engineer, Equipment Qualification
T. Shea	SWEC RBS	Senior Electrical Engineer, Site Engineering Group
M. Duke	SWEC RBS	Senior Electrical, Field Support Engineer
T. Nicolan	SWEC CHOC	Controls Engineer
L. Rajkowski	SWEC CHOC	Electrical Engineer
J. Cicalese	SWEC CHOC	Squad Leader, Controls Design
N. Borreggine	SWEC CHOC	Electrical Engineer
R. Seger	SWEC RBS	Preliminary Test Engineer (PTO)
J. James	SWEC RBS	Section Manager, Equipment Qualification
K. Smith	SWEC CHOC	System Engineer
B. Curtiss	SWEC Boston	Manager
A. Pezzuto	SWEC CHOC	Designer
V. Deo	SWEC CHOC	Engineer, Equipment Qualification

7.5 continued.

Name	Organization	Title
A. Giancatarino	SWEC CHOC	Engineer, Equipment Qualification
W. G. Drummond	SWEC CHOC	Electrical Engineer
A. Lew	SWEC CHOC	Electrical Engineer
B. Gupta	SWEC CHOC	Controls Engineer
A. Guarriero	SWEC CHOC	Electrical Engineer
G. Fan	SWEC CHOC	Electrical Engineer
A. Wadiva	SWEC CHOC	Power Engineer
A. Khanna	SWEC CHOC	Div. Elect. Sup.
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R. Fay	SWEC RBS	QC Elect. Sup.

7.6 Persons Contacted - Instrumentation & Control

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Philip Graham	Asst. Plt. Mgr., Administ.	Gulf States Util. (RBS)
James Glazar	Director, Nucl. Plt. Eng'g	Gulf States Util. (HQ)
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Philip Porter	Sr. Electrical Engineer	Gulf States Util. (HQ)
Charles Bogolin	Operations Supervisor	Gulf States Util. (RBS)
Donald Jeroigan	Startup & Test Engineer	Gulf States Util. (RBS)
Robert Evans	Startup & Test Engineer	GE Field Supp./GSU (RBS)
James Pawlik	Startup & Test Engineer	GE Field Supp./GSU (RBS)
John Hamilton	Supervisor, Site Eng'g	Gulf States Util. (RBS)
Philip McGill	Sr. Elec/Engr., Site Eng.	Gulf States Util. (RBS)
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William Curtis	Engineering Assurance	Stone & Webster (HQ)
William Tucker	Engineering Assurance	Stone & Webster (RBS)
Thomas M. Bates,Jr.	Asst. Superint. Eng'g.	Stone & Webster (RBS)
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7.6 continued.

Name	Title	Organization
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P.J. Kinder	C&I Project Engineer	GE River Bend Project
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Attachment A-1 Preparation, Check and Review of Design Calculations

OBJECTIVE

During the inspection each team member reviewed the adequacy of the checking and review of design calculations in their discipline. This aspect of the design process is described in the Stone & Webster procedure EAP-5.3, "Preparation and Control of Manual and Computerized Calculations (Nuclear Projects)", Rev. 3, Attachment 3.0. This procedure also sets forth how calculations are to be prepared (format) and the information that must be contained within them. Basic to the preparation of any calculation is the timely checking and review of it after completion. The type of review (alternate calculation, comparison to a previously approved similar calculation or number-by-number check) depends on the type of calculation. If a calculation is manual and has a high degree of system specificity, there is the need for a number-by-number check. EAP-5.3, Attachment 3.0 recognizes this fact in the stated requirements for the review of results.

As the inspection progressed, internal discussion amongst the team members revealed a common concern pertaining to checking and review of design calculations. The basis for the concern was the type and number of errors in the calculations being inspected that should have been corrected if an effective check and review had taken place. The nature of these errors were; a) use of wrong methods, b) stating one reference while using another, c) arithmetic, d) page to page transfer of data, and e) page references. Another concern expressed by certain members of the team was the long time between completion of a calculation and when it was checked and reviewed. A situation surfaced where a calculation was not checked, but the design had progressed to the point where installation had taken place. Also, there were concerns with inconsistent sign-offs which involved different individuals signing for the efforts of others and multiple signatures without any indication as to who checked or reviewed what portion of the calculation. The method of review was also not being identified as required by ANSI Quality Assurance standards. All of these concerns culminated in a decision to have as one of the objectives of the inspection a coordinated inter-discipline evaluation of the adequacy of the Stone & Webster preparation, checking and review of design calculations.

EVALUATION

In order to assess this concern in mechanical systems, four calculations were selected for checking and review in a manner consistent with the common practice of the industry. The checking exercise revealed errors in simple addition, in page to page transfer of data and in page references (see Deficiency D.A.1-2). The review exercise encountered inconsistent application of the requirements of Stone & Webster procedure EAP 5.3. This procedure requires that a preparer of a calculation indicate which inputs, assumptions or references need "later confirmation". The reviewer of a calculation is required to assure that this has been done properly. The review revealed instances of items needing later confirmation either being marked "no confirmation required" or of nothing being indicated (see Observation O.A.1-3).

The inspection of other mechanical systems calculations produced additional concerns that should have been corrected during checking and review. There is an instance of an incorrect method being used which directly resulted in an inappropriate design decision (see Deficiency D2.3-1). The other concerns were; a) the use of an assumption that did not correspond to the design basis supplied by General Electric (see Deficiency D2.4-1), b) calculational results that do not conform to the equipment purchased and installed (see Deficiency D2.3-5 and Deficiency D2.4-2) and use of system operational conditions that did not correspond to those used as the basis for the FSAR safety analysis (see Deficiency D2.4-6).

Inspection of the stress analysis calculations uncovered instances where assumed values were used in the calculation without being listed either as a reference, assumption or input. Also, preliminary data was used in the calculations without being identified for "later confirmation required." There was an error found in the distance between two lumped masses that should have been corrected if a thorough check and review had been conducted. The later finding could have a significant affect on the result of that analysis. All these findings should have been identified and corrected during the process of checking and reviewing of the calculations (see Deficiency D3.4-2 and Deficiency D3.3-2).

The Stone & Webster Structural Division has supplemented EAP-5.3 with their procedure STP-11.5. Violations of STP-11.5 in the area of checking and review were also found. There is the instance of an alternate method calculation being performed which produced results that were different from the original calculation. The governing procedure (STP-11.5.2) requires the alternate calculation to be checked if there is a difference in the results. The alternate calculation was not checked. It was found that both the original calculation and the alternate contained errors (see Deficiency D4.15-1). EAP-5.3 requires only the cover sheet to be signed whereas, STP-11.5 requires each page of a calculation be initialed when checked. Violations of this aspect of the STP procedure were encountered in the civil calculations (see Deficiency D4.6-2). It was also found that extended periods of time existed between the completion of a calculation and when it was checked and reviewed (see Observation O4.4-1).

Problems were also found with the electrical calculations that should have been corrected by checking and review. There is the case of a reference being listed as the source of data when another reference with different data was used. Also, in the same calculation an assumption was used but the reference from which the information was taken was not listed (see Deficiency D5.4-1). Another finding also involved the use of key assumptions without their source being referenced (see Deficiency D5.11-1).

Errors were encountered in the instrumentation and control calculations which also gave rise to a concern about the checking and review process. Certain basic calculational inputs that directly affect the results were not included in the method used (see Deficiency D6.6-2). This can be considered as an error in the calculational inputs as well as incomplete assumptions and inputs. The checking and review of these calculations should have uncovered the shortcomings.

Stone & Webster has committed to follow ANSI N45.2.11 as the quality assurance standard for the design process. This standard requires the review process to be auditable. EAP-5.3 states that signing the calculation cover sheet as reviewed is sufficient to indicate that the activity has been performed. There is no requirement in the procedure to identify which review method was used. The mechanical calculations which have been prepared in accordance with EAP-5.3 did not conform to the ANSI N45.2.11 commitments (see Deficiency D.A.1-1). The Structural Division recognized the requirement of ANSI N45.2.11 in their procedure STP-11.5; calculations inspected from this division conformed with the requirement.

CONCLUSION

The inspection team found cases where the calculation check and review process was inconsistent and was not properly implemented. Further the inspection team found the problems with the process to be widespread. Therefore, the conclusion is that the problem is systematic in nature. It is also the opinion of the team that EAP-5.3 is a root cause because it fosters the concept of just referring to the need for a review/check without thorough documentation that the activity has been properly performed. This is a weakness in the Stone & Webster design process and should be strengthened. The team also observed many examples of high quality documentation and calculations. The errors noted did not lead to significant design problems, thus mitigating the team's concerns in this area.

RELATED FORMS

D.A.1-1 (Deficiency) Compliance of Calculations with the Requirement for an Auditable Review/Check of Results

D.A.1-2 (Deficiency) Arithmetic and Reference Errors in Calculations

O.A.1-3 (Observation) Compliance of Calculations with Established Stone & Webster Procedure Preparation

Other forms referred to above are discussed in their respective report sections.

Attachment A-2 Design Verification Process

OBJECTIVE

The objective of this portion of the inspection was to evaluate the adequacy of the design verification process.

EVALUATION

Stone & Webster's design verification process was reviewed to gain an understanding and an appreciation for how the process is intended to work. The following is a synopsis of that process to assist in understanding the team's concerns in this area.

Engineering Assurance Procedure (EAP) 3.1, "Verification of Nuclear Power Plant Designs," establishes the requirements for verification of Stone & Webster nuclear power plant designs. The stated purpose of EAP 3.1 is "to verify the adequacy of design by substantiating that the design inputs have been correctly selected, and that the design meets the specified inputs." This is accomplished by independent objective reviews of key design documents. Stone & Webster has defined key design documents as those documents that establish design criteria, describe the design approach or otherwise define the design to the detail necessary to allow preparation of the final design output documents. Detail A.A.2-3 identifies by document type the key design document, the individual groups responsible for independent objective review, and the method of documenting approval.

EAP 3.1 also describes the general sequence for verification of the power plant design. It is initiated by independent objective review of the key design documents that first identify the design requirements that apply to the project and the design approach developed to satisfy the design requirements. For Stone & Webster's nuclear power plant projects, the first key design documents are normally system descriptions. However, EAP 3.1 recognizes that because of project schedules a PSAR may be prepared before the issuance of project system descriptions and consequently requires the independent objective review of the PSAR as the first step in the verification process. As described in EAP 3.1, the PSAR is a key design document only when it is the first documentation of the design inputs and remains a key design document until subsequent documents are issued to record the design information. These succeeding lower level key design documents are conceptual drawings (e.g. site plan, plot plan, and general arrangements), structural support design specifications, structural design criteria, electrical design criteria, specifications, calculations, flow diagrams, one line electrical diagrams, and logic diagrams. EAP 3.1 requires that succeeding lower level key design documents be subjected to independent objective review to assure that requirements established by the previously verified key documents have been met and that design information added meets the intended design requirements. To maintain the verification current, all subsequent revisions to key design documents are subject to independent objective review.

The design verification process on the River Bend Project differs from the standard Stone & Webster nuclear project in that there are no system descriptions. Early in the project, Gulf States Utilities made a decision not to prepare system descriptions; although this was a standard Stone & Webster practice. Consequently a system description, which identifies and defines the system design requirements, does not exist for project and off-project personnel to refer to when performing independent reviews of key documents. For River Bend, overall knowledge of system requirements must be gleaned from individual lower level key design documents. Other sources of overall knowledge of system requirements are the Final Safety Analysis Report and General Electric interface documents. However, these documents are not key design documents as defined by EAP 3.1.

The team has reviewed numerous documents defined by Stone & Webster as key design documents. In every instance the team found that the independent objective reviewer had documented a satisfactory completion of the verification by signature or initials. However, the team was unable to assess the quality of those reviews, because there is no requirement to document the reviewer's comments and the resolution of those comments.

During the inspection the team was informed that Division Managers periodically have technical reviews performed to ensure a consistency of quality and technical adequacy on the various engineering projects at the Cherry Hill Office Center (see Detail A.A.2-1). The team was instructed that these reviews were not part of the design verification process. The team was also informed that the Engineering Assurance Division had conducted system level design audits (see Detail A.A.2-2). These audits were carried out to verify compliance with design quality assurance programs.

The team identified deficiencies which ranged from minor documentation errors to design oversights. In the final analysis, these deficiencies should have been found by the checks and reviews prescribed by Stone & Webster's design process. In a practical sense the team recognizes that some errors will slip through; however, the design process should be tight enough to preclude all errors except insignificant items. The team examined the deficiencies found on the River Bend Project to assess which of those deficiencies should have been identified by Stone & Webster's design verification process. The following are examples of deficient design conditions that should not have gone undetected by Stone & Webster's design verification process and that had the potential for causing significant design inadequacies.

Deficiency D2.3-1 described calculation PN-268 as generally conservative for determining the low pressure coolant injection pump head requirements and non-conservative with respect to limiting runout flow. If this deficiency had gone undetected, including the preoperational testing program, the excessive flow could result in a net positive suction head required greater than that available and cause pump cavitation and damage.

Deficiency D2.3-2 identified problems with calculation PN-283 in that the assumptions used to demonstrate sufficient suppression pool level following leakage into the largest emergency core cooling system equipment room was not selected to maximize loss of suppression pool water. In addition, the calculation did not address the consequences of leakage from the first isolation valve outside the suppression pool as required by the objective of the calculation. If this deficiency had gone undetected, the potential existed for significant degradation of the emergency core cooling system had a leak occurred during a loss of coolant accident.

Deficiency D2.3-3 identified that a quantitative analysis/evaluation had not been performed to determine the effect on the ability to maintain core cooling and safe shutdown assuming a passive piping failure in one of the emergency core cooling system suction lines post-LOCA. If this deficiency had gone undetected, the potential existed for significant degradation of the emergency core cooling system had a passive piping failure occurred during long term post-LOCA cooling.

Deficiency D3.3-3 identifies that the nozzle loads on the inlet flange of the safety relief valves might be exceeded because General Electric nozzle load limits were missed and not considered in the pipe stress calculation. If this deficiency had gone undetected, the safety relief valve inlet flanges could have been subject to overstress conditions and could have resulted in a LOCA.

Detail A5.3-1 identified that the motors for motor-operated low pressure coolant injection valves located inside the Reactor Building are not environmentally qualified and cannot be relied on to accomplish their design basis safety function. The motor actuators were procured, released, technically reviewed, and installed with this error undetected until October 13, 1983. Although Stone & Webster's Equipment Qualification Group discovered this error the procurement specification had been reviewed and design verified several times since 1974 (e.g., 9 addendums and 1 revision) without detecting the error. Had Stone & Webster's Equipment Qualification Group not discovered the error, two low pressure coolant injection lines may have been subject to common mode failure following a LOCA.

Deficiency D5.4-1 identified that the engineer used the incorrect time current trip curves in a cable sizing calculation. This error, considered to be obvious, was not detected by two independent design reviews.

Deficiency D6.5-2 identifies that interruptible, Class 1E AC power sources were chosen for the Emergency Core Cooling System output relay contact multiplication circuits used to actuate emergency power sources and loads. If this deficiency had gone undetected, then whenever off-site AC power is lost, the LOCA output relays would be rendered inoperable until AC power is restored to the 4KV busses.

As stated previously, these are examples of deficiencies that should not have gone undetected and that had the potential for causing significant design inadequacies.

The team also found numerous deficiencies that appear to reflect on the completeness of Stone & Webster's technical understanding of the General Electric BWR interface with various Balance-of-Plant systems. Of the five technical disciplines represented on the Integrated Design Inspection team, the Mechanical Systems, Mechanical Components, Electrical, and Instrumentation and Control teams each identified one or more deficiencies involving a General Electric interface requirement. The following presents a summary of these individual deficiencies to illustrate the team's concern with respect to external interface control.

In the Mechanical Systems area, the design output rating of the pneumatic supply for the automatic depressurization system valves, permissible leakage characteristics for the automatic depressurization system accumulator check valves, and the elimination of a resistance orifice to prevent residual heat removal system pump runout have been identified as deficiencies involving this technical interface. For example, General Electric specified that a 150 psig minimum pneumatic supply be available for charging the automatic depressurization system valve accumulators; Stone & Webster provided a safety grade supply rated at 105 psig for this purpose with non safety booster compressors to raise the pressure to the specified value. Stone & Webster did not specify a maximum leakage characteristic for the automatic depressurization system accumulator check valves even though General Electric criteria states a "bubble tight" leakage requirement, erroneously believing that their safety related upgrade fully satisfied the interface requirements. A resistance orifice is suggested by General Electric to preclude residual heat removal system pump runout; however, Stone & Webster eliminated this orifice from the design based on a non-conservative calculation. In each of these instances, technical coordination of these differences remains to be accomplished between General Electric and Stone & Webster.

In the Mechanical Components area, stress analysis of safety/relief valve nozzle loads, review of safety/relief valve piping ball joint test reports, and selection of equipment support stiffness values have been identified. For example, General Electric specified permissible inlet and outlet nozzle loads for the safety relief valves, and the Stone and Webster Power group alerted the stress analysis group regarding these requirements; however, these nozzle loads were not considered in the Stone & Webster stress analysis calculations. Stone & Webster's review of ball joint qualification test reports did not question the adequacy of neutron radiation effects on ball joint lubricants, the cyclic motion tests, and the shock tests. In addition, General Electric specifies stiffness design criteria applicable to the four residual heat removal system heat exchangers supported by structural steel. In this instance, Stone & Webster did not independently validate the applicability of the General Electric stiffness criteria, and did not perform an independent verification of the dynamic aspects of the residual heat removal system heat exchanger component support configuration. Decoupling criteria stated in the FSAR were not used in the design calculations for these components.

In the Electric Power area, cable and conduit installation and sealing recommendations provided by General Electric for the junction boxes housing these terminal blocks have not been rigorously followed by Stone & Webster.

In the Instrumentation and Control area, the team identified lack of actuation of essential equipment based on an accident initiation signal (i.e., LOCA signal to start the Standby Service Water System). Use of interruptible AC power sources for starting of the emergency diesel generators and other essential loads post LOCA is inconsistent with both General Electric and industry design practices. Use of only manual control for four standby service water system motor-operated valves used for cooling of the high pressure core system diesel generator is inconsistent with General Electric design practices regarding automatic valve alignment at the onset of a postulated accident.

The collective effect of these individual and varied deficiencies suggest that Stone and Webster's design process has not been effective in addressing General Electric interface requirements. These deficiencies can be categorized as external interface control deficiencies and are representative of a weakness in the design verification process. Specifically, EAP 3.1 requires that an independent reviewer answer the question, "Have the design interface requirements been satisfied?"

The River Bend design verification process differs from observed industry practive in two areas. The first area is system design descriptions. It is general practice to use system design descriptions to identify all pertinent design requirements and to be the primary design document. In many Architect-Engineer companies system descriptions are maintained throughout the design and this seems to be Stone and Webster's corporate policy. In some instances the system design description is only maintained until completion of major design evolutions. After that subsequent design documents such as piping and instrumentation diagrams, logic diagrams, elementary drawings, etc. are considered sufficiently complete to reflect the system design without keeping the system descriptions current. System design descriptions are relied upon to focus the design requirements.

Another area of difference is the lack of an integrated design review. ANSI N45.2.11 states that "the extent of the design verification required is a function of the importance to safety of the item under consideration, the complexity of the design, the degree of standardization, the state of the art, and the similarity to previously proven designs." The design of safety-related systems are in many cases complicated by various design and interface requirements. How well a system fulfills its safety function is not readily judged by review of individual parts of the design. During the inspection the team could not confirm from Stone & Webster personnel that integrated design reviews of the low pressure coolant injection and automatic depressurization systems had been performed. Following completion of the inspection Stone & Webster addressed the issue of system level design reviews in a synopsis to the team. We were informed of Stone & Webster's system engineering approach coupled with an integrated design control process to ensure system level design review. The synopsis stressed the conceptual engineering phase, use of a system's engineer approach and independent review process in the production engineering phase, Nuclear Steam Supply System interface controls (including the design freeze process), the overview of the design process by Stone & Webster Engineering Assurance and Gulf States Utilities Quality Assurance organizations, and a short description of additional Gulf States involvement in system level reviews. Stone & Webster also addressed two special design review programs performed for the River Bend Project. One of these was a corporate review board composed of experienced off-project technical personnel chartered to conduct a thorough evaluation of several systems, plant features and equipment. This review was performed in 1976, a time period which predates most of the production engineering work which raised the team's concerns. An additional design review was sponsored by Stone & Webster management in late 1981/early 1982. This review was stated to be conducted by experienced off-project technical personnel and focused on the technical adequacy of several sample systems. The team did not review the latter special design reviews.

Notwithstanding an evaluation of these reviews, the findings detailed above were still observed by the team and raise a question concerning the design verification process used for River Bend. The team did not perceive that the design was grossly flawed, nor could it conclude that system level design reviews of functional requirements should be conducted for all safety related systems. In spite of this information and based upon the type of deficiencies found, the team believes that the lack of an integrated design verification program is a significant weakness on the River Bend Project. It has been general practice within architect-engineer companies to conduct a system level design review of various design documents to insure that the system can fulfill its safety function. This system level design review is in addition to the normal production reviews. This system level design review is normally performed off project at a Chief Discipline Engineer or Engineering Department level and is part of the design verification process. It has also been general practice that if significant changes are made to the system then the system is reviewed again or, as a minimum, the affected portion of the system is reviewed. For example, the River Bend automatic depressurization system design change from 150 psig pneumatic supply to a lesser value such as 110 psig would warrant another design review.

SUMMARY/CONCLUSION

Considering the type and nature of the design deficiencies found, it is the team's conclusion that the River Bend design verification process should be improved. The River Bend design verification process relies on the independent review of unique key documents such as a flow sketch, a calculation, or a specification. The River Bend design verification process is not structured to facilitate an integrated review of key documents. Although EAP 3.1 states that succeeding lower level key design documents are subjected to independent objective review to assure that the requirements established by the previously verified key documents have been met, the team has observed significant deficiencies which indicate that the intent of EAP 3.1 is not being met. Likewise EAP 3.1 requires that independent reviews verify that design interface requirements have been satisfied; however, the team has observed numerous deficiencies which also indicate that the intent of EAP 3.1 is not being met. The lack of system design descriptions aggrevates this situation, because independent reviewers do not have a reference to use to which defines the system design or the interfacing requirements.

RELATED FORMS:
A.A.2-1 (Detail) Division Manager Technical Reviews
A.A.2-2 (Detail) Engineering Assurance Division Audits
A.A.2-3 (Detail) Independent Objective Review and Dccumentation

A2.3-1 (Detail) Design Process and Equipment Purchase

DESCRIPTION

The procedure used by Stone & Webster for purchase equipment was described to the inspection team as follows: (For this discussion it is assumed that a pump is being purchased).

Step (1) - A master specification is prepared by the Boston-based pump specialist. This specification is supplied to the project.

Step (2) - The project responsible system engineer fills in the calculations or other acceptable design basis information.

Step (3) - The specification (which at this point in sequence of events is the bid specification) is sent out to approved vendors.

Step (4) - Bids are returned. Both the project responsible system engineer and pump specialist review them for compliance with the specification. A recommendation is developed and submitted to the client (in this case Gulf States Utilities) for approval.

Step (5) - When the client approves the recommendation, the specification is sent to the successful bidder as Addendum No. 1. This addendum identifies the specification as a Purchase Specification and supercedes all previous versions or revisions of the specification. After this point the methods available to change the specification are by Addenda, Revision or Engineering and Design Change Request. Addendum No.1 of specification contains a section which states what certified information must be supplied by the vendor.

Step (6) - As the certified vendor information is received by the project it is made part of either the vendor drawing files (if the information contained physical dimensions) or technical manuals (if the information is performance data). It may be made part of both, if the information is applicable to both.

Step (7) - If the information is technical (i.e. certified pump head curve), it is reviewed by both the project responsible system engineer and the project responsible purchase engineer. The project responsible system engineer reviews the information against the original design basis (which could be a calculation). If there is a difference the responsible system engineer will reject or redo the calculation based on the new information. The project responsible purchase engineer reviews the information only against the Performance Data Sheet in the specification. Any needed changes to the specification are usually handled by a letter to the vendor, followed by an Addendum to the specification.

A2.5-1 (Detail) Moderate Energy Crack Analysis

River Bend FSAR section 3.6.2.1.5.2.2A indicates that the residual heat removal system qualifies as a moderate energy fluid system. The basis for this conclusion is that the system operates within the pressure-temperature conditions specified for high energy system less than 2 percent of the time that the system operates as a moderate energy fluid system. FSAR Appendix 3C, "Failure Mode Analysis For Pipe Breaks and Cracks", is provided to describe the specific piping failure protection provided; however, the effects of jet impingement, moderate energy pipe cracks and of spraying and compartment flooding as a result of breaks or cracks have not been addressed except to indicate that the information will be provided later.

Although the team recognized that the information was to be provided later, the team expected to find design activity in progress to support verification that the plant is protected from moderate energy pipe cracks.

Upon arriving at Stone & Webster's engineering office, the team found this not to be case. The team found that very little activity had been done. During a Stone and Webster presentation on high energy line break and moderate energy crack analysis, the team was given a copy of PMM-163, "Moderate Energy Line Crack (MELC) Evaluation." This memorandum establishes the procedure for performing an evaluation to verify that the plant can be brought safely to a cold shutdown condition and maintain spent fuel pool cooling (decay heat removal) following a moderate energy line crack coincident with a loss of offsite power, seismic event, and a worst case single active failure of a safety-related component. PMM-163 was issued on March 12, 1984. Since little work had been accomplished and since the implementing procedure was issued after the team's inspection cutoff date, the team decided not to spend any additional time reviewing this area.

A2.8-1 (Detail) Potential Loss of Suction Flow for the High Pressure Core Spray Pump

DESCRIPTION

The high pressure core spray pump draws suction from the condensate storage tank which is not Seismic Category 1. A Category 1 backup source (suppression pool) is provided. The pump is lined up to either one of the two sources by valves with preference being the condensate storage tank. Transfer to the suppression pool is made upon loss of tank static head in the line between tank and the shutoff valve (VF001). The valve that provides a path to the suppression pool is VF015. It is intended that only one of the valves be open at any one time. A single failure of the switch controlling the tank valve can cause the valve to close and produce loss of suction situation for the high pressure core spray pump if it were running when the valve was closed. This condition is not alarmed nor is transfer attempted as would be case on loss of the condensate storage tank. The pump would either be damaged or tripped. If tripped it can only be restarted by locally resetting the breaker at the pump location. There is a single switch for each of the suction line valves (VF001 and VF015). These switches are located in the control room. Depending on the availability of the other emergency core cooling system trains this could have safety implications. Although valve VF001 receives a signal to open upon initiation of a LOCA signal, annunciation of closure of this valve would enhance safety by providing indication of possible system inoperability.

REFERENCES

1. S&W Drawing 12210-FSK-32-4A.

A3.1-1 (Detail) Drawing Control

Changes are made to the issue-for-construction EP-series ASME III pipe drawings at Cherry Hill by periodic revisions to the drawings which incorporate outstanding Engineering and Design Coordination Reports and Nonconformance and Dispositions. Field changes are made to the ASME control drawings for pipe and support via Construction Revision Notice and Conditional Construction Revision Notice, or "redline" (Conditional Construction Revision Notice); see Reference 1.

The EP-series piping drawings that are issued for construction and which form the basis (along with the Shaw fabrication drawings) of the ASME control drawings prepared at the site are revised to reflect all revisions made to the ASME control drawings at the site. These revisions are effected by incorporating Engineering and Design Coordination Reports issued to reflect any changes made to the ASME control drawings due to Construction Revision Notices/Conditional Construction Revision Notices.

This dual revision process does not apply to pipe support drawings. Once Cherry Hill issues the BZ-series pipe support drawing for construction it is "frozen". Only the corresponding field prepared ASME control drawing is revised.

REFERENCES

1. S&W RBP 18.10-2 Handling Changes to QA Category I (ASME III) Pipe Supports and Piping, and Documentation/Approval of QA Category I Instrumentation Tubing Installation On-site.

A3.1-2 (Detail) References for Section 3

References for Section 3.1

1. GE Dwg. 76E424AA, Rev. 4.
2. S&W Specification No. 228.000, Piping Engineering and Design Specification, Rev. 1, Add. 8, dated 2/28/84.
3. S&W Specification No. 228.310, Design and Fabrication of Power Plant Piping Supports, Rev. 1, Add. 1, dated 9/11/81.
4. S&W Specification No. 228.315, Design and Fabrication of Mechanical Snubbers for Nuclear Power Plant Piping Supports, dated 10/3/83.
5. S&W Dwg. Nos. 12210-EP-71A through H, J, Rev. 1, various dates.
6. S&W Dwg. No. 1-BZ-71RC-CD, Rev. 1, dated 5/7/83.
7. S&W Dwg. Nos. 12210-EZ-71ZA, B, D, E, F, K original issues, various dates.
8. S&W Standard Line Designation Table for FSK 27.7 Residual Heat Removal (RHS), LDT Issue 33, dated 2/21/84.
9. S&W Calc. Nos. 12210-AX-71AF, -71P, -71AK, -71C, -71AE, -71K, various revisions and dates.
10. S&W EMTG-2 Stress Report for ASME Class 1 Piping.
11. S&W EAP 5.3 Preparation and Control Manual and Computerized Calculations (Nuclear Projects).
12. S&W PTP 0.1.2 Application of Safety Classes for BWR's.
13. S&W PTP 0.2.1 Establishing the Design Pressure and Temperature for a Fluid System.
14. S&W PTP 26.1.15 Information for Pipe Stress Analysis and Pipe Support Design/Analysis for Nuclear Power Plants.
15. S&W EMTP 9.3 Use of ARS Data by the Pipe Stress Analysis and Support Section.
16. S&W EMTP 12.9.12 Procedure to Evaluate Functional Capability for BWR Essential Piping.
17. S&W EMDM 82-7 Guidance for Locating Lumped Mass Points in a Piping System.
18. S&W EMDM 83-03 Pipe Stress Checklist for As-Built and Reconciliation Programs.
19. S&W EMTG 15A Standard Allowable Reactions on Equipment Nozzles.
20. S&W EMDM 80-4 Use of Pipe Support Stiffness Values in Pipe Stress Analysis.
21. S&W EMBM 81-04 Effect of the Spring Hanger Assembly Weight on Piping Systems.
22. S&W Doc. ME-110 NUPIPE-SW Piping Stress Analysis.
23. S&W RBP 18.10-2 Handling Changes to QA Category I (ASME III) Pipe Supports and Piping, and Documentation/Approval of QA Category I Instrumentation Tubing Installations Onsite.
24. S&W Specification No. 228.312, Field Fabrication and Erection of Pipe Supports ASME III, Code Class 1, 2, 3, and ANSI B31.1.
25. Design and Fabrication of Mechanical Snubbers for Nuclear Power Plant Piping Supports ASME III, Code Class 1.
26. S&W E&DCR C-31,464A.

References for Section 3.3

1. S&W Procedure for the Preparation, Review, Approval and Control of the Power Input Control Listing RBP 6.24-0, January 13, 1984.

A3.1-2 (Detail) continued.

References for Section 3.4

1. S&W Specification No. 228.000, Piping Engineering and Design Specification, Rev. 1, Add. 8, dated 2/28/84.
2. S&W Specification No. 228.310, Design and Fabrication of Power Plant Piping Supports, Rev. 1, Add. 1, dated 9/11/81.
3. S&W Specification No. 228.315, Design and Fabrication of Mechanical Snubbers for Nuclear Power Plant Piping Supports, dated 10/3/83.
4. S&W Dwg. Nos. 12210-EP-71A through H, J, Revs. 1, various dates.
5. S&W Dwg. No. 1-BZ-71RC-CD, Rev. 1, dated 5/7/83.
6. S&W Dwg. Nos. 12210-EZ-71ZA, B, D, E, F, K, original issues, various dates.
7. S&W Standard Line Designation Table for FSK 27.7 Residual Heat Removal (RHS), LDT Issue 33, dated 2/21/84.
8. S&W Calc. Nos. 12210-AX-71AF, -71P, -71AK, -71C, -71AE, -71K, various revisions and dates.

References for Section 3.5

1. S&W Specification No. 228.310, Design and Fabrication of Power Plant Piping Supports, Rev. 1, Add. 1, dated 9/11/81.
2. S&W Specification No. 228-312, Field Fabrication and Erection of Pipe Supports ASME III, Code Class 1, 2, 3, and ANSI B31.1.
3. Design and Fabrication of Mechanical Snubbers for Nuclear Power Plant Piping Supports ASME III, Code Class 1.
4. S&W E&DCR C-31,464A.

A3.2-1 (Detail) Stone & Webster Training and Assignments

Training of division personnel is a staff responsibility in the Engineering Mechanics Division. Data files on training of personnel are maintained by Engineering Mechanics Division management. Files may be sorted on the basis of individual staff members or by course offering. Selection of personnel for job assignments or for additional training assignments can be expedited using these files.

Training of entry level personnel consists of formal classroom training and job training under the guidance of senior level personnel. Entry level personnel are categorized as Career Development Engineers for their first year with the company. Schedules of training courses for 1981, 1982, and 1983 were made available. These courses are concentrated in July and August when they would provide the earliest opportunities for new college graduates. Training courses are offered for administrative procedures, engineering quality assurance requirements, and detailed technical design and evaluation procedures. Details of course content were not reviewed. The schedules which were made available showed that individual courses range from one hour to one day in length, e.g., the one course presented each year on equipment qualification was of one and one-half hours duration while courses on the subject of integral attachments continued for two days.

Training requirements have been recently established to provide certification of personnel to satisfy ASME code requirements. Training is also offered to assist in obtaining a professional engineering license. Management displays pride in the number of licensed professionals on the Cherry Hill staff.

Training records were reviewed for a sample of Engineering Mechanics Division personnel. Two junior and two senior individuals were selected from each of the areas of pipe stress analysis, pipe supports, and equipment qualification plus other individuals who had made specific contributions to work products examined during this inspection.

The following records were submitted by Stone & Webster for review:

PIPE STRESS ENGINEERS - 7 individuals; 2 Ph.D.s, 2 M.S.s, and 3 B.S.s; 3 with a professional license

Experience	7+ yrs.	4-5 yrs.	0-3 yrs.
Number of Individuals	5	1	1
Stone & Webster	3	2	2
Nuclear	3	2	2
River Bend	1	2	4

A3.2-1 (Detail) continued.

PIPE SUPPORT ENGINEERS - 6 individuals; 1 M.S., 5 B.S.; none with a professional license

Experience	7+ yrs.	4-5 yrs.	0-3 yrs.
Number of Individuals	3	1	2*
Stone & Webster	1	3	2*
Nuclear	1	3	2*
River Bend	0	4	2*

* Career Development Engineers

EQUIPMENT QUALIFICATION ENGINEERS - 6 individuals; 1 Ph.D., 3 M.S., and 2 B.S.; 1 with a professional license

Experience	7+ yrs.	4-5 yrs.	0-3 yrs.
Number of Individuals	4	2	0
Stone & Webster	3	1	2
Nuclear	3	1	2
River Bend	1	2	3

In general the records of Career Development Engineers showed numerous course listings directed at their present duties. Engineers with four to five years experience had fewer courses with less intensive coverage. Records of the more senior engineers indicated that they have a great deal of training but in some cases have taken the same course three or four times. The limited training for personnel who have been with Stone & Webster for four or five years suggests that the training effort was possibly neglected during a busy period or has been given a renewed emphasis in recent years and has not been required for individuals who joined Stone & Webster during that previous time frame.

REFERENCES

1. S&W EMTP 3.12.12-0 issued 3/12/84.
2. ANSI/ASME N626.3 Qualifications and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code, Section III, Division 1 and 2, Certifying Activities.

A3.2-2 (Detail) Stone & Webster Technical Procedures

Pipe Stress Information Index was issued to all Engineering Mechanics Division pipe stress engineers on March 23, 1983. This index tabulates technical memoranda, procedures, specifications, codes and regulations, and computer programs requiring the cognizance of the pipe stress engineer. The Index tabulates seven different types of in-house technical documents:

Engineering Assurance Procedures

Cherry Hill Engineering Mechanics Division Memoranda

Engineering Mechanics Technical Guidelines

Engineering Mechanics Technical Memoranda

Engineering Mechanics Technical Procedures

Engineering Mechanics Technical Reports

Power Division Technical Procedures

References 1-12 list a number of the documents tabulated in the index.

The Index also lists a number of ASME, ANSI, SRP, NUREG and RG specifications and regulations, pertinent Stone & Webster specifications and project procedures, and technical procedures governing the design and analysis of small bore piping. A compilation of the computer programs available for stress analysis is provided. Reference 13, for example, documents the in-house version of NUPIPE available to the Engineering Mechanics Division pipe stress engineers.

REFERENCES

1. S&W EMTG-2, Stress Report for ASME Class 1 Piping.
2. S&W EAP 5.3, Preparation and Control Manual and Computerized Calculations (Nuclear Projects).
3. S&W PTP 0.1.2, Application of Safety Classes for BWR's.
4. S&W PTP 0.2.1, Establishing the Design Pressure and Temperature for a Fluid System.
5. S&W PTP 26.1.15, Information for Pipe Stress Analysis and Pipe Support Design/Analysis for Nuclear Power Plants.
6. S&W EMTP 9.3, Use of ARS Data by the Pipe Stress Analysis and Support Section.
7. S&W EMTP 12.9.12, Procedure to Evaluate Functional Capability for BWR Essential Piping.
8. S&W EMDM 82-7, Guidance for Locating Lumped Mass Points in a Piping System.
9. S&W EMDM 83-03, Pipe Stress Checklist for As-Built and Reconciliations Program.
10. S&W EMTG 15A, Standard Allowable Reactions on Equipment Nozzles.
11. S&W EMDM 80-4, Use of Pipe Support Stiffness Values in Pipe Stress Analysis.
12. S&W EMDM 81-04, Effect of the Spring Hanger Assembly Weight on Piping Systems.
13. S&W Doc. ME-110, NUPIPE-SW Piping Stress Analysis.

A3.3-1 (Detail) Quencher Inputs and Interface

Interoffice Correspondence DP-760 transmitted data from the lead power engineer to the pipe stress engineer for safety/relief valve discharge pipe stress analysis. The Interoffice Correspondence does not appear to reference any vendor information relative to the quencher which is at the bottom end of the discharge pipe. The quencher is supplied by General Electric and an interface control drawing reference provides allowable nozzle loads on the quencher.

The pipe stress engineer referenced the available quencher drawing and other pertinent information in the stress analysis (Reference 3). Also listed in the analysis are some assumptions regarding the quenchers which are flagged as needing verification. The stress analysis also references a May 20, 1981, Stone & Webster letter to General Electric indicating that some quencher interface load limits have been exceeded in the results of the analysis. The letter requests that General Electric consider additional design features to provide increased allowable interface limits which Stone & Webster results could satisfy. The response from General Electric provides adequate limits and the matter is considered resolved subject to updating the necessary documents. It appears that Stone & Webster pipe stress personnel handled this situation very well notwithstanding the lack of formal input from the lead power engineer.

REFERENCES

1. S&W Interoffice Correspondence DP-760.
2. GE Drawing No. 795E574 Rev. 2, S&W File No. 0222-212-000-053C.
3. S&W Calculation 12210-AX-11H-1.
4. S&W Letter to GE RBV-1550, 81-056, 5/20/81.
5. GE Letter to S&W GSS-3108, 8/11/81.
6. S&W Letter to GE RBV-1669, 82-001, 1/04/82.

A3.4-1 (Detail) Stone & Webster Modeling Procedures

The six AX-series stress packages which comprise Loop B of the Low Pressure Coolant Injection mode of the Residual Heat Removal System were reviewed. The stress packages prepared by the Stone & Webster Engineering Mechanics Division pipe stress engineers were reviewed for technical accuracy and conformance to code requirements.

The piping geometry was reviewed to verify:

- pipe location in plan and elevation (see Deficiency 3.4.2)
- pipe support locations, restraint directions, and stiffnesses (see Deficiency D3.4-4).
- penetration, valve and equipment locations
- branch line decoupling
- node point locations

The piping material properties were reviewed to verify:

- pipe material
- stress allowables
- presence of insulation

The piping loads were checked to verify:

- load combinations
- damping values for amplified response spectra (see Deficiency D3.4-5)
- seismic anchor displacements
- pressures and temperatures
- added mass for trapeze hangers (see Deficiency D3.4-6)
- pipe dead load
- hand calculations for input thermal displacements

The stress packages were also reviewed to verify:

- correct ASME Class analysis
- functionality criteria (see Deficiency D3.4-7)
- frequency cutoff
- mode combination in accordance with RG 1.92

With respect to ASME classification Loop B of the Residual Heat Removal System Low Pressure Coolant Injection mode is composed of both Class 1 and Class 2 pipe. However, Class 1 pipe is initially analyzed per Class 2 requirements by Engineering Mechanics Division. Class 1 analysis is deferred until stress analysis of the final as-built pipe configuration.

Line E and Line H of the safety/relief valve discharge lines and main steam line C were also reviewed as part of the Automatic Depressurization System to evaluate the valve modeling procedures employed by Stone & Webster (see Deficiencies D3.4-1, D3.4-2).

A3.6-1 (Detail) Pump Design and Qualification

The residual heat removal pump, PC002B, is in the General Electric scope of supply. It was purchased to a specification which specified construction to codes and standards in effect on the date of award of the contract. Standards referenced include both ASME Section III and ASME Section VIII Division 1. The pump is required to be a class 2 component as defined by Section III with pressure parts to meet Section III NC-3300, for Class 2 vessels. Interface loadings between the piping and the pump are defined by a General Electric interface control drawing. The drawing specifies nozzle forces and moments and how they are to be combined for normal and upset conditions and for faulted conditions. It also specifies that natural frequencies of the attached piping shall be less than 14 Hz or greater than 28 Hz.

The specification requires design to an Safe Shutdown Earthquake spectrum of 7.5 g maximum horizontal and 6.5 g maximum vertical acceleration. This appears excessive for River Bend, but allows the pump to be used at any site. The Operating Basis Earthquake is specified to be 2/3 of the Safe Shutdown Earthquake. Calculations are required from the pump vendor for stresses in 41 items plus the seal cooling heat exchanger.

The vendor report was made available and examined. The report is certified to comply with Section III of the Code 1971 Edition through the Summer 1973 addenda. The vendors report was a thorough evaluation addressing all of the items required by the specification. Additional items were also addressed.

Many of the pressure parts were evaluated to the requirements of Section VIII rather than Section III. This includes the suction nozzle and discharge pipe and the discharge head. The suction and discharge flanges were evaluated to the requirements of both codes. Section III did not have detailed requirements for design of pumps at that time and it was recognized that the design requirements for Class 2 pressure vessels were essentially the same as the design requirements for Section VIII. No significance is attached by the team to this apparent lack of consistency in the report. These same areas were analyzed without consideration of the corrosion allowance, but with no significant impact on design since stresses were well within allowable limits. The nozzles of the pumps were evaluated using the Bijlaard method. Calculated principal stresses of 25,106 psi are compared to an allowable stress of 26,250 psi. This appears to be the critical area on the pump for the normal and upset loading condition and is sensitive to the nozzle loadings. Stone & Webster calculations do not report exceeding the specified interface loads. Natural frequencies of all parts satisfy requirements. Stresses due to seismic loadings are within limits.

REFERENCES

1. GE Purchase Specification No. 21A9514, Pumps, Auxiliary for Boiling Water Reactors (RHR & LPCS), S&W File No. 0221-435-000-004E.
2. GE Interface Control Pump and Motor, Residual Heat Removal System, GE Drawing No. 105D5064 Rev 4. S&W File No. 0221-431-000-009E.
3. Byron-Jackson Co. Design Report for Pump Serial Number 741-5-1420/22 1424/26.
4. Welding Research Council Bulletin No. 107.
5. S&W Calculation No. 12210-AX-71P-1, 1/11/83.

A3.6-2 (Detail) Valve Qualification Program

ORGANIZATION

The qualification program at Stone & Webster Cherry Hill has high level management attention. The section manager of the equipment qualification section reports directly to engineering management as would a division manager. Staff for the seismic/dynamic qualification group are part of the Engineering Mechanics Division. Staff are not assigned to the River Bend Project. A coordinator for equipment qualification for the River Bend Project reports directly to the Project Engineer.

PROCEDURES

Procedures for seismic qualification of equipment, References 1-8, are developed and maintained by the Engineering Mechanics Division. Engineering Mechanics provides input for seismic/dynamic qualification requirements in specifications and notifies the project principal pipe stress engineer of the design input acceleration levels for in-line equipment. Stress cycle information is also transmitted to provide a basis for fatigue evaluation. When pipe stress analysis is completed, the predicted acceleration values of that equipment is reported to the seismic qualification group. The present procedures and guidance appear adequate to meet NRC requirements and licensing commitments if followed by highly competent engineers in all the fields involved.

A procedural problem was noted in interface controls. Procedures require that input be provided to pipe stress analysts on equipment interface controls and that the supplier of that information be notified only if the results of analysis predict that the interface controls will be exceeded in service. This can result in situations such as that noted in Deficiency D3.3-3 related to the safety/relief valves wherein the analyst did not evaluate the interface controls and did not report results. An improved procedure (Reference 18) for input to the pipe stress analysis engineer was recently developed. This would alleviate this problem by requiring review and approval of data by the pipe stress engineer before its official submittal by the power group.

TRAINING

Training programs (References 9-12) have been conducted for members of the Electrical Division, Controls Discipline, Power Discipline, and the responsible engineers and staff of Engineering Mechanics Division working on the River Bend Project. The training consists of a one and one-half hour presentation on administrative procedures.

STATUS

The status of the program for seismic/dynamic qualification of equipment for the River Bend Project is in a state of transition. While the industry has been aware for some time of the hydrodynamic loads resulting from safety valve discharges, large scale test results have caused large increases in estimates of these loads. Significant changes have been required in qualification requirements since the dominance of hydrodynamic loading was established for some equipment. Hydrodynamic loads may have acceleration response peak values which exceed those from seismic response and also have significant response at frequencies as high as 100 Hz. The frequency range can be the most significant factor for equipment which might normally be considered to be fully qualified but which was not previously evaluated for amplified response above the seismic cut off frequency of 33 Hz.

A3.6-2 (Detail) continued.

Stone & Webster is participating with General Electric in a cooperative program to qualify valve operators for acceleration response spectra more appropriate for hydrodynamic loads. Stone & Webster-River Bend Project also plans to utilize the Stone & Webster staff to extrapolate previous vendor submittals to establish and certify seismic/dynamic adequacy of some components. Apparently Gulf States will accept Stone & Webster certification in lieu of vendor certifications. Special procedures may be needed for this effort since the expertise to understand and predict the limits of performance of active mechanical equipment is usually found only in personnel of established vendor engineering and test organizations.

There is no quality controlled list by equipment mark number of the status of equipment which is to be seismically qualified. Equipment qualification status is followed by its procurement specification number. Since a variety of valves are procured to a given specification, the status of individual valves cannot be determined with ease. Such a quality controlled list is to be developed for the submittal to the NRC Seismic Qualification Review Team.

The existing documents (References 13, 14, and 15) for seismic qualification of valve 1E12*MOV FO42B were examined to evaluate the quality of work and conformance with procedures for seismic/dynamic qualification. Procedures (Reference 3) still provide for a cut off frequency of 33 Hz. for seismic qualification. The vendor report on seismic analysis of FO42B was marked approved as revised on 3/3/83. Four reviewer comments are provided which are to be included in a revision. One comment requests justification for the claim that valve nozzle analysis is not required. A revised report in response to the comments was not yet available at the time of the inspection. A major revision to the report would be needed to satisfy later developments in loadings.

Reference 13 was intended to provide justification for several similar valves. Valve FO42 was selected for evaluation since it was considered to have the heaviest operator. A weight of 1000 lbs was assumed for the operator. The assumption was not confirmed in the report but other data indicates no significant error resulted from this assumption. A seismic load of 3 g in each of the directions was the original specified seismic input to the vendor analysis, plus a 1 g load from the weight. Subsequent Stone & Webster pipe stress analysis predicted that the valve would be subject to 4.98 g. This value exceeds the 3.5 g value which was the specified limit for the Stone & Webster calculation.

In the vendor report (Reference 14), calculated stresses for the valve yoke are the most critical area. These are compared to allowable stress values. Two different allowable stresses are presented; 22,500 psi and 26,250 psi. Neither of these is referred to a specific material or temperature for the yoke. The vendor drawing provides for use of three different materials for the yoke. Other portions of the calculation do refer to specific materials and temperatures for values of design stress. It is concluded that review of this vendor report was not sufficiently thorough to detect such inconsistencies and omissions. The calculated lowest natural frequency of the valve is 58 Hz., which indicates potential problems with respect to the hydrodynamic loads.

REFERENCES

1. S&W EMTP 8.22, Seismic Report Standard Review Procedure.
2. S&W EMTP 10.2, Generation and Control of ARS by the Mechanical Section.
3. S&W EMTP 10.16, Equipment Seismic Requirements.

A3.6-2 (Detail) continued.

4. S&W EMDM 82-12, Preparation, Review, and Control of Manual and Computer Calculations.
5. S&W EMDM 82-22, Hydrodynamic Loads for Equipment Qualification Efforts.
6. S&W EMDM 82-23, Load Combinations for Equipment Qualification and Piping Affected by Hydrodynamic Loads.
8. S&W EMDM 81-03, Valve Modeling Procedure.
9. S&W Notice of Conference call by A. Blum, J.O. 12210, 2/8/84.
10. S&W Notice of Conference call by A. Blum, J.O. 12210, ?/7/84.
11. S&W Notice of Conference call by A. Blum, J.O. 12210, 2/7/84.
12. S&W Notice of Conference call by A. Blum, J.O. 12210, 2/10/84.
13. Velan Report SR 6575, Seismic Analysis of 10" Gate Valve, 8/3/82.
14. Theory for Velan Nuclear Valves, 11/22/73 S&W File No. 4278-212-047-024A.
15. Velan Print P.3-5501-N18 Rev H., S&W file No. 0228-212-047-008I.
16. S&W EMD Transmittal of Design Information, RBI-EQ-316, 4/27/84.
17. S&W Calculation No. 12210-AX-71K-3, 7/28/82
18. S&W RBP.6.24-1, Procedure for the Preparation Review, Approval, and Control of the Power Input Control Listing, approved 2/24/84, pages dated 2/16/84.

A3.6-3 (Detail) Vacuum Breaker Valve Design and Qualification

BACKGROUND

Vacuum breaker valves are added to the safety/relief valve discharge lines to avoid drawing water from the suppression pool up into the discharge lines when steam in the line condenses. Two valves are installed in each line. Leakage through these valves would discharge steam into the drywell during safety/relief valve discharge. Although this would not constitute a violation of the containment, excessive leakage into the drywell could result in an unnecessary scram. Repeated large leakages into the drywell could call into question the environmental qualification of equipment located in the drywell. Special requirements related to opening time and flow characteristics are applied to the valves.

DESCRIPTION

The valves selected for service are 10" 600 lb. design Velan swing check valves. The valve specification defines test requirements to qualify these valves for this service. The general piping specification allows selection of 300 lb. design valves for this service but the heavier valve was selected. General Electric specified specific performance requirements for the relieving capability of these valves. Several addenda to the specification were needed to properly establish these requirements. Test results were provided to demonstrate the capability of the valves to meet the functional requirements. However, these results would not qualify the valves as code stamped safety valves. Gulf States and Stone & Webster correspondence established that these valves would not need to be classified as NV stamped safety valves. This can be considered an acceptable conclusion since the valves were not intended to provide assurance of pressure boundary integrity.

The valves were designed as Code Class 3 valves which do not require detailed stress analysis, but may be designed by alternate design rules of ANSI B16.34.

Questions were raised about the capability of the valves to resist damage observed in other vacuum breaker and check valves. Stress analysis was submitted by Velan to [illegible]tify that check valve slamming will not damage the valve with the anticipated se[illegible]e conditions.

Results of pipe stress analysis predicted accelerations as high as 16.8 g for the lower vacuum breaker valves. No stress analysis is available to demonstrate the capability of these valves to withstand such forces and no seismic qualification tests were required since the valves do not have an extended structure. Simplified analysis performed as part of this review does not predict that damage from such accelerations should occur.

Repeated severe thermal cycles from safety/relief valve operation must also be considered as a source of damage to the valves related to observed damage. No thermal cycling tests are planned and no analysis is presented to demonstrate the capability of the valves to resist damage from cyclic thermal loading. Since such analyses are costly and of questionable value, testing or monitoring of the valves in service can be more effective.

A3.6-3 (Detail) continued.

Reference 6 commits Gulf States to monitoring the valves in service. This should be included as part of the Inservice Inspection program. In addition to leakage tests, a simple test to measure the force required to partially open the valve would constitute effective monitoring of functional adequacy.

REFERENCES

1. S&W Specification RBS 228.211, Addendum 6.
2. S&W Specification RBS 228.000.
3. Velan Test Report, 10" Swing Check Valve Vacuum Breaker Tests, 2/1/82.
4. S&W Letter to GSU RBS-7163, 12/15/81.
5. S&W Letter to GSU RBS-7419, 3/24/82.
6. GSU Letter to S&W RBG 12,494 S-6,392.
7. 2.7-1 (Deficiency) Interim Problem Report.
8. Velan Engineering Calculation SC-8045, 12/16/80.
9. S&W Calculation 12210-AX11 (A thru R), attachment 10C to Piha, 5/14/82.
10. GE Design Specification, Nuclear Boiler System, No. 22A4622 6/4/76 Rev. 5, 3/16/83.

A4.1-1 (Detail) References for Section 4

References for Section 4.1

1. GE Document No. 22A4365, Interim Containment Loads Report, Rev. 2, October 1978.
2. ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containment, Article CC-3000, July 1, 1977.
3. American Concrete Institute, ACI 318-71, Building Code Requirements for Reinforced Concrete (with Commentary).
4. American Institute for Steel Construction, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, 1969 Edition (including Supplements 1, 2, & 3).
5. S&W Structural Design Criteria, Document No. 200.010, July 8, 1983.
6. S&W Geotechnical Design Criteria, Document No. 210.090, May 9, 1980.

References for Section 4.2

1. S&W EAP 5.3, Preparation and Control of Manual and Computerized Calculations, Rev. 3, 1/31/79.
2. S&W STP 11-5, Control of Structural Division Calculations, Rev. 2, 11/3/82.
3. S&W CHOC-EMTR-401-0, Design Procedure for Category I Conduit Systems for River Bend Project, 8/16/80.
4. S&W Technical Guideline No. STG 19.4-1, Seismic Design of Conduit Support Systems, 12/30/82.

References for Section 4.3

1. S&W Calculation, 201.130.085, Verification of Reactor Building Seismic Analysis, Revision 0, 9/25/80.
2. S&W Calculation, 201.120.124, Seismic Analysis of Reactor Building With Concrete Fix, 1/22/82.
3. S&W Calculation, 201.130.010, Seismic Analysis of Auxiliary Building, Revision 1, 3/29/76.

References for Section 4.4

1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments, 1977 Edition.
2. S&W Calculation No. 12210-201.120-105, Drywell Design, Rev. 0. 1/13/83.
3. S&W Structural Design Criteria for River Bend Station Units 1 and 2, Document No. 200.010, August 3, 1983.
4. S&W Calculation No. 201.120-068, Primary Shield Wall Design (Supplementary Calculations), Rev. 0, 1/10/84.
5. American Institute of Steel Construction, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Seventh Edition, February 12, 1969.
6. S&W Calculation No. 201.120-048, Weir Wall Design, Rev. 0 12/19/83.
7. S&W Calculation No. 201.120-070, Design of RPV Pedestal - Supplemental Calculations, Rev. 0, 5/10/83.

A4.1-1 (Detail) continued.

References for Section 4.5

1. S&W Calculation, 219.710-EBA-2039, SHELL-1 Containment Stress Analysis, Revision 1, 12/20/83.
2. S&W Calculation, 219.710-EBA-1093, Buckling Analysis of Steel Containment, 12/14/82.
3. W.J. Woolley Company, Submittal For Design Specification No. 219.711, Revision 3.

References for Section 4.6

1. S&W Calculation No. 201.120-020.
2. S&W Calculation No. 201.120-067, 10/29/81.

References for Section 4.7

1. S&W Calculation No. 201.120-096, Verification of Reactor Building Mat Design.

References for Section 4.8

1. GE CLR 22A4365, Containment Load Report, Rev. 4, Jan. 1980 (not updated).
2. GSU Report on Safety Relief Valve Loads, River Bend, Units 1&2, Rev. 0, April 1977 (not updated).
3. NUREG 0802 Safety Relief Valve Quencher Loads Evaluation for BWR Mark II and III Containments, T.M. Su, October 1982.
4. NUREG 0978 Mark III LOCA Related Hydrodynamic Load Definition, February 1984.
5. S&W Calculation No. S54.3500 Floor Grating Design, Drywell Reactor Building.
6. S&W Calculation No. 201.120-035, GHOSH Axisymmetric Finite Element Model for Safety Relief Valve and LOCA Loads, Rev. 0, 10/26/83.
7. GESSAR 238 General Electric Nuclear Island Standard Design.

References for Section 4.9

1. S&W Structural Design Criteria for River Bend Station, Rev. 3, 7/8/83.
2. S&W Calculation No. 201.120-105, Rev. 0, 1/13/83.
3. S&W DEM 1033, 3/20/80 and DEM 1370, 2/27/81.
4. S&W Calculation No. 554.700.
5. S&W Calculation No. 554.3500, Floor Grating Design, Drywell, Reactor Building.

References for Section 4.10

1. S&W Pipe Support Specification No. 228.312, Rev. 3.
2. S&W Structural Design Criteria for River Bend Station, Rev. 3, 7/8/83.

References for Section 4.11

1. S&W Calculation, C66.201, Auxiliary Building Foundation Mat Analysis & Design, Revision 2, 4/16/82.
2. S&W Calculation, C67.114, Auxiliary Building Floor Slab at Elevation 114'-0", Revision 1, 12/15/81.

A4.1-1 (Detail) continued.

3. S&W Calculation, C67.600, Auxiliary Building - Floor Slab at Elevation 97'-9" in Main Steam Tunnel, Revision 1, 1/11/82.
4. S&W Calculation, C67.700, Auxiliary Building - Floor Slab at Elevation 114'-0" in Main Steam Tunnel, Revision 1, 10/29/81.
5. S&W Calculation, S66.296D, Auxiliary Building - Floor Framing at Elevation 95'-9", Revision 3, 3/27/84.
6. S&W Calculation, S66.185, Auxiliary Building - Floor Framing at Elevation 141'-0", Revision 3, 3/27/84.
7. S&W Procedure For Structural Load Verification, Approved By: K.S. Jadeja, 11/7/83.

References for Section 4.12

1. S&W Addendum No. 1 and Specification No. 228.180, Shop Fabrication, Field Fabrication, Field Erection and Testing of Control Rod Drive System Piping, ASME Code, Section III, Division 1, Class 2 and ANSI B31.1, Issued 10/27/80.
2. RCI Outline of the Analysis of the Control Rod Drive Hydraulic System Piping and Supports for the River Bend Nuclear Power Station Unit 1, Document: SA-932-DAO, 6/1/81.

References for Section 4.13

S&W Drawings:	1-BZ-71RA Hanger No. IRHS	PSST 3087A2	dated 4/10/1982
	1-BZ-71RE Hanger No. IRHS	PSSP 3091A1	dated 8/23/1983
	1-BZ-71RF Hanger No. IRHS	PSSH 3092A1	dated 1/27/1982
	1-BZ-71TL Hanger No. IRHS	PSSP 3128A2	dated 2/18/1982
	1-BZ-71CW Hanger No. IRHS	PSST 2094A2	dated 4/4/1982
	1-BZ-71EP Hanger No. IRHS	PSA 21314A2	dated 11/10/1982
	1-BZ-71TK Hanger No. IRHS	PSSP 2276A2	dated 5/19/1982
	1-BZ-71LZ Hanger No. IRHS	PSSP 2276A2	dated 5/19/1982
	1-BZ-71CZ Hanger No. IRHS	PSSH 2097A2	dated 5/6/1981

References for Section 4.14

1. S&W Calculation, C66.401, Auxiliary Building - Equipment Foundations and Anchor Bolts, Revision 2, 4/19/82.
2. S&W Calculation, 221.900 HBA 1699, Component Support RHR Heat Exchanger, Revision 4, 8/18/80.

References for Section 4.15

1. S&W Guidelines for Battery Racks.
2. S&W E&DCR C-20, 908A, dated 12/7/82.

References for Section 4.16

1. RCI Quality Assurance Manual, Second Edition, Revision 9, September 21, 1983.
2. RCI Quality Assurance Instruction Book, January 9, 1981.
3. RCI Appendix C to SA-932-DAO, Pipe Rupture and Pipe Whip Evaluation, Rev. 0, dated March 4, 1983.
4. RCI Appendix D to SA-932-DAO, Verification Description of Computer Programs Used for the River Bend Project, Rev. 0, May 27, 1983.

A4.1-1 (Detail) continued.

5. RCI Document SA-932-DAO, Outline for the Analysis of the Control Rod Drive Hydraulic System Piping and Supports for the River Bend Nuclear Power Station, Unit 1, Rev. 4, November 14, 1983.
6. Standard Review Plan (NUREG-0800) Section 3.7.2, Seismic System Analysis, Rev. 1, July 1981.

References for Section 4.17

1. S&W Drawing No. 12210-EC-67A-4, Floor Plan - EL. 95'-9" Outline Auxiliary Building, Revision 4, 4/14/81.
2. S&W Drawing No. 12210-ES-66A-7, Floor Framing Plan & Dets. El. 95-9" Auxiliary Building, Revision 7, 5/5/80.
3. S&W, Various Concrete Strength Test Reports.
4. S&W Drawing No. 12210-EC-66G-5, Eqpt. Pad & Anc. Bolt Sched. Auxiliary Building, Revision 5, 1/7/84.
5. S&W Drawing No. 12210-EC-58BD-5, Anchor Bolt Schedule & Details Control Building, Revision 5, 9/27/83.
6. S&W Nonconformance and Disposition Report No. 2636, 8/10/82.

References for Section 4.18

1. F.S.A.R. Section 2.4 - Hydrologic Engineering.
2. F.S.A.R. Section 2.5.4 - Stability of Subsurface Materials.
3. F.S.A.R. Figures 2.4.-1 to 52.
4. F.S.A.R. Figures 2.5-66 to 90.
5. F.S.A.R. Figures 2.5-98 to 106.
6. F.S.A.R. Tables 2.5-9 to 21.
7. F.S.A.R. Appendix 2K - Selected Tables and Figures from Report on "Soil Testing Undisturbed Samples of Tertiary (Pascagoula) Clays at River Bend Station."
8. F.S.A.R. Appendix 2L - Report on "Additional Sampling and Testing of the Foundation Soils at River Bend Station."
9. S&W Report on the "Liquefaction Potential of Plant Backfill" (Copy #42)
10. S&W Report on "Standard Geotechnical Procedures for Nuclear Power Plants" Doc. No. 50TRT0508 (March 1977).
11. S&W Calculation #85A (Rev. 3/28/77) Dynamic Lateral Pressures for Aux, Control and D.G. Bldg.
12. S&W Calculation #131A (Rev. 8/11/77) Dynamic Lateral Pressures for Reactor, Fuel and S.S.W.T. Bldg.
13. S&W Calculation #143 (6/28/78) Frequency Distribution of Density for Class IA Backfill.
14. S&W Calculation #168 (9/24/79) Revised Geotechnical Design Criteria.
15. S&W Calculation #201.120-092 (5/9/79) Verification of Soil Spring Constants for Reactor Building Mat.
16. S&W Document #210.090 (5/9/80) Geotechnical Design Criteria.
17. S&W Calculation #228 (11/28/83) Shear Modulus of Foundation Soils.
18. S&W Specification No. RBS -210.100 (10/27/80) Site Development Work.
19. S&W Specification No. RBS -210.100 Add #1 (12/18/81) Site Development Work.
20. S&W Specification No. RBS -210.100 Add #2 (3/2/83) Site Development Work.
21. S&W Specification No. RBS -210.076 (4/29/81). Field Procurement, Operation and Maintenance of Geotechnical Instrumentation.
22. S&W Specification No. RBS -210.076 Add #1 (3/28/84).

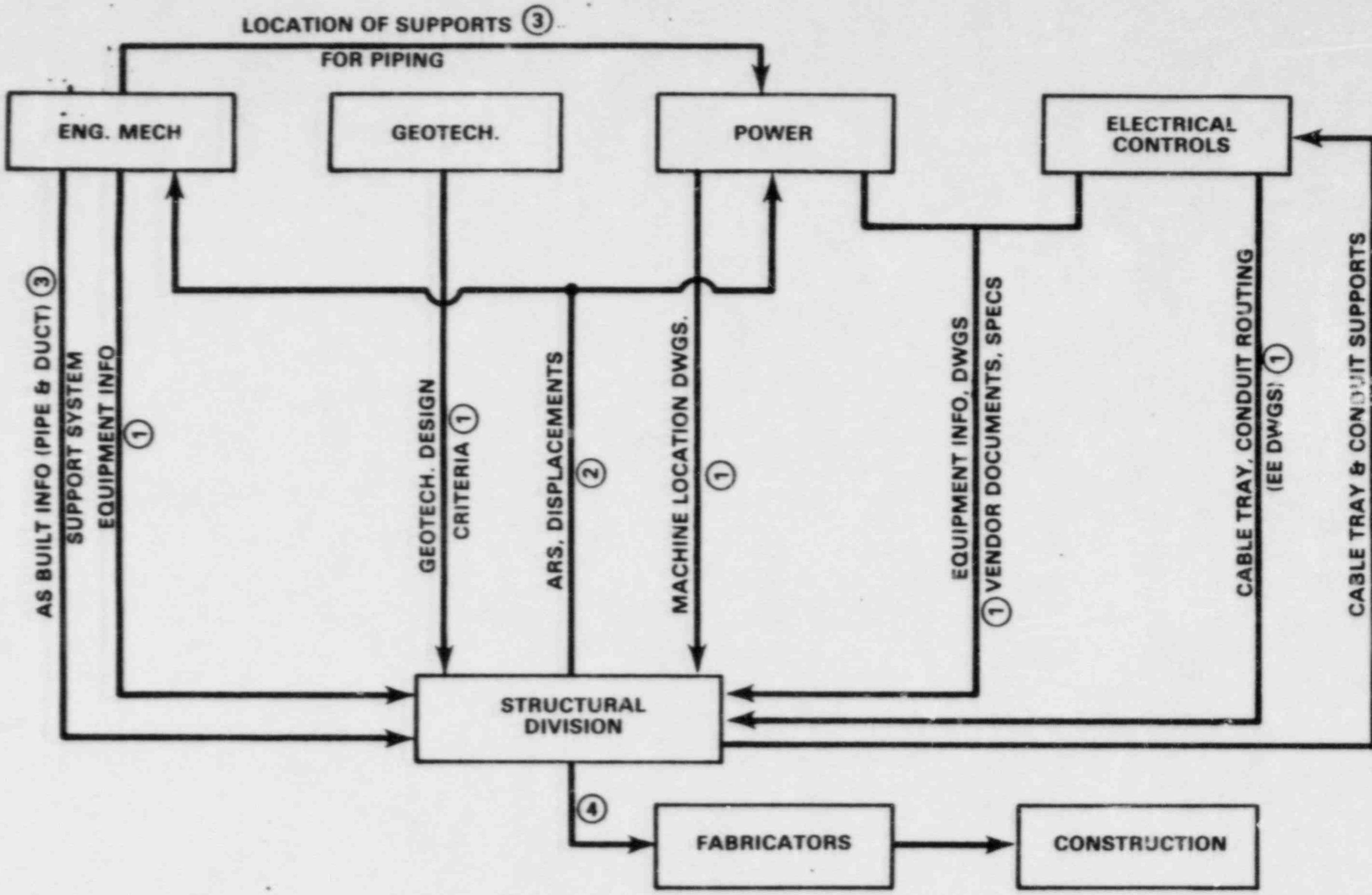

Figure 4.1-2 Stone & Webster Interdisciplinary flow

LEGEND:

◯ Indicate the Sequence of Flow of information

A4.3-1 (Detail) Mass and Stiffness Properties of Reactor Building

A lumped mass model with weightless beams connecting the masses was used in the seismic analysis of the Reactor Building. Each mass point had six degrees of freedom. The masses and their properties were calculated from the structural drawings by manual and computer calculations. The lumped masses included all dead and live load, and large pieces of equipment. The weights of the attachments on the primary shield wall from Reactor Controls, Inc. were not included in the calculation of the lumped masses. The team examined this aspect and found that this had no impact on design or analysis because the masses of these attachments were small compared to the lumped mass at the same elevation. The center of gravity of each mass was calculated to address torsional effects.

The seismic mathematical model of the Reactor Building consisted of the stick models for the shield building, the steel containment, the pedestal and the primary shield, the reactor pressure vessel (the shell and the internals), and the drywell. The mass and stiffness properties of the reactor pressure vessel were obtained by simplifying the model provided by General Electric.

The bending and shear stiffnesses of each member connecting the masses were obtained by manual and computer calculations. Different concrete and steel properties were used for the various structural elements in the Reactor Building. The mathematical model was revised due to the concrete fix between the shield building and the steel containment.

Elastic soil springs were attached to the lumped mass model at the mat level to account for soil structure interaction. Translational, vertical, rocking and torsional soil springs were included to represent the subgrade.

The team judged that the methods used were appropriate.

A4.3-2 (Detail) Development of Amplified Response Spectra Curves

The amplified response spectra curves were developed for the Safe Shutdown Earthquake and Operating Basis Earthquake. These curves were generated for three motions (2 horizontal and 1 vertical). The curves obtained from three soil spring constants were enveloped to account for the variation in subgrade modulii. The amplified response spectra were developed for frequencies varying from 0.83 Hz to 100 Hz. Frequencies for the calculation of amplified response spectra were selected to include the natural frequencies of the structure which correspond to the peak of the amplified response spectra. Greater computational details were used around the peaks.

The response spectra curves were developed by the Structural Division Staff Group and were written on a tape file. This file contains the amplified response spectra curves which were the envelope of three soil cases. A transmittal from the structural group to other groups within the River Bend project identifies the tape number and the revision of the amplified response spectra curves. Other groups which use these curves can access this tape and peak spread them as needed. Usually a peak spreading of -20 to +25% was used.

A4.4-1 (Detail) Consistency of Computer Programs

In order to test the consistency between the output of the SHELL 1 and the input of the NEWSECT programs, as well as the accuracy of the tabulations in the calculations, the inspector requested that the tabulated value be traced to the actual SHELL 1 printout. The designer readily produced the corresponding microfiche records which provided the required documentation.

This verification provided assurance that the documentation and controls of the calculations which have been filed in the Central Document System are readily traceable.

A4.4-2 (Detail) Traceability of Design Documents

We requested that some information referenced in the calculations be provided as evidence of traceability of the design parameters and of an effective filing system. One of the documents selected was the General Electric "Interim Containment Loads Report," G.E. Document No. 22A4365 Rev. 2, October 1978, which was the basis for pool swell loads used in the design of the drywell. A copy of this report could not be found in the central files, but later revisions (Nos. 3 and 4) were listed in the index. We accepted this as an evidence of filing of the document. A copy of Revision 2 was found in the Structural Division files and produced as evidence that the design documents are retained so that the design information can be traced to its source.

Another document selected was a memorandum entitled "Pedestal Temperature Profile for Normal Operating and Accident Conditions," with attachments, from V. P. Gupta to B. Ebbeson, dated November 27, 1979. This document could not be found in the central files, although it was located as an attachment to the calculations.

Traceability of two other documents were tested:

(a) Calculations No. 201.120-113, Verification of Primary Shield Wall and Reactor Pressure Vessel Bolt Designs and Embedments Rev. 0, dated December 8, 1981, and

(b) General Electric Document No. 22A46-71, "Reactor, Type: Installation Instructions", Rev. 1, dated June 2, 1980.

Both of these documents were found in the central document file system. They were stored on tapes.

The above cases indicate that although in the past the central document file system was not complete, the system appears to be improved and recent documentation seems to be filed in an orderly fashion and is readily retrievable. In conclusion, the team found the documentation system to be controlled.

A4.4-3 (Detail) Verification of Computer Codes

During the inspection, the team observed that not all of the computer codes used in design and analyses were identified in Appendix 3A of the FSAR. We discussed this matter with the Stone & Webster staff and were informed that the codes specified in Appendix 3A are those which were mentioned in the body of the FSAR. The remaining codes have been verified by the Stone & Webster Headquarters in Boston and the pertinent documentation is available on demand. As part of our inspection we requested that the user's manual for the computer code "ENVELOPE", ST-241, be brought to us for review. The manual was kept in central files under controlled conditions.

A4.9-1 (Detail) Floor Grating

Floor grating is used in the Reactor Building to provide walkways. They are not used for support of safety related equipment; therefore, Stone & Webster treats them as non Category I, but requires that they not become missiles under seismic and LOCA loads. The grating material is commercial grade steel. Support for the grating is provided by structural steel members. They are held down by bolted clamps.

Inasmuch as the froth loads are 12 psig on the entire area of grating and the fact that jet impingment pressures are high, the grating are loaded to high stresses and high reactions. Since the steel grating is commercial grade, without mill certification reports, there is some uncertainty concerning the yield stress. Therefore, the grating may yield; however, yielding by itself is not of concern because it would not be expected to cause failure.

The clamping of the edges were apparently not designed for these high loads. The team noted that Stone & Webster identified and is addressing the problem. The team anticipates that this area will be adequately tracked and resolved.

REFERENCES

1. S&W Calculation No. 554.3500 Floor Grating Design, Drywell, Reactor Building (currently being designed).

A4.11-1 (Detail) Analysis & Design of Mat

The Auxiliary Building foundation mat was analyzed by the use of a finite element computer program called ICES STRUDL. Shell elements were employed to model the mat. The walls that are attached to the top of the mat were modeled as beams to account for the stiffness of these structural elements. The input to the computer program was prepared by manual calculations. Soil springs were calculated for each node point using the contributory areas. The mat analysis considered three soil shear moduli (12, 18, 24 ksi) in calculating the springs attached to the nodes. The analysis performed included various loadings and loading combinations. The maximum bending moments were screened by the computer and summarized manually on the mathematical model. The forces and moments due to pump shaft housings were later superimposed.

Flexural steel was provided in the mat in accordance with the requirements of the ACI code. Shear reinforcement was not provided (see Deficiency D4.11-2). Pump housings were designed in accordance with the code requirements.

A4.16-1 (Detail) Stone & Webster Specification for Reactor Controls, Inc. Contract

The purchase specification for Reactor Controls Inc. services, Reference 1, has undergone significant changes. Addendum No. 2 added the pipe supports to the scope of the contract. Addendum No. 3 added water hammer from rapid opening of the control rod drive inlet line scram valve as an additional loading condition. This was related to a board notification of this problem which was prompted by a Reactor Controls, Inc. report. Addendum No. 5 added about 1,000 pages of loading condition details to the specification and defined pipe break loading conditions. Fourteen Engineering and Design Coordination Reports have been issued against the specification after Addendum No. 5. The first seven of these changes address specification changes and information which appear to be in response to needs developing during the design evaluation process. Subsequent changes arose from the need to upgrade strap supports for control rod drive piping which had been supplied by General Electric. One problem which resulted in a nonconformance report concerned lack of 100% inspection by Reactor Controls, Inc. of fit up welds. Reactor Controls, Inc. had conformed to their procedure requiring 10% inspection rather than the 100% inspection required by the ASME code.

Review of Reactor Controls, Inc. problems at the site yielded a minor deficiency wherein a piping drawing change was reissued to the fabrication shop without being noticed as a revision. Since the change was in response to a problem detected in the shop, the effect of such a deficiency is negligible.

REFERENCES

1. S&W Specification 228.180, Shop Fabrication Field Fabrication Field Erection and Testing of Control Rod Drive System Piping, 2/1/80.
2. NRC Memorandum Board Notification No. 82-31 with attachment.
3. Nonconformance Report RB-051.
4. RCI Drawing RB-110.

A5.3-1 (Detail) Motor Operated Valves

During the inspection, the team observed that Stone & Webster was actively engaged in the equipment environmental qualification program for River Bend. We also observed that detailed work lists, schedules, and a problem tracking log have been developed in preparation for a NRC environmental qualification audit scheduled for October 1984. A summary of the Stone & Webster qualification effort and our review of environmental qualification of motor operated valve actuators are presented below.

The team determined that the Stone & Webster balance of plant environmental qualification effort consists of five parts: seismic and dynamic, environmental, mechanical, General Electric interface, and licensing.

With respect to environmental qualification, the effort consists of a review of 50 specifications and associated vendor qualification test reports and analyses. Detailed Environmental Design Criteria have been developed and issued, and equipment post-operability periods have been developed. System component worksheets for each equipment item and a class 1E master equipment list are being developed.

Specification reviews were completed in March 1984 and action items were added to the tracking log. Of 1750 equipment mark numbers 1000 are located in harsh environment areas. As a result of this effort, Stone & Webster anticipates specification updates, supplemental analyses and reports, and equipment replacement or relocation. Prior to October, 1983, Stone & Webster used project procedure RBP-3.6 which required a review of the vendor's environmental qualification documentation solely with respect to the requirements of the specification. Subsequently, project procedure PMM-110 was issued which revised the methodology of qualification review to require a review of the vendor documentation against the actual equipment location and the Environmental Design Criteria. We learned that Stone & Webster deliberately made a management decision to delay significant efforts on environmental qualification until this stage of the project in order to derive a checking and verification process on the design. As a result, the Stone & Webster Equipment Qualification group has now become a focal point on the project.

Stone & Webster has also developed an Equipment Qualification Document which summarizes the results of the environmental qualification program for safety-related balance of plant equipment and supports section 3.11 of the Final Safety Analysis Report. Safety-related equipment for River Bend is required to be qualified to the Criteria of NUREG-0588 for Category I plants (Construction permit SER dated July 1, 1974 or later), the guidelines presented in Regulatory Guide 1.89, and the requirements of IEEE-323(1974).

With respect to equipment supplied by General Electric, Gulf States has contracted General Electric to upgrade the environmental qualification of safety-related equipment within their scope of supply to comply with the criteria of IEEE -323-(1974). The General Electric qualification program is a generic program for Boiling Water Reactor plants designated as the Phase III program. Each plant will transmit to General Electric post-accident environmental parameters for harsh environmental zones. General Electric will then produce composite environmental envelopes and qualify the equipment to the composite curves. A project unique qualification report will then be issued for General Electric supplied equipment. Stone & Webster has been assigned the responsibility to review the General Electric pretest reports and evaluations for proper interfacing with the Environmental Design Criteria and product performance and qualification

A5.3-1 (Detail) continued.

specfications. Gulf States has the responsibility to review the General Electric test reports.

We focused our review on Stone & Webster's activities with respect to determination of environmental qualification of safety-related motor operated valve actuators.

Motor operated valves 1E12*F042A and B, loop A and B Low Pressure Coolant Injection valves, are installed inside the containment. These valves are normally closed and are required to open under post-accident design basis conditions (Reference 5). The nameplate for the valve actuator identified the motor insulation system as Class B. Class B motor insulation systems are not environmentally qualified for in-containment applications due to temperature and radiation limits.

Stone & Webster Specification 228.212 was used to procure the valves and their Limitorque motor actuators. The Stone & Webster valve data sheet identified the valve location as "Auxiliary Building, Outside Containment." This data sheet was signed only by the data sheet preparer (i.e., a checker or an approval signature was not on the data sheet); the initial Stone & Webster data sheet was prepared on February 20, 1975 and was subsequently revised on June 27, 1975 (Revision 1) and December 10, 1979 (Revision 2). The vendor complied with the specification data sheets and supplied "outside containment" Limitorque motor operated valve actuators.

We determined from the environmental qualification data master list and the environmental design criteria zone maps that valves 1E12*F042A and B are installed in the Reactor Building in Zone CT-G. The post-accident environmental conditions at Zone CT-G are 165^{o}F, 9 psig, 100% RH and 7 x 10^{7} rads. We reviewed the Limitorque environmental qualification test report B0058 and determined that these motor actuators are not environmentally qualified for the environments to which they are exposed, therefore, they can not be relied on to accomplish their design basis safety function. If these valves were to fail to open during an accident, Residual Heat Removal pumps A and B would be unable to deliver the required flow to the reactor vessel.

The Stone & Webster valve data sheet for 1E12*F042A and B, Specification 228.212, incorrectly specified the location of these valves as Auxiliary Building outside containment. However, these valves are actually located inside the containment. The vendor supplied Limitorque motor actuators with Class B insulation and non-metallic actuator parts for outside containment applications in accordance with the design environment specified in Specification 228.212. Since the actual location of the valves is inside containment, the Class B insulation and non-metallic actuator parts are unacceptable for this application. An in-containment type actuator with qualified grease, seals, limit and torque switches, and Class H Type RH motor insulation is required.

The valves and motor actuators were procured, released, technically reviewed, and installed with this error undetected until October 13, 1983 (Reference 11). Specification 228.212 has been reviewed and design verified several times since 1974 (i.e., 9 addendums and 1 revision); the specification review process and design verification process did not uncover the data sheet error. Engineering Assurance Procedure EAP 4.13, paragraphs 3.2 and 3.2.3 require lead engineer and specialist review. Although these were accomplished, they failed to detect the subject error. Section 2.2 of Engineering Assurance Procedure EAP 3.1 requires an independent review. In particular, the reviewer must ascertain whether the specified parts and equipment are suitable for the required application

A5.3-1 (Detail) continued.

and are compatible with the design environment. Although the independent review was conducted, the review failed to detect the subject error.

Numerous other balance of plant and General Electric supplied motor operated valve actuators have been reviewed and found to be deficient with respect to qualification by the Stone & Webster Equipment Environmental Qualification Group. We determined that the Stone & Webster Equipment Qualification Group recently completed a generic review of Limitorque actuators procured for the River Bend Station (Reference 11). The objective of this review was to ensure that the installed actuators were qualified for the harsh environment defined by the Environmental Design Criteria. The Qualification Group found that certain Limitorque actuators were supplied with motors using a Class B insulation system which was not suitable for the environments specified. As a result of this review, Stone & Webster identified approximately 32 safety-related Limitorque motor operator valve actuators, which are located in the drywell, reactor building, or the Auxiliary Building, for which the Class B motors and actuators are not qualified for the accident environment to which they are exposed. It was recommended that the Class B motors be replaced with qualified Class H type RH motors. Stone & Webster also determined that 6 General Electric supplied Limitorque actuators have class B motors which are not qualified for the accident environment in the Auxiliary building. Action has been initiated to replace these General Electric supplied Limitorques. In addition, approximately 80 other Limitorque motor operated valve actuators with Class B motors located in the Auxiliary Building must be evaluated by Stone & Webster to determine their suitability with respect to post-accident radiation levels.

On March 31, 1984 (Reference 22), Stone & Webster completed a more detailed review of 247 Limitorque motor operated valve actuators associated with Specifications 228.212, -214, -216, -241 and -243. The generic Limitorque test reports (Reference 7) were reviewed and compared to composite environmental envelopes developed from the Environmental Design Criteria. The results of this review identified certain actuators that require: replacement of Class B motors with Class H type RH motors and inside containment type nonmetallic materials throughout the actuator; actuators that may require refurbishing for harsher environments; actuator motors that are not identified; and actuator motors that are not manufactured by Reliance Electric Company.

We concluded from our evaluation of Stone & Webster's review activities with respect to Limitorque motor operated valve actuators that the recent review process was well organized and controlled and that resolutions and actions were initiated to resolve the numerous identified problems in this area.

We determined that Stone & Webster engineers previously issued a change order to Specification 228.212 to procure qualified in-containment Limitorque motor operated valve actuators for 1E12*F042A and B. These units will replace the existing valve actuators. We also reviewed vendor telex messages (References 14 and 16) which acknowledge the technical details of this change.

During our review of safety-related motor operated valve actuators, we evaluated the Stone & Webster Equipment Qualification Group review comments on Limitorque Test Report B0058 and associated test reports. We reviewed Stone & Webster's review comments 6228-212-047-068A and -068B; the Limitorque response letter (Reference 19); and Stone & Webster comments 6228-212-047-068C. We also reviewed the Stone

A5.3-1 (Detail) continued.

& Webster equipment qualification checksheet for Limitorque actuators (Reference 21). We found that the Stone & Webster review comments were technically detailed and adequate.

We also reviewed the electrical design associated with 120 volt ac space heaters within the Limitorque motor actuators. We found that the design has unqualified Limitorque motor actuator space heater loads supplied from Class 1E power distribution panels in violation of industry and regulatory requirements for maintaining electrical and physical independence of redundant safety-related equipment (see Deficiency D5.3-1).

REFERENCES

1. River Bend Final Safety Analysis Report, Environmental Qualification Document (EQD), Amendment 12, 3-29-84.
2. NRC Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, 11-74.
3. IEEE STD 323, Qualifying Class 1E Equipment for Nuclear Power Generating Stations, 1974.
4. SWEC Environmental Design Criteria, 215.150, Rev. 2, 2-3-84.
5. SWEC RHR System Flow Diagram 12210-FSK-27-7E and 12210-FSK-27-7A, Rev. 8, 10/27/83.
6. SWEC Specification for Motor Operated Valves 2" and Larger, Specification No. 228.212, Rev. 1, 10/24/83.
7. Limitorque Test Report B0058 and associated documents SWEC #6228-212-0470-068A, 1/11/80.
8. SWEC RBP 2.1-2 Procedures for the Preparation and Maintenance of Procurement Specifications, Section 2.3.3., 1/15/82
9. SWEC EAP 4.13 Processing of Project Specifications Sections 3.2, 3.2.3, and 3.3.2., Rev. 0, 8/25/78.
10. SWEC EAP 3.1, Verification of Nuclear Power Plant Designs, Section 2.2, Rev. 2, 2/8/77.
11. SWEC Interoffice Memorandum to A. Blum from V. G. Deo, NSSS/BOP MOV Qualification Program, 10/13/83.
12. SWEC Environmental Qualification Data Master List, Page 20, 3-24-84.
13. SWEC Environmental Design Criteria Zone Map, Figure No. 3, Rev. 1, 8-4-83.
14. Velan Telex of J. J. Finkelstein to SWEC E. P. Grochouski, Replacement Limitorque Actuators with RH Insulation, 2-6-84.
15. SWEC Memorandum of Change to Specification 228.212, New Actuators for 1E12-MOV F042A, B, 3-20-84.
16. Velan telex of J. J. Finkelstein to SWEC R. Tate, RH Motor Actuator, 4-13-84.
17. SWEC Qualification Report Comments to Velan, Specification 228.216, P. Sandberg, SWEC # 6228-212-047-068A, 4-8-83.
18. SWEC Qualification Comments to Velan, Specification 228.216, D. Sandburg, SWEC # 6228-212-047-068B, 7-19-83.
19. Limitorque Corp. Letter of B. Kidd to R. J. McMorland (SWEC), Response to Qualification Comments, 10-10-83.
20. SWEC Qualification Comments, Specification 228.212, R. Smith, SWEC #6228-212-047-068C, 10-10-83.
21. SWEC Evaluation of Vendor Qualification Submittal, Review Checklist, SWEC # 6228-216-050-017A; 6228-214-059-010A; 6228-212-047-068A, D. Sandberg, 4-8-83.

A5.3-1 (Detail) continued.

22. SWEC Interoffice Memorandum, Review of Specifications 228.212, -241, -216, -241 and 243 to the Revised EDC Conditions in Accordance With PMM-110, R. J. Smith, 3-31-84.
23. USNRC NUREG-0588, Interim Staff Position on Environmental Qualfication of Safety-Related Electrical Equipment, Rev. 1, 7-81.

A5.4-1 (Detail) Cable Design and Analysis

The team selected 600 volt branch circuit power cabling (1RHSBBK017 and 18) from motor operated valve 1E12*F042B, located inside containment, to 480V motor control center 1EHS*MCC2K, located in the auxiliary building. We then reviewed Calculation E-137 to evaluate the analyses and calculations used by Stone & Webster to verify proper cable selection and cabling design, such as voltage drop, length, and rating. We obtained cable identification and interconnection data from the computer data base cable information reports EC-1 and EC-5 and the cable block diagram.

Calculation E-137 defines the criteria, assumptions, and analyses necessary to justify the selection of 600V power cable for use in 460V ac, 120V ac and 125 V dc applications. Principal cable criteria for the calculation were obtained from Industry Standard IPCEA-46-426 and project engineering memorandum ETG-IV-4-1. Calculation E-137 is based on cable allowable ampacity, short circuit and voltage drop limitations. We found one instance in calculation E-137 where an assumption regarding maximum cable current, based on thermal overload setting for motor operated valves, was incomplete. The documentation, rationale, and references for this assumption should be improved. However, this did not affect the validity of the calculation (see Deficiency D5.4-1).

The engineer assumed a short circuit current of 26500 amperes available at the 480V ac load center based on project engineering memorandum ETG-IV-4-1. We found that this assumption is conservative.

The engineer stated in the calculation an assumed fault clearing time of 0.02 seconds for circuit breaker protective trips based on "GE Breaker Curves TD-4999 and 8088" (References 3 and 4). However, we found that these circuit breaker time/current coordination curves are from Gould, not GE. In addition, they are not the correct curves for this application. However, the assumption of 0.02 seconds for circuit breaker clearing time remains conservative (see Deficiency D5.4-1).

Calculation E-137 shows the calculated minimum allowable cable size for a motor operated valve load such as 1E12*F042B (7.9 Horsepower, 95 locked rotor amperes, at 65°C) to be #6 AWG. The calculation also shows a calculated maximum allowable cable length to be 723 ft. for 70% starting voltage at the motor operated valve. The team examined the installed cable size and length for this application and compared this data to the allowable size and length derived from the calculation. The team determined from the EC-1 cable report that installed cable 1RHSBBK018 (cable from motor control center to penetration) is 2/0, 110ft. and installed cable 1RHSBBK017 (cable from penetration to motor operated valve) is #6 AWG, 150ft. We concluded that the installed cable size and length were in agreement with the calculation. In addition, the methods used in the calculation were judged to be technically correct.

REFERENCES

1. SWEC - Cable Block Diagram, 480V 1EHS*MCC2K, 12210-CBD-399-03-3J, Rev. J, 4-10-84.
2. SWEC - 600V Cable Sizing Calculation, 12210-E-137, Rev. 2, 3-1-84.
3. Gould Curve TD-4999 Sheets 1 & 2, Time Current Curves, HE Circuit Breaker, Rev. 3, 12-8-76.

A5.4-1 (Detail) continued.

4. Gould Curve TD-8088, Time Current Curves, HL and LL Circuit Breakers, Rev. 0, 12-9-76.
5. SWEC - Interoffice Memorandum ETG-IV-4-1, Cable Sizing Criteria.
6. SWEC - Cable Schedule EC-1, Issue 76, page 1850, 4-19-84.
7. SWEC - Cable Schedule EC-5, Issue 76, page 495, 4-19-84.

A5.5-1 (Detail) Cable Routing

The team selected cabling to a motor operated valve located inside the containment to evaluate the drawings and documents associated with the design cable routing. We selected 600 volt branch circuit power cabling from motor operated valve 1E12*F042B to 480 volt motor center 1EHS*MCC2K. The cable design route consisted of cable 1RHSBBK017 from the motor operated valve to electrical penetration cabinet 1RCP*TCR14A, and cable 1RHSBBK018, located in the auxiliary building, from penetration cabinet 1RCP* TCA14 to cubicle 3D of the motor control center.

We reviewed the following design drawings: cable block diagrams for power, control, and heater circuits to motor operated valve 1E12*F042B (References 1, 2, 3, and 4); low voltage power distribution (References 5, 6, and 9); wiring diagrams (References 7, 8, and 12); and elementary diagrams (References 10 and 11).

Stone & Webster uses a computer data base for cable tabulation, automatic or manual route points, and compilation of cabling and raceway data. The subject cabling was routed automatically by computer and the specific "from-to" information, conduit and tray coordinate points, and installation status was shown in the EC-38 computer sort. The EC-5 computer sort listed the subject equipment, equipment location, terminated cabling, and the Stone & Webster Mark Number for each cable.

We traced the subject cable routing on the following drawings using the raceway information provided in the EC-38 Report: Conduit Installation Drawing (Reference 13); Conduit Installation, Tray Identification, and Tray Arrangement Drawings (References 14, 15, and 16); and Cable Tray Arrangement and Identification Drawings (References 18 and 17). We found all design documents and drawings to be consistent and in good order.

We conducted a complete field walkdown of the subject cable route to compare the accuracy of the installed cable route to the design route. We observed two items of interest with respect to conduit routing and one deficiency with respect to cable routing within trays as discussed below.

We observed that the actual installed conduit (1CK500BB), located inside containment, for cable 1RHSBBK017 was routed to tray 1TK502B whereas the Conduit Installation and Tray Identification drawings (References 14 and 15) and the EC-38 report show that the conduit design routing is to tray 1TK500B. We subsequently determined that the conduit routing for 1CK500BB had been changed in the field to a preferred field run route. This route change was documented on an electrical Telecon/Note (Reference 23) and a "marked-up" Raceway Ticket (Reference 22) prepared at the time of the conduit installation. The Cable Pull Ticket for cable 1RHSBBK017 (Reference 25) was also "marked-up" to indicate the conduit routing change. The telecon (Reference 23) identified Engineering and Design Coordination Report's C-24063, C-23530, and C-24501 as authorization for this change; the team did not review these Engineering and Design Coordination Report's. We noted that the Raceway Ticket (Reference 22) was not signed off by the installer or the quality control group; we were told that the Raceway Ticket was unsigned because the conduit support installation was not completed. We determined that the Electrical Installation Specification 248.000, Section 3.2.2 allows cable to be installed in raceways prior to completion of the support installation. We concluded that this field developed change to the design conduit routing was performed in a controlled and documented manner. It is anticipated that Stone & Webster will revise the computer data base to reflect the as-installed

A5.5-1 (Detail) continued.

raceway and reissue the corrected cable pull and raceway tickets in accordance with established practice.

The team also discovered that conduit was field run and that as built drawings were not being maintained (see Unresolved U5.5-1).

During our field walkdown of the subject cabling we found that cable 1RHSBBK017 is required by design to run in tray 1TK501B (References 19 and 25), however, we observed that this cable was not installed in this tray (See Deficiency D5.5-1). The jacket of the installed cabling was marked:

Cable	Mark No.	Manufacturer	Type
1RHSBBK017	NGP-55	Okonite	#6 AWG, 600V, 1/C Triplex
1RHSBBK018	NGP-52	Okonite	2/0, 1/C, EP Power Triplex

Cable jacket marking data was compared to the EC-1 report, the cable procurement Specification (241.234), and the respective pull tickets (References 25 and 29) and was found to be in agreement. The EC-1 Report listed an estimated length for 1RHSBBK017 and -KO18 of 113 and 80 feet respectively. The pull tickets showed an actual pull length of 150 and 110 feet respectively. The estimated cable length and the actual pull length are not in reasonable agreement. However, we noted that the tie points for 1RHSBBK017 were changed as previously discussed.

The cable tray loading is tracked by computer data base. We randomly selected tray 1TL818B, which is a vertical tray above cubicle 3D of 1EHS*MCC2K, and compared the installed cabling to the cables identified in the EC-6 report and the tray percent fill. The EC-6 report was found to be in agreement with the installed configuration.

We also observed that the conduit and tray identification were properly identified and were physically separated from other separation groups as required by Stone & Webster separation criteria. All cable tray coordinate points had a one-to-one correspondence to the design drawings.

REFERENCES:

1. SWEC Cable Block Diagram, 480V One Line 1EHS*MCC2K, 12210-CBD-399-03-3J, Rev. 3, 4-10-84.
2. SWEC Cable Block Diagram, RHR, 12210-CBD-6RHS16-4A, Rev. 4, 8-18-83.
3. SWEC Cable Block Diagram, RHR, 12210-CBD-6RHS10-4A, Rev. 4A, 3-1-84.
4. SWEC Cable Block Diagram, 120V Control Circuit, 12210-CBD-7RHS01-1D, Rev. 1C, 8-10-83.
5. SWEC One Line Diagram, STBY Bus A and B Low Voltage Distribution System, 12210-EE-1ZC-1, Rev. 1, 6-23-83.
6. SWEC Power Distribution Panelboard Schedule, 12210-CBD-SCV2B1-2, Rev. M2, 9-20-83.
7. SWEC Wiring Diagram, Penetration Cabinet 1RCP*TCR15A and 1RCP*TCA15, 12210-EE-36BW3, Rev. 3, 1-22-84.
8. SWEC Wiring Diagram, Penetration Cabinet 1RCP*TCR14A and 1RCP*TCA14, 12210-EE-36BU-3, Rev. 3, 1-24-84.
9. SWEC 480V One Line, 1EHS*MCC2J and 2K Aux. Bldg., 12210-EE-1TE-2, Rev. 2, 8-17-83.

A5.5-1 (Detail) continued.

10. SWEC Elementary Diagram, 480V Control Circuit RHR, 12210-ESK-6RHS16, Rev. 4, 8-12-83.
11. SWEC Elementary Diagram RHR 120V ac, 12210-ESK-7RHS01, Rev. 1, 10-13-81.
12. SWEC Wiring Diagram 1EHS*MCC2K, Aux. Bldg., 12210-EE-9PG-2, Rev. 2 to be issued.
13. SWEC Seismic Conduit Installation, 114' Reactor Building, 12210-EE-460X-5, Rev. 5, 1-25-84.
14. SWEC Seismic Conduit Installation, 141' Reactor Building, 12210-EE-460AF-4, Rev. 4, 2-13-84.
15. SWEC Cable Tray Identification, Reactor Building, 12210-EE-34EB-4, Rev. 4, 3-11-83.
16. SWEC Cable Tray Arrangement, 141' Reactor Building, 12210-EE-34AC-5, Rev. 5, 6-22-83.
17. SWEC Cable Tray Identification, Auxiliary Building, 12210-EE-34FF-3, Rev. 3, 6-22-83.
18. SWEC Cable Tray Arrangement, Auxiliary Building 141', 12210-EE-34AQ-6, Rev. 6, 6-18-83.
19. SWEC EC-38 Report, Cable and Raceway Installation Status, Pages 4573 and 4574, Issue 76, 4-19-84.
20. SWEC EC-5 Report, Cable List, Page 495, Issue 76, 4-19-84.
21. SWEC Procedure PMM-152, Interoffice Memorandum, High Energy Line Break Evaluation Procedure, Rev. 2, 4-6-84.
22. SWEC Raceway Ticket 1CK500BB, Advance Copy, Issue 2A, Revised via mark-up on 3-16-84, 8-25-83.
23. SWEC Telecon - Electrical Telecon/Note, Raceway/Cable Deletion/Revision, J. Baldnsky, 3-18-84.
24. SWEC Specification for Electrical Installation, 248.000, Rev. 7, 12-30-83.
25. SWEC Cable Pull Ticket 1RHSBBK017, Advance Copy, Issue 2A, Revised via mark-up on 4-24-84, 4-19-84.
26. SWEC - Seismic Conduit Support Standard Details, Reactor Building, 12210-EE-450AJ-2 thru 450AN, Rev. 2, 2-20-83.
27. SWEC Seismic Conduit Installation, Reactor Building, 12210-EE-460A-3, Rev. 3, 11-14-83.
28. SWEC Cable Schedule EC-1, Issue 76, Page 1850, 4-19-84.
29. SWEC Cable Pull Ticket 1RHSBBK018, Installed 12-15-83, Issue 5-5-83.
30. SWEC Specification for Insulated 600V cable 241.234, Attachment #2.
31. SWEC EC-6 Raceway Section List, Issue 82, Page 2287, 5-3-84.

A5.6-1 (Detail) Containment Electrical Penetrations and Fault Currents

The team reviewed calculation E-107 performed by Stone & Webster for short circuit current determination on circuits connected to containment electrical penetration assemblies. We selected a class 1E circuit at random and followed the circuit through the calculations and design documents. The circuit selected was 480V branch circuit power cabling from motor operated valve 1E12*F042B, located inside containment, to cubicle 3D of motor control center 1EHS*MCC2K, located in the Auxiliary Building. The circuit includes Conax containment electrical penetration assemblies 1RCP*TCR14A and TCA14. We reviewed the circuit design shown on motor control center 1EHS*MCC2K and load center 1EJS*LDC1B and 2B one-line diagrams.

To obtain maximum fault current available a fault located just inside the containment at the penetration was postulated. The fault current for each conductor size was calculated. The responsible engineer used the following assumptions: maximum rating of the supply transformers; cable length from the load center to the motor control center; cable length from the motor control center to the fault; 90° C conductor temperature; and primary and backup interrupt protection clearing times. The engineer also assumed that cables are sized to withstand short circuits equal to the primary clearing time, however the penetration feedthroughs must withstand fault currents for a duration equal to the back-up clearing time. We observed that back-up protection for the subject motor operated valve is provided by a 30 ampere fuse located at the motor control center cubicle. The Stone & Webster Boston Office made available computer program Wang EL-061 for this calculation. The responsible engineer used this program to calculate respective fault currents for calculation E-107.

We reviewed calculation E-107 which showed the principal data used to obtain fault current values associated with class 1E 480V ac motor control center loads. We determined that the switchgear rating was correct by review of Gould technical data provided in Specification 242.521. However, we found by inspection of one-line diagram EE-IAB-5 and the Powell technical data provided in load center specification 242.533 that the load center transformer rating (1000 KVA) and impedance ratio (5.8%) used in the calculation was incorrect. The correct transformer rating was 1500 KVA and the impedance ratio was 8.63%.

The results of calculation E-107 indicated that a 14,338 root mean square equivalent ampere fault current would be available at the penetration for a size 2/0 cable. We selected this size cable for comparison to our sample cable (RHSBBK018). We determined from the EC-1 Cable Schedule that cable 1RHSBBK018 is 2/0 triplex, 110' in length. This fault current is based on a back-up clearing time duration equal to 10 cycles. We reviewed the calculation results which show the calculated fault current for each conductor size and compared this data to the fault current data provided to the penetration manufacturer in Table 1 of specification 241.211. We found the data provided in specification 241.211 was consistent with the results of calculation E-107.

As mentioned above, the 480V load center transformer rating and impedance assumptions for calculation E-107 are incorrect, therefore, the calculation results are in error. The calculation was performed in 1980 in order to provide design data and short circuit current values to Conax for procurement of containment electrical penetrations via specification 241.211. We learned that it was Stone & Webster's intent to revise calculation E-107 to account for receipt of actual vendor data on load centers and other equipment and to factor in the actual installed cable lengths. However, Stone & Webster subsequently determined that several problems existed in the computer program EL-061 used to develop the results. We therefore focused our review on this problem area.

A5.6-1 (Detail) continued.

In January, 1982 the Stone & Webster Boston Office issued Interoffice Memorandum PR-E-35 which notified all projects that Wang computer program EL-061 was found to be in error with respect to calculated values of short circuit current and final cable temperatures. The program was inactivated and projects were advised to revise their fault current calculations and penetration specifications. On Janaury 27, 1982 (Reference 18) the River Bend Project replied to Memorandum PR-E-35. The River Bend response stated that: calculation E-107 used the subject computer program; a preliminary review of Conax data indicated margins in the specified fault current values; the Conax design compensates for errors in the computer program, however, a verification is needed; calculation E-107 will be recalculated; and specification 241.211 will be revised. The team found that this response was based on engineering judgement and not on a documented review. We learned that the engineer compared the maximum current values of the penetration test indicated in Table 5.9.2 of Conax Report IPS 589.2 to the calculated fault current in calculation E-107 and concluded that adequate margin existed. We reviewed the test data in Table 5.9.2 of IPS 589.2 and compared it to the results of calculation E-107 and concluded that there appeared to be adequate margin (i.e., for #2 AWG. conductors IPS 589.2 showed a test level of 21,800 amperes, whereas calculation E-107 indicates a calculated fault current of 13,882 amperes).

The Stone & Webster Engineerng Assurance Problem Tracking System continues to identify calculation E-107 as an open item that requires recalculation and verification (References 8, 9 and 19). Calculation E-107 had not been revised as of the date of our inspection.

Stone & Webster started calculation E-162 in 1982 in an attempt to replace calculation E-107. E-162 was based on an IBM computer program. Subsequently, Stone & Webster decided to recalculate penetration fault currents by hand calculation since this IBM program was not verified; therefore, E-162 was not issued. We reviewed calculation E-162 and observed that the fault current results for size 2/0 conductors were comparable to the results obtained in calculation E-107 (i.e., for 2/0 conductors, calculation E-162 fault current was calculated to be 13,454 amperes and calculation E-107 fault current was calculated to be 14,338 amperes). We found that the methodology used in calculation E-162 was acceptable.

We reviewed the present activities of Stone & Webster's electrical technical support group with respect to containment electrical penetration fault current calculations. Stone & Webster has completed samples of time/current thermal capacity and protective trip coordination curves to demonstrate the adequacy of the electrical design and assure that the penetrations are properly protected. Curves have been completed for a sample load center, 5KV switchgear, a motor control center and a 120V distribution panel. Stone & Webster intends to incorporate these curves into the fault current calculation which must be completed. They also intend to use the corporate cable impedance tables in ETP-104.3.1-0, obtain actual loads from the one-line diagrams and electrical load list, and use the Electrical Cable Schedule Information System cable reports to obtain installed cable lengths.

We reviewed sample figure 8.3-13B, thermal capacity and protection coordination curve, for motor center 1EHS*MCC2K. The typical load indicated on the one-line sketch was motor operated valve 1E12*F009. The sketch showed the Gould A80 series type J10 circuit breaker trip coil, the Gould 30 ampere fuse used for back-up protection, the penetration, a calculated fault current of 11,600 amperes, and the 8 horsepower motor operated valve load. We found that the horsepower, locked rotor and full load

A5.6-1 (Detail) continued.

current shown on the curve were consistent with the data in the motor and load list. We reviewed the manufacturer's data which showed: the Gould class RK5 fuse melting time/current curves; the Gould A80 series instantaneous time/current trip curves; and the Conax current thermal capacity curves for each penetration conductor size. We found that the Stone & Webster sample thermal capacity and protection coordination curves clearly show that the fault current verses time conditions are well within the circuit breaker trip rating (the primary protection) and the circuit breaker will trip earlier than the maximum time for which the penetrations are qualified. The curve also shows that the fuse provides back-up protection. We concluded that the design is consistent with Section 8.3.1.1.4.3 of the Final Safety Analysis Report which requires the design to withstand the maximum short circuit current and requires that primary and back-up protection be provided.

We had no further questions in this area.

REFERENCES

1. SWEC Specification for Electrical Penetrations 241.211, Rev. 1, 4-17-81, Addendum 2, 6-20-83.
2. Conax Design Qualification Report IPS 589.2, Low Voltage Power and Control Electrical Penetration Assemblies, 4-28-81, SWEC # 6241.211-156-003B.
3. SWEC 480V One-Line Diagram 1EHS*MCC2J and 2K, 12210-EE-ITE-2, Rev. 2, 8-17-83.
4. SWEC 480V One-Line Diagram 12210-EE-IAB-5, 1EJS*LDC1B and 2B, Rev. 5, 11-10-82.
5. SWEC Calculation 12210-E-107, Electrical Penetration Fault Current, Rev. 0, 4-25-80.
6. SWEC Specification for Stby Load Center 242.533, Rev. 1, 3-18-83.
7. SWEC Interoffice Memorandum, PR-E-35, Wang Programs EL-061, 1-22-82.
8. SWEC Engineering Assurance Problem Report G15.16, Subject PR-E-35.
9. SWEC Interface Correspondence, Specification 241.211, S. DiJoseph to P. Guha/J. Gelston, 2-23-84.
10. SWEC Calculation 12210-E-162, Electrical Penetration Short Circuit, Preliminary.
11. SWEC Figure 8.3-13B, Coordination Curve Penetration Protection, Preliminary.
12. Conax Thermal Capacity Curves 12t, Electrical Penetration Assemblies and Conductor Seal Assemblies, IPS -701, Rev. A, 7-16-81, SWEC #6241-211-156-024A.
13. Gould Electrical Fuse Division, TRI-ONIC Dual Element Fuse, Class RK5, Melting Time/Current Data, SWEC # 7242.562-082-001A.
14. Gould Unitized Circuit Breaker Curve, Time/Current, SWEC # 7242-561-081-005A, 6-8-82.
15. Computer Program on Containment Electrical Penetration Fault Current Calculations, Wang EL-061, Program #5564, #6403, #8904, #8695, Including Verification program, SWEC Boston Office.
16. SWEC Specification for 4.16KV Metal Clad Switchgear and 125V dc Switchgear, 242.521, Addendum 2, 1-29-82.
17. SWEC Cable Schedule EC-1, Issue 76, page 1850, 4-19-84.
18. SWEC Memorandum, J. Kirby (SWEC) to D. Shelton (SWEC), Response to Interoffice Memorandum PR-E-35, 1-27-82.
19. SWEC Boston Memorandum to SWEC CHOC, Subject: PR-E-35, 8-9-82.
20. SWEC Electrical Technical Procedure ETP-104.3.1-0, Cable Impedances for 90C Conductor Temperature, 9-16-82.
21. SWEC Electrical Motor and Load List, page 44, 12-14-83.

A5.7-1 (Detail) Motor Control Centers

The team reviewed the Stone & Webster design with regard to application of thermal overload protection for motor operated valves. These devices are located at the motor control center and are an integral part of the combination starter circuit breaker. Final Safety Analysis Report table 1.8-1 and Section 8.3.1 require the design to be in conformance with Regulatory Guide 1.106.

On River Bend Class 1E motor operated valves, thermal overloads are either automatically bypassed under accident conditions, bypassed continuously, or unbypassed. We determined that Stone & Webster has recently reviewed the application of thermal overload, limit switch, and torque protection for all motor operated valves (Reference 4). As a result of that review, Stone & Webster has incorporated the following thermal overload protection design for class 1E motor operated valves. Valves which have spring return control switches or valves located outside the drywell which have maintain contact control switches for throttling applications have their thermal overloads automatically bypassed under design basis accident conditions. Valves located inside the drywell which have maintain contact control switches for throttling applications have thermal overloads which can be bypassed manually via the control switch.

As a basis for our review of calculation E-164-3 , we selected Low Pressure Coolant injection valve 1E12*F042B which is fed from cubicle 3D of motor control center 1EHS*MCC2K. Elementary diagram ESK-6RHS16 indicates that the thermal overload for 1E12*F042B is normally in the circuit but is automatically bypassed in the opening circuit during design basis accident conditions. We found this design in compliance with Final Safety Analysis Report requirements. We determined from the vendor wiring diagram (Reference 5) that the subject motor operated valve load data is: 7.9 horsepower, 12.0 full load amperes, and 95.0 locked rotor amperes. We found that the motor and electrical load list contained the correct load data for 1E12*F042B.

We determined that Stone & Webster policy, as presented in ETG-V-2-3, set motor operated valve thermal overload trip settings at 45%-55% of locked rotor amperes to avoid nuisance trips. However, a recent engineering division memorandum EDVM 83/18-1 requires that class 1E motor operated valves which have thermal overloads bypassed under accident conditions should have overloads set at 125%-140% of full load amperes. For class 1E motor operated valves with overloads not bypassed, the recommendations of ETG-V-2-3 still apply.

We found that calculation E-164-3 was performed and documented in a manner consistent with the requirements of Engineering Assurance Procedure EAP 5.3. We also found that the methods were in agreement with the requirements of EDVM 83/18-1.

Attachment 4, Table 1 of calculation E-164-3 presents the manufacturer's (Gould) thermal overload relay selection chart. We determined from this chart that a Gould type G30T41 overload heater is required for a motor with a range of 12.4-13.8 full load amperes. Since the Gould overload trip setting is set at 125% of the minimum full load amperes value within the identified range on the chart, the engineer performed a calculation to proportionally lower each full load amperes heater range to achieve an overload trip setting of 125%-140%. The calculation shows that a Gould type G30T41 overload heater is to be used for a motor with a 11.3-12.4 full load amperes range; the resultant trip occurs at 15.5 amperes or 137% of the minimum full load amperes value. Since the load data for 1E12*F042B indicated 12 full load amperes, calculation E-164-3 requires that a Gould type G30T41 overload heater be used; thus providing a 129% trip setting in this application.

A5.7-1 (Detail) continued.

We reviewed the Gould industrial control catalog and found that data used in calculation E-164-3 was in agreement with the Gould data.

Overload relays are manufactured to be used with continuous duty motors; however, the Limitorque motors are generally rated for 15 minute duty. If the rotor is locked for greater than 10 seconds, motor damage can occur. Since the relay will carry 115% of rated current continuously, the overload relay must operate at 700% overload in order to open the circuit within 10 seconds. We determined from Gould curve P.C. No. 409399-1 that the G30 class device will provide a trip in 10 seconds at approximately 700% overload.

The team was impressed with Stone & Webster knowledge of the issues and detailed study of this subject area. We concluded that the calculation was in good order.

The Gould thermal overloads were procured by the Site Engineering Group and installed by the preliminary test organization. Selection is based on the overload chart provided in Table 1, Appendix F of the Electrical Installation Specification 248.000. We found this chart to be consistent with the data presented in calculation E-164-3. We also found during our site visit that a Gould type G30T42 thermal overload was installed for 1E12*F042B based on an actual motor nameplate rating of 12.5 full load amperes. We found this selection to be consistent with good engineering practice and requirements presented in the installation specification.

We reviewed the site activities with respect to procurement, installation and testing of the Gould type G30T42 thermal overload for 1E12*F042B. Non-engineered Item Data sheet 1051 specifies that spare and replacements parts be designed and manufactured as exact duplicates and under the same Quality Assurance program as equipment supplied via the motor control center specification 242.562. A Certificate of Compliance attesting to these requirements is required. We reviewed: Field Purchase Requisition 431631 and Purchase Order 12210-23077 which procured the subject overloads and other spare parts; Quality Assurance Inspection Report IR-E3100441 Material Receiving Report and Material Certification Checklist; and Gould Statement of Conformance and Report of Inspection and Test. We found these documents to be in agreement with the requirements.

We also reviewed the installation, inspection and test record Boundary Inspection Program No. RHS.000 for cubicle 3D of motor control center 1EHS*MCC2K. This record showed that the correct overload was installed (type G30T42) and was properly tested. We also reviewed generic test procedure 1G-EE-29 for Gould unitized combination starters. We found this document to be detailed and technically adequate.

We concluded that this area was in good order.

REFERENCES

1. Final Safety Analysis Report, Thermal Overload Protection for Motor Operated Valves, FSAR Table 1.8-1 and Section 8.3.1.
2. SWEC Elementary Diagram, 480V Control Circuit, RHR System, 12210-ESK-6RHS16 Rev. 4, 8-12-83.
3. USNRC Regulatory Guide 1.106, Thermal Overload Protection for Electric Motors on Motor - Operated Valves, Rev. 1, 3-77.

A5.7-1 (Detail) continued.

4. Letter of J. A. Kirekebo (SWEC) to W. J. Cahill (GSU), Valves - Thermal Overload Protection, RBS-8749, 7-14-83.
5. Limitorque Wiring Diagram 15-477-4071-3, SWEC #0228-212-047-052A, Rev. 1, 11-6-79.
6. SWEC Motor and Electrical Load List, 12-14-83.
7. SWEC Interoffice Memorandum ETG-V-2-3, Selection of Motor Running Overcurrent and Locked Rotor Protection, ac dc Motors, G. Barney, 1-10-75.
8. SWEC Engineering Division Memorandum EDVM CHOC 83/18-1, Clarification of ETG-V-3, J. C. Gabriel, 7-7-83.
9. SWEC Specification for Electrical Installation 248.000, Rev. 7, 12-30-83.
10. Gould Industrial Control Catalog, Controlfax 1982, Page 146, Class G30 Heater Units.
11. Gould Curve P.C.NO. 409399-1, Multiples of Overload Relay Current Element Rating.
12. SWEC Engineering Assurance Procedure EAP 5.3, Preparation and Control of Manual and Computerized Calculations, Rev. 3, 1-31-79.
13. SWEC Calculation No. E-164-3, Procedure for Selecting Trip Coils and Motor Overload Heater for 120V ac, 460V ac and 125V dc Normal and Safety Class Motors and MOVs Fed from Gould MCC and Local Starters, Rev. 3, 2-29-84.
14. SWEC Nonengineered Item Data Sheet, Electrical, Specification 211.161, Page 1051, Rev. 3, 3-12-84.
15. SWEC Field Purchase Requisition 431631, Page 8 of 13, Item 63, 6-20-83.
16. SWEC Purchase Order 12210-23077 to Gould.
17. Preliminary Test Organization, BIP No. RHS000, 1EHS*MCC2K/1E12*MOVFO42B, Unitized Combination Starter Inspection and Test Record, Enclosure 11.1, 3-15-84.
18. SWEC QA Inspection Report Type-A, IR-E3100441, for P.O. #23077, NEI R1051, 8-6-83.
19. SWEC Material Receiving Report MRR #83-1140, Page 5 of 8, Item 63.
20. SWEC Material Certification Checklist, for MRR 83-11440, 8-6-83.
21. Gould Statement of Conformance, for MRR #83-11440, Specification 211.161, 7-21-83.
22. Gould Report of Inspection and Test, P.O.12210-23077 Item 63, 7-21-83.
23. GSU Generic Test Procedure 1-G-EE-29, Unitized Combination Starter Test Procedure, Rev. 5, 4-13-84.
24. Gould Drawing 84-51380-23, Rev. D, 2-6-84, Stby MCC 1EHS*MCC2K, SWEC #0242-562-082-113G, 2-15-84.

A5.8-1 (Detail) Power Generation Control Complex Design Changes

The main control room, designated as the Power Generation Control Complex, is manufactured by General Electric. The Power Generation Control Complex arrangement, circuitry, and configuration is a product of the design efforts of both General Electric and Stone & Webster. General Electric has design responsibility for the Nuclear Steam Supply System and Stone & Webster has design responsibility for the Balance of Plant area. Due to the complexity of the Power Generation Control Complex design and the numerous interfaces involved, we focused our efforts on design changes to determine whether the process was conducted in a controlled manner.

Initial design efforts on Power Generation Control Complex arrangement were conducted in 1976. At that time, General Electric issued standard plant panel from view arrangement drawings (Reference 1), and Stone & Webster constructed a front view full-scale mockup of Balance of Plant panels 1H13*P870, P877, P808, P863, and P601. Initial Balance of Plant design details, based on meeting with Gulf States, were then transmitted to General Electric (Reference 2). The subsequent Balance of Plant design effort included development of: panel outline drawings, equipment lists, nameplate lists, elementary and analog diagrams, line coding, and issuing various Balance of Plant vendor drawings. The entire Power Generation Control Complex was fabricated at the General Electric's Valley Forge facilities during the 1979-1982 time period. The Power Generation Control Complex floor sections, panels, and termination cabinets were shipped to General's Electric San Jose facilities for testing. The Power Generation Control Complex was installed at the River Bend site in July, 1982. During this time period, Stone & Webster issued one major Delta Change Package (Reference 3) which was incorporated into the design before Power Generation Control Complex shipment, and three major Field Change Packages were incorporated into the design at the site. Subsequent to the issuance of the major change packages, Stone & Webster has completed approximately 50 Change Request forms which detail Power Generation Control Complex modifications initiated individually on a case-by-case basis via Field Deviation Disposition Request and Engineering and Design Coordination Report. At present design changes are few in number. Gulf States has contracted General Electric to maintain Power Generation Control Complex configuration control and maintain the production design base. Consequently, a General Electric site engineering group was established in January, 1984 to control revisions to the Power Generation Control Complex.

We randomly selected and reviewed several Power Generation Control Complex design changes associated with safety-related systems and circuitry.

We reviewed Work Package Change Notice WPCN-786 which deals with implementation of Regulatory Guide 1.97 Revision 2, Post-accident Monitoring Instrumentation. Modification No. 1 of the Work Package Change Notice WPCN-786 was randomly selected for review. This change requires addition of safety-related temperature sensors in the discharge line of the Residual Heat Removal heat exchangers. We reviewed the Stone & Webster project position on conformance to Regulatory Guide 1.97 with respect to these temperature sensors and the Stone & Webster request to Gulf States for implementation of this change (References 18, 19 and 20). Approval for the Work Package Change was provided by Gulf States via letter RBG-14,981. Final Safety Analysis Report Table 7.5-2 identifies the Residual Heat Removal heat exchanger outlet temperature as a type D Category 2 variable.

In order to obtain approval for a change to the Power Generation Control Complex portion of the design, project procedures require that a Change Request Form be developed to allow the project Power Generation Control Complex Task Force to

A5.8-1 (Detail) continued.

determine the necessity and feasibility of the proposed change. We therefore reviewed Change Request Form CF-0300 which outlined the modification required by Work Package Change Notice WPCN-786 to add temperature recorders 1RHS*TR47A and B and all associated wiring to divisional inserts on Power Generation Control Complex panel 1H13*P601. We observed that the change request form listed the affected documents which required change and specified that the recorders are required to be Quality Assurance Category I devices.

We determined that the Stone & Webster diagrams for Residual Heat Removal system were revised to indicate the addition of the heat exchanger outlet temperature sensors. However, the General Electric flow diagrams (References 26 and 27) and the system design freeze punchlist did not reflect this design change. Stone & Webster maintained that the General Electric flow diagrams and the design freeze punchlist only require revision for significant Balance of Plant changes which General Electric should be appraised of. We then reviewed the following documents associated with Change Request Form CF-0300: Transmittal Package to General Electric San Jose; and Transmittal Package to the Site; Loop Diagram 1RHS*47; Drawing/Document Index IS-256; Analog Wiring Diagrams AWD-27-7.4 and AWD-27-7.5; Equipment List ESK-4CES601; Cable Block Diagram CBD-RHS400-1A; Electrical Cable Schedule Information System File Source Document; Analog Wiring Diagrams AWD-1-1.67 and AWD-1-1.66; and General Electric Field/System Interface Termination Summary Report for 1H13*P702D and P703D. We found that all documents were revised correctly with the exception of an error on analog wiring diagram AWD-1-1.67 (see Deficiency D5.8-2). This item had no effect on design or analysis and the error was considered to be a minor documentation discrepancy.

In addition to development of Change Request Form CF-0300, Project Procedures require that other associated documents be issued. A 5000 series Change Request Form is required for changes external to the Power Generation Control Complex to notify affected disciplines and obtain management approval. A Boundary Inspection Program Support Plan is also required to inform affected disciplines that changes will be made to the respective system. Finally, Engineering and Design Coordination Report's are issued as the vehicle to accomplish the actual field modification work at the site. We therefore reviewed these documents which are associated with Change Request Form CF-0300.

We reviewed Change Request Form CF-5068 which outlines modifications external to Power Generation Control Complex required by Work Package Change Notice WPCN-786. This document addresses the addition of temperature transmitter and converter modules and internal wiring to local panels ICES*PN6A and 6B for the Residual Heat Removal heat exchanger outlet temperature sensors. We reviewed the following revised documents associated with Change Request Form CF-5068: equipment list ESK-4CES191 and 192; nameplate engraving list ESK-4CES291 and 292; Electro Mechanics panel assembly drawings 40490 and 40480; and Electro-Mechanics panel wiring drawings 40491B and 40471B. We found that the revisions had been correctly incorporated onto the drawings.

We then reviewed Boundary Inspection Program support Plan RHS .000-10 and a subsequent revision to this plan, RHS .000-15. We determined that these plans correctly listed the affected drawings and also identified that an Engineering and Design Coordination Report must be developed and temperature sensors must be procured via Specification 247.461. We reviewed Specification 247.461 and found that safety-related resistance temperature detectors were procured as required by the design. We also reviewed Engineering and Design Coordination Reports P40286B and P40575 which instructed the field to install temperature transmitters and converter cards within specific card frames on panels 1CES*PNL6A and 6B and install respective nameplates. We found these documents to be in good order. We reviewed Engineering and Design Coordination Report No. P21294; which showed sketches of the required cable terminations for the Residual Heat Removal heat exchanger outlet temperature channels. We found that the document correctly reflected the design change. We then reviewed the cable external connection drawings EE-7CQ, -7CK, -3LQ, and -3FA and determined that these drawings were not yet updated to show the additional cabling. However, we found that these drawings were tracked in the change control document reporting system with Engineering and Design Coordination Report No. P21294 identified as the required change against the drawing.

In order to accomplish modifications to panels, circuits or equipment within Power Generation Control Complex, the Stone & Webster Change Request Form (CF-0300 in this case) is issued to General Electric, San Jose and Gulf States at the River Bend site. A Field Deviation Disposition Request Form is then developed by either Gulf States or General Electric site engineers. The Field Deviation Disposition Request provides the suggested disposition of the required change identified in the Change Request Form and on the Stone & Webster drawing package. The Field Deviation Disposition Request normally includes marked up drawings and sketches (i.e.., assembly and wiring drawings, device lists, Field/System Interface Termination Summary cable schedules, etc.) reflecting the detailed revisions required to accomplish the change. After General Electric San Jose approval of the Field Deviation Disposition Request, an Engineering Change Notice (ECN) and revised drawings reflecting the required changes are issued by General Electric. Upon receipt of the approved Field Deviation Disposition Request, an Engineering and Design Coordination Report is issued to the field by Stone & Webster Cherry Hill to accomplish the work under the electrical installation specification 248.000. We therefore reviewed Field Deviation Disposition Request LDI-1443 which reflects the detailed changes required to incorporate the addition of Residual Heat exchanger outlet temperature recorders 1RHS*TR47A and B and associated wiring to panel 1H13-P601 as required by Change Request Form CF-0300.

Field Deviation Disposition Request LDI-1443 included marked up drawings and documents which reflected the detailed implementation of Change Request Form CF-0300. We reviewed the marked up panel assembly drawings, parts lists, Field/System Interface Termination Summary cable lists, wire lists, engraving list, and wiring connection drawings. We found these drawings and documents to be in good order. We also reviewed the panel wire list 287A6401 and 287A6416 for Power Generation Control Complex floor sections U702 and U732, respectively, and found that the Stone & Webster recommended terminations at Power Generation Control Complex termination cabinets were implemented by General Electric.

We noted during our review of Field Deviation Disposition Request LDI-1443 that Stone & Webster Cherry Hill did not indicate in Change Request Form CF-0300 the suggested location of temperature recorders 1RHS*TR47A and B on panel inserts 1H13-P601-20B and -17B (respectively). We found that the panel arrangement drawings, 914E644 RHR Division II insert 17B and 914E647 RHR Division I insert 20B, were not marked up by Stone & Webster to indicate the approximate position of the temperature recorder on the panel inserts. Our concern was that this apparent lack of detail could

A5.8-1 (Detail) continued.

lead to a design process wherein the controls engineer selected a post-accident temperature recorder as a receiver and integrated this equipment into the design; however, the recorder may not fit on the selected panel insert due to existing equipment and panel bracing density. We believe that panel arrangement drawings 914E644 and 914647 should have been marked up with the suggested location of the temperature recorders, thereby substantiating that the design was indeed feasible. Stone & Webster stated that the panel inserts were routinely checked by the engineer to determine feasibility of locating the recorders on the inserts; however, it is Stone & Webster policy to allow the field group to actually "spot" the location of the devices on the panels via the issuance of the Field Deviation Disposition Request. Stone & Webster also has final approval for construction via the Engineering and Design Coordination Report.

We reviewed Field Deviation Disposition Request Field Deviation Disposition Request LDI-925, revision 0 and 1. We found that Stone & Webster inadvertently missed incorporation of changes on the elementaries in two cases (see Deficiency D5.8-1). This item had no affect on the design and we considered this to be a minor error.

We randomly selected Reactor Pressure Vessel Head Transmitter 1E31*N092 and reviewed the design documents and the field cabling terminations at the Power Generation Control Complex termination cabinet. General Electric Elementary Diagram 828E535AA sheet 10 establishes the circuit design. We reviewed the General Electric Field/System Interface Termination Summary cable report, Block Diagram Source Document, cable Block Diagram, Connection Wiring Diagram, and EC-1 and EC-5 Cable reports (References 12 thru 17) and found these documents to be consistent with the design as shown on the elementaries.

The team also reviewed Stone & Webster Change Request Form CF-0288. This change modified the design of existing control switches on Power Generation Control Complex panel H13*P870 by incorporating 2-position "spring-return-to-center" control switches to accomplish throttling control for Residual Heat Removal heat exchanger Service Water discharge valves. We reviewed the following documents associated with Change Request Form CF-0288: Gulf States approval letter for Change Request Form CF-0288, RBG-16533; transmittal package RBV-R-2384; equipment lists ESK-4CES154 and 155, control switch contact diagram ESK-3D and 3K; elementary diagram ESK-6SWP38; and panel outline drawings ESK-4CES54 and 55. We found that these documents were revised correctly to incorporate the required change.

We also reviewed Field Deviation Disposition Request LDI-1331 which includes marked up drawings and documents reflecting the detailed modification required to incorporate the control switch changes described in Change Request Form CF-0288. We found several errors on sketches and marked up drawings attached to Field Deviation Disposition Request LDI-1331 (see Deficiency D5.8-3). These errors appeared to have no impact on design. We reviewed Engineering and Design Coordination Report C-51164 which was issued to the field to accomplish this modification. We found the change to be in good order. Stone & Webster maintains a change control document reporting system designated as Report 217. Report 217 correctly indicated that Engineering and Design Coordination Report 51164 was issued to incorporate modifications detailed in Field Deviation Disposition Request LDI-1331.

During the course of our review, we observed that the electrical design group maintained a controlled file of design drawings such as connection wiring diagrams. Similarly, the electrical engineering group maintained a controlled file of elementary and analog

A5.8-1 (Detail) continued.

wiring diagrams. Both groups had uncontrolled copies of vendor prints. However, the Document Control Center is used as the main source of the latest vendor drawings issued for the project. The team questioned Stone & Webster concerning the use of uncontrolled drawings. We were told that the engineers and designers checked the drawing index to ensure they had the latest revision prior to using vendor drawings. The team had no further questions.

During our site visit, we reviewed the Power Generation Control Complex design modification activities of the General Electric site engineering group.

We again selected Field Deviation Disposition Request LDI-1443, addition of heat exchanger outlet temperature recorders, to evaluate methods of document tracking and to determine General Electric's activities with respect to this modification. We observed that transmittal letter GES-4654/84 issued this to the Stone & Webster site organization. We also determined that General Electric maintains an Engineering Information System computer data base which tracks assigned Engineering Change Notices against each design drawing and drawing status. We found that General Electric had not yet completed the Engineering Change Notices incorporating Field Deviation Disposition Request LDI-1443 into the design documents. However, General Electric Report D04 correctly indicated that Engineering Change Notice ECN-LDI-1443 was listed against each design drawing required to be revised in accordance with the Field Deviation Disposition Request. General Electric Report D03 also correctly indicated the subject Engineering Change Notice against each design drawing associated with this Field Deviation Disposition Request. We also found that General Electric had not yet completed the Engineering Change Notice incorporating the change into the design documents. However, General Electric D04 Report correctly indicated that Engineering Change Notice LDI-1331 was listed against each design drawing required to be revised in accordance with the Field Deviation Disposition Request. The team was impressed with General Electric's methods of tracking the Power Generation Control Complex production design base documents and changes to the design base.

We randomly selected a Power Generation Control Complex design change and reviewed the General Electric site engineering group's activities with respect to this change. We selected Stone & Webster Change Request Form CF-0263 which included revisions to safety-related and non-safety related circuitry in several systems. We focused our review on the safety-related circuitry changes associated with the Fuel Handling Building Ventilation System. This change involved the grounding of signal cable shields to the chassis ground bus for dual output temperature converters 1HVF*TY1A and B and revising the signal input notation at the signal resistor units. This circuitry is located in Power Generation Control Complex panels 1H13-P841 and 1H13-P842. We reviewed the analog wiring diagrams AWD-22-6.3, -6.4, -6.6, -6.7, and -6.8, and Signal Resistor unit drawings AWD-1-1.40 and -1.42. We found that these drawings were correctly revised by Stone & Webster to incorporate the required changes. We then reviewed General Electric Field Deviation Disposition Request LDI-1460 which was developed in response to Change Request Form CF-0263. This Field Deviation Disposition Request was approved by General Electric San Jose. The Field Deviation Disposition Request listed Engineering Change Notices NJ54867, -868, -869, and NJ55085 which incorporated drawing changes associated with the change. We then reviewed the following drawings and documents associated with Field Deviation Disposition Request LDI-1460: Connection wiring design 914E577 and 914E578; General Electric

A5.8-1 (Detail) continued.

Reports D03 and D04; Engineering Change Notices ECN-NJ54867; and ECN-NJ54868. We found that these documents were revised correctly to incorporate the required change.

We had no further questions in this area.

REFERENCES

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18. Letter of J. Kirkebo (SWEC) to W. Cahill (GSU), RG 1.97, Rev. 2 , RBS-7062, 11-13-81.
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22. Letter of T. Crouse (GSU) to B. G. Schultz (SWEC), Approval of WPCN-786, RBG-14,981, 5-9-83.
23. SWEC Change Request Form CF-0300, 11-9-83.
24. SWEC System Flow Diagram, RHR Loop A, 1221[illegible]-27-7G, Rev. 9, 4-17-84.
25. SWEC System Flow Diagram, RHR Loop B, [illegible]-27-7C, Rev. 9, 4-5-84.
26. GE P&ID, RHR System, 762E424AA Loop A, [illegible] 5, SWEC # 0221-434-000-012E.
27. GE P&ID, RHR System, 762E424AA Loop B, Sh. 3, Rev. 5, SWEC # 0221-434-000-014E.
28. SWEC System Design Freeze Punchlist, RHR System E12, RBV-2063, 4-18-84.
29. SWEC Control Loop Diagram 1RHS*47, Issue 2, 11-1-83.
30. SWEC Drawing/Document Index IS-256, 4-2-84.
31. SWEC Analog Wiring Diagram Loop B, 12210-AWD-27-7.4, Rev. 2, 10-28-83.
32. SWEC Analog Wiring Diagram Loop A, 12210-AWD-27-7.5, Rev. 2, 10-28-83.
33. SWEC BOP Equipment List 1H13*P601, 12210-ESK-4CES601, Sh. 8, Rev. 6, 12-21-83.

A5.8-1 (Detail) continued.

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35. SWEC ECSIS Cable Block Diagram File Source Document, 3-19-84.
36. SWEC Analog Wiring Diagram, Instrumentation Power Supply and Distribution, Panel 1H13*P869, 12210-AWD-1-1.67. Rev. 6, 1-6-84.
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43. SWEC BIP Support Plan RHS .000-15
44. GE Drawing 914E644, Panel H13-P601 insert 17B, RHR Division II, Rev. 2
45. GE Drawing 914E647, Panel H13-P601 insert 20B, RHR Division I, Rev. 3.
46. SWEC Equipment List, Panel ICES*PNL6A, 12210-ESK-4CES191, Rev. 4, 10-26-83.
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56. SWEC Wiring Diagram, 12210-EE-3LQ-1, Panels ICES*PNL6B and 6C, Rev. 1, 8-12-83.
57. SWEC Wiring Diagram, 12210-EE-3FA-1, Panel ICES*PNL6A, Rev. 1, 8-24-83.
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61. SWEC E&DCR No. P21294, 10-26-83.
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A5.8-1 (Detail) continued.

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85. SWEC Analog Wiring Diagram 12210-AWD-22-6.6, Panel 1H13-P841, Rev. 6, 5-14-83.
86. SWEC Analog Wiring Diagram 12210-AWD-22-6.7, Panel 1H13-P841, Div. I, Rev. 4, 5-20-83.
87. SWEC Analog Wiring Diagram 12210-AWD-22-6.8, Panel 1H13-P842, Div. II, Rev. 4, 5-14-83.
88. SWEC Analog Wiring Diagram 12210-AWD-1-1.40, Dev. I, Instrumentation Panel 1H13*P841, P819, Rev. 5, 5-5-83.
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A5.9-1 (Detail) Electrical Separation

The team reviewed Stone & Webster Electrical Independence Design Criteria 12210-240.200 and the Final Safety Analysis Report (FSAR) to detemine commitments made to regulatory criteria and industry standards for electrical separation. We observed that: FSAR Table 1.8-1 cites Regulatory Guide 1.75 Revision 2 as the principal design standard; FSAR Section 8.3.1.4.2 provides summary criteria for separation of raceways; and Design Criteria 12210-240.200 cites IEEE-Standard 384-1974 and IEEE-Standard 420-1982 as the principal industry design standards. We observed that the design incorporates three safety-related electrical divisions (Division I, II, and III) and one non-safety-related division. The design also incorporates divisional cable chases rather than cable spreading rooms.

We determined that the Electrical Installation Specification 248.000 provides separation requirements pertinent to the installation and inspection activities. We also determined that spatial separation requirements to achieve independence of redundant safety-related circuits for cable tray, conduit, and cables in free air are provided in Drawings 12210-EE-34ZE, -ZF, -ZG, and -ZH. These drawings were recently developed. We determined that Stone & Webster has incorporated these drawings as clarifying documents into Section 3.3 of the Electrical Installation Specification 248.000 via Engineering and Design Coordination Report No. C-23836 in order to preclude misinterpretation of requirements. We reviewed these drawings and found that they were detailed and complete, and that they appeared to be consistent with Final Safety Analysis Report requirements and criteria presented in IEEE-384-1974.

We observed that Final Safety Analysis Report Table 8.3-7 lists various non-class 1E loads which are supplied from class 1E electrical buses. The Final Safety Analysis Report states that these non-1E loads are automatically tripped under accident conditions upon receipt of a LOCA signal to prevent degradation of the buses. We also observed that Section 3.3 of the Electrical Installation Specification 248.000 specifically lists these non-1E loads and requires the field to maintain required spatial separation for field cabling. We selected non-1E normal battery charger 1BYS-CHGR1A supplied from 480V 1E load center 1EJS*SWG1A, and reviewed the drawings to determine whether the design incorporated the required automatic trip on LOCA. We reviewed one-line diagram EE-1AA-5 and elementary diagram ESK-6EJS03 and found that the required LOCA trip feature was incorporated into the design.

We randomly selected main control room termination cabinets 1H13*P702 and 1H13*P730. We reviewed the main control room arrangement drawing and external connection wiring diagrams (EE-7 series) associated with each bay to determine whether the required electrical separation was incorporated into the design of field cable terminations. We found that all cables were routed to their respective divisional bays (e.g., Div. II, III or non-divisional) as required by the separation criteria.

We concluded that the principal reasons for the apparent lack of separation problems at the main control room termination cabinets are: the Stone & Webster Electrical Cable Schedule Information System computer data base cable program automatically routes cables to termination cabinet bays based on their respective divisional assignment, the program will not route cables to a divisional bay other than its own division; and Stone & Webster maintained control over assignment of field cables to bays, line coding of Field Interface Termination cables, and assignments of T-modes for field cable termination. These aspects appeared to reduce interface problems and therefore reduced the potential for separation conflicts.

A5.9-1 (Detail) continued.

We observed that project memorandum PMM-160 states that separation internal to panels was addressed by engineers on a case-by-case basis and less than 200 cases of potential separation conflicts existed in the design. The team also observed that PMM-160 stressed the use of Engineering and Design Coordination Reports and Nonconformance Dispositions to resolve problems and that no other problem solving or tracking mechanism is required. As a result of the Stone & Webster separation review effort Engineering and Design Coordination Report C-23836 was recently issued. This change incorporated the results of the review program into the Electrical Installation Specification 248.000 in order to document the review and facilitate inspection and resolution where necessary. We determined that the Stone & Webster separation review program resulted in the identification of three groups of equipment: equipment for which the existing design for separation is acceptable; equipment for which the as-designed configuration and internal wiring are acceptable, but a further review of actual separation of field cables is required; and equipment for which evaluation and analysis is required. We reviewed Engineering and Design Coordination Report C-23836 and randomly selected panel 1CES*PNL6B. This panel is a division II temperature transmitter panel located in the control building. We reviewed the EC-5 cable report and determined that cable 1CESNNC528 (non-1E 120V power receptical circuit) was the only cable terminated at this panel which was not a division II cable. We then inspected panel 1CES*PNL6B in the field. We verified that the subject cable was the only non-divisional cable in the panel, thereby confirming the EC-5 report. The cable was coiled and unterminated. We noted that the installer had maintained proper separation at the conduit entry gland; however, it appeared that there was insufficient room internal to the panel to maintain the required 6-inch separation without use of an enclosed raceway. We reviewed the cabling of several other equipment items and found no deviations from the results of the Stone & Webster separation review effort.

Based on our review, we concluded that Stone & Webster design was consistent with the Final Safety Analysis Report requirements with regard to electrical separation. We also concluded that few separation conflicts exist and that a controlled process is in place to resolve identified problems.

REFERENCES

1. SWEC Interoffice Memorandum, Electrical Separation, PMM-160, March 2, 1984.
2. SWEC Specification for Electrical Installation 248.000, Rev. 7, 12-30-83.
3. SWEC E&DCR No. C-23836, Separation Requirements, 4-12-84.
4. River Bend FSAR Section 8.3-7, Non-1E loads on 1E Buses.
5. SWE . 480V One Line Diagram, STBY Bus 1EJS*LDC 1A and 2A, 12210-EE-1AA-5, Rev. 5, 11-18-82.
6. SWEC Elementary Diagram, 12210-ESK-6EJS03, Rev. 10, 3-8-83.
7. SWEC Main Control Room Arrangement Drawing, 12210-EE-27A-9, Rev. 9, 9-8-82.
8. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay A, 12210-EE-7CH-2, Rev. 2, 2-19-84.
9. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay D, 12210-EE-7CK-2, Rev. 2, 9-26-83.
10. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay E, 12210-EE-7CL-3, Rev. 2, 12-28-83.
11. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay A, 12210-EE-7A-4, Rev. 4, 10-13-83.

A5.9-1 (Detail) continued.

12. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay B, 12210-EE-7B-4, Rev. 4, 10-18-83
13. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay D, 12210-EE-7C-4, Rev. 4, 10-24-83
14. SWEC External Connection Diagram, PGCC Termination Cabinet, 1H13*P702 Bay E, 12210-EE-7D, Rev. 4, 11-3-84
15. SWEC List EC-5 Report, Cable Location, Issue 77, 5-3-84.
16. SWEC Electrical Independence Design Criteria, 12210-240.200 Rev. 2, 9-10-79, and Rev. 3, Preliminary.
17. USNCR Regulatory Guide 1.75, Physical Independence of Electric Systems, Rev. 2, 9-78.
18. IEEE-384, IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits, 1974.
19. IEEE-420, IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations.
20. SWEC Standard Details for Separation Requirements, 12210-EE-34ZE-2, Rev. 2, 4-3-84.
21. SWEC Standard Details for Separation Requirements, 12210-EE-34ZG-2, Rev. 2, 4-3-84.
22. SWEC Standard Details for Separation Requirements, 12210-EE-34ZF-2, Rev. 2, 4-3-84.
23. SWEC Standard Details for Separation Requirements, 12210-EE-34ZH-2, Rev. 2, 4-3-84.

A5.11-1 (Detail) References for Section 5.11

REFERENCES

1. SWEC Calculation E-149, Verification Battery Size IENB*BAT01A, Rev. 2, December 23, 1983.
2. SWEC Calculation E-150, Verification Battery Size IENB*BAT01B, Rev. 2, December 23, 1983
3. SWEC Procedure, Engineering Assurance Procedure, EAP 5.3, Rev. 3, January 31, 1979.
4. SWEC Drawing, SWEC Arrangement Sill Detail and Welding Procedure in Control Building, No. 12210-EE-38AA-2, Rev. 2.
5. SWEC Procedure, QA Inspection Plan R1248000F0512, Rev. A, August 11, 1982.
6. SWEC Report, QC Report #2-05-12-CB-1319, December 27, 1982.
7. Standard, IEEE Recommended Practice for Sizing Large Lead Storage Batteries, IEEE 485-1978.
8. Calculation, Verification of Battery Size IENB*AT01A, Rev. 3, May 5, 1984.
9. SWEC Field Form, E&DCR No. C-2904B.
10. SWEC Field Form, E&DCR No. C-20908A.

A6.3-1 (Detail) Design Record Review, Specification and Qualification of Equipment, Field Design Changes, and 24Vdc Power Supplies

(a) Design Record Review

The team reviewed the Low Pressure Coolant injection valve interlock modification. The proposed modification (Reference 11) was reviewed and approved by the change control board dated November 4, 1983. The team found the review and approval controlled in accordance with GE's design review procedure (Reference 17). This modification involved 1 out of 2-twice logic of K relays and effected drawing ED-828E534AA and device list DL828E534AA. These drawings were in the process of revision.

Similarly in response to item II.K.3.18 of NRC NUREG (Reference 19), the Automatic Depressurization System actuation logic was to be modified to eliminate the need for manual actuation to assure adequate core cooling. The Boiling Water Reactor owners group performed a study and concluded to modify the Automatic Depressurization System logic to automate the system for the Design Bases Event and postulated plant accidents or transients. The change control board concurred with the position taken by the General Electric design team and indicated its preference for two of the six proposed logic modifications (Reference 12). The team reviewed the preferred logic modification and had no further questions.

In the area of compliance to the commitments to Regulatory Guide 1.97 (Reference 15), General Electric has no program to qualify all instrumentation for River Bend project. To comply with the Code of Federal Regulations (Reference 51) the scope of qualified equipment should include certain post accident monitoring components. Stone & Webster's letter to General Electric (Reference 13) identifies Nuclear Steam Supply System devices required for post accident monitoring for compliance with regulatory guide 1.97 (Reference 15). Gulf States endorsed this letter (Reference 14) and General Electric is in the process of providing design to comply with this requirement. Gulf States will direct the implementation. The General Electric Standard Safety Analysis Report submitted to the NRC includes compliance to Regulatory Guide 1.97. General Electric will include this requirement in their phase 3 program for Nuclear Steam Supply System electrical equipment qualification (in response to NUREG-0588 (Reference 16)).

(b) Specification and Qualification of Equipment

The team reviewed a Residual Heat Removal System pump motor purchase specification (Reference 1). The pump was purchased from Byron Jackson. The motor data sheet of the specification (Reference 28) includes pump data and the qualification parameters. The motor qualification report (Reference 23) from the motor division of General Electric (motor supplier) was reviewed and found controlled.

The team reviewed qualification documents for Crosby safety relief valves used in the Automatic Depressurization System application and the associated General Electric topical report (References 24 and 25). Qualification of this component is scheduled to be completed under phase 3 of electrical equipment qualification program.

The Rosemont transmitter purchase specification (Reference 26) was a generic specification and included qualification parameters, but did not include IEEE 323 (Reference 27) as a design basis document. The environment values in the specification were

A6.3-1 (Detail) continued.

from General Electric's general environment specification (Reference 30). We reviewed the qualification documents from Rosemont (Reference 29) and found that IEEE 323 qualification was included. We also reviewed General Electric/Stone & Webster correspondence relating to NRC Inspection and Enforcement Bulletin 80-16 with regard to the problems identified in Rosemont pressure transmitters Model 1151, 1152 with either A or D output code and in Model 510DU trip unit. Correspondence (11 letters) between General Electric, Stone & Webster and Gulf States was identified to ensure that the problem models were not used in the River Bend Project. A complete evaluation of Rosemont transmitters is included in phase 3 program of electrical equipment qualification.

(c) Field Design Changes

General Electric Specification "process instrumentation" (Reference 43) provides instrument line slope requirements. Stone & Webster locates the instruments. Gulf States' startup group noted, while writing the startup procedure in 1982, that the instruments indicating suppression pool level (normal suction source of the Residual Heat Removal System) were at least 6 feet above the suppression pool high water level. A high point vent in each instrument line was at a higher level. The instrument line to the high point vent would thus drain down to the suppression pool level, causing indication on the instruments to be meaningless. Stone & Webster wrote several instrument support design change notices providing details and isometric piping for the instruments to be relocated on Stone & Webster designed wall mounted racks. The revised position would not cause instrument line drain to the suppression pool level and would provide correct pressure indication. The team reviewed documentation of this problem and installation of the relocated instruments from Residual Heat Removal System instrument rack H22-PO21 (References 33-38, 44). The team had no further question.

While reviewing this change, the team noted that a revision of a panel drawing (Reference 41) indicated change in essentiality code of transmitters E12-NO50B and E12-NO51B from an "A" (essential with active safety function) to "P" (essential with passive safety function). Previous revision of this drawing (Reference 40) included an "A" classification. The team reviewed the Engineering Change Authorization (References 42 and 39) for this change and concluded that the change was controlled and the transmitter is used for an essential passive safety function. The power supply to the transmitter was retained as Class 1E source of control power. The team had no further questions.

(d) 24Vdc Power Supply Qualification

General Electric's Emergency Core Cooling System initiation logic uses 24Vdc power supplies rectified from 120Vac power supplies. The Division 2 supply is similarly derived. The rectifier module with 120 to 24 volts transformer is procured from General Electric or SOLA Corporation as a commercial grade power supply. The team reviewed detail drawings of this module (Reference 47) and noted that they are located in the control room. General Electric Equipment Qualification division has an in-house program to qualify these commercial grade units for use in the safety system. The team reviewed the qualification specification (Reference 48). This specification includes temperature, humidity, and radiation parameters and requires compliance with the IEEE-323-71 requirements and the regulatory guide for seismic qualification. General Electric performs tests on sample components. If the second lot purchased from the same manufacturer has no change, then qualification of the second lot is established by

A6.3-1 (Detail) continued.

similarity. This procedure is called "In House dedication" (Reference 49) and is used to demonstrate that a commercial grade item is acceptable for nuclear safety-related application. It requires a two step procedure.

(1) A representative sample qualification.
(2) Quality Assurance testing to establish that the items received are representative of the one qualified for the Class 1E application in step 1.

The team reviewed this procedure and a sample qualification report (Reference 50). The team found the process controlled and had no further questions.

(e) References referred to in this detail are listed in Detail A6.3-2.

A6.3-2 (Detail) References for Section 6.3

1. GE Specification, RHR Pump Motor for BWR/621A3504, Revision 1, 7/14/77.
2. GE Specification, RHR system design #22A3845, MPL #E12-4010, Revision 2, 2/1/77.
3. GE Data Sheet, RHR System Design Specification Data Sheet #22A3845AB MPL#E12-4010, Revision 5, 7/15/83.
4. GE Specification, Nuclear Boiler (ADS) Design #22A4622 MPL#B21-4020, Revision 5, 3/16/83.
5. GE Data Sheet, Nuclear Boiler, Design Specification Data Sheet #22A4622AT MPL #B21-4020, Revision 3, 3/23/84.
6. GE Parts List, RHR system, MPL #18N06B041, Rivision 6, 1/4/84.
7 GE Specification, Regulatory Requirements and Industrial Standards. Design Bases #22A5276, MPL #A42-4070, Revision 2, 10/19/81.
8. GE Drawing, RHR System Functional Control Diagram (FCD) for River Bend Fig.#7.3-4, Revision 1.
9. GE Drawing, RHR System P&ID, Figure 5.4-12, Revision 4
10. GE Verification Sheet, Regulatory Requirements for RHR System MPL #E12-1050, 6/17/82.
11. GE Document, ECA & PWA #4214LD, Modification 02/84 approved by Change Control Board on 11/4/83.
12. GE Document, Modification of ADS Logic, NEDE-30045, Concurred by the Change Control Board, 2/83.
13. S&W letter to GE, Identifying NSSS Devices for Reg. Guide 1.97 Compliance, 11/22/83.
14. GSU letter to GE, Endorsing NSSS Device List from S&W for Reg Guide 1.97 Compliance, 3/6/84.
15. NRC Regulatory Guide, Instrumentation for LWCNPP to Assess Plant and Environmental Conditions During and Following An Accident (RG 1.97), Revision 2, 1980.
16. NRC Guide, Staff Position on Environmental Qualification of safety-related Electrical Equipment NUREG-0588, Revision 1.
17. GE Procedure, Engineering Operating Procedure EOP-40-700, Revision 4, 4/25/83.
18. River Bend FSAR, Safety Related Display Instrumentation Section 7.5, Table 7.5.2, 2/83.
19. NRC Guide, TMI Action Plan NUREG 0737.
20. NRC Regulatory Guide 1.75 for Physical Independence of Electric Systems.
21. GE Specification Electrical Equipment, Separation of Protection Systems, 22A3728 MPL #A62-4050, Revision 4, 8/26/81.
22. GE Drawing, RHR Elementary Diagram No. 838E534AA sheet 8, Revision 1, 2/14/84.
23. GE Report, RHR Pump Motor Qualification Report #456HA898-1, Revision 12, 5/20/83.
24. Crosby Report, Solenoid Pilot Valve Qualification Crosby IMF-2, approved by GE on 6/3/82, 5/6/82.
25. GE Report, Licensing Topical Report on GE Environmental Qualification Program NEDE 24326-P-1.
26. GE Specification, Rosemont Transmitters, Generic Specification #249A1945, Revision 2, 11/5/80.
27. GE Report, Rosemont Transmitter Qualification #169C8391, Revision 2, 12/20/79
28. GE Data Sheet, RHR Pump motor 21A3504BV MPL No. E12-C002, Revision 1, 9/7/81.
29. IEEE Standard, Qualifying Class 1E Electric Equipment IEEE 323, 1971.

A6.3-2 (Detail) continued.

30. GE Specification, GE's General Environment Specification #22A3008 for BWR5 and #22A3093 for BWR 6.
31. S&W/GE Letters, (11 letters) dated 11/4/80, 8/25/80, 9/25/80, 9/5/80, 12/2/80, 12/4/80, 5/13/81, 6/19/81, 10/2/81, 4/27/83, 5/3/83, and 2/6/84.
32. GE Document, ECA/ECN for Changing the Design Specification Data Sheet #B21-4020 for the Nuclear Boiling System (ADS), 5/18/84.
33. GSU letter to S&W, Identifying Instruments to be Relocated from the RHR Instrument Racks STG-L-8-89, 7/19/82.
34. S&W Memo, Interoffice Memo from S&W (CHOC) to S&W Field, 8/4/82.
35. S&W Memo, S&W to GE for Initiating FDDR to Authorize Change and Relocation, 2/3/83.
36. GE Document, GE FDDR #LDI-521, Authorizing the Change Revision 0, 2/10/83.
37. S&W Drawings, Various ICRNs Providing Details (isometric) of the Relocation of RHR Instruments.
38. GE Document, E&DCR # C-40,096 to Include FDDR #LDI-521 in the Installation Specification #247.00, Revision 1.
39. GE Document, ECA #820330-1A - for Essentiality Classification Change, Revision 0, 3/30/82.
40. GE Drawing EDL, RHR Local Panel B.H22-P021 Dwg #368X554BA, Revision 9, 7/25/81.
41. GE Drawing EDL, RHR Local Panel B.H22-P021 Dwg #368X554BA, Revision 14, 7/2/83.
42. GE Document, ECA#790918-1 for Essentiality Classification Change, Revision 0, 9/24/79.
43. GE Specification, Process Instrumentation No. 22A3137, Revision 6
44. GE Document, FDDR #LDI-521 Confirming Seismic Qualification of the Rack, Revision 2, 2/2/84.
45. GE Drawing, Elementary Diagram, B28E534AA, Sh. 9 and 17.
46. S&W Drawing, Sketch No. ESK-5-RH501 for Pump A.
47. GE Drawing, 24Vdc Power Supply, 1184C4548, Sh. 1 and 2.
48. GE Specification Qualification Specification, 184C4571, Sh. 1 and 2.
49. GE Procedure,"In House Dedication" Procedure, MP5.25, 7/7/80.
50. GE Report, Qualification Report, DRF No. A00-01294-2, Received and Approved on 9/8/82, 2/4/82.
51. Federal Regulation, Code of Federal Regulations 10 CFR50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.

A.A.2-1 (Detail) Division Manager Technical Reviews

In April of 1980 the Engineering Department Manager of the Cherry Hill Office Center directed each Division Manager, as part of administrative routine, to review at least one drawing series, one specification, and one calculation per month. The purpose of the directive was to help ensure a consistency of quality and technical adequacy on the various engineering projects at the Cherry Hill Office Center. Each Division Manager was to select products for review at random and representative of all engineering work in progress (i.e., nuclear, fossil, or industrial).

The inspection team was shown copies of the Power Division Manager's technical review schedules for 1981 through 1984. The following identifies the engineering products scheduled for technical review and directly applicable to the River Bend Project.

Date	Drawing Title	Specification Title	Calculation Title
4/80	High Pressure Core Spray System: EP 83 A-5, B-5, C-5	Insert type Check Valves No. 2 1/2 in. and larger No. 228-229 and addendum 1 & 2 No. PB-008	Radwaste Building Standpipe Sizing and System Flow Requirements
9/81	Main Steam Piping: EP2A, B, C, D, E, F, and G	Shop Fabricated Piping ASME III, Code Class I, II, and III: ANSI-B31.1, Class IV: 228.150 Rev. 1	Fuel Building Exhaust Rate: PB-120
6/82	Control Rod Drive System: FSK-36-1	Fabrication and Erection of CRD System Piping 228.180 Rev. 1	CRD System Pump/Piping Verification 12210-PN-217 Rev. A
2/83	Tunnel Piping: 12210-EP-108's	Pipe Erection: 228.160	RHR System Sizing: 12210-PN-268, Rev. 1
10/83	Feedwater: FSK 6-1	Standby Service Water Pumps: 232.920	Standby Cooling Tower Sizing: 12210-PM-194-1
2/84	Penetration Valve Leakage Control System BP-339R-1 Sheets 1&2	Field Fabricated and Erected Piping 228.160, Rev. 4	Design Minimum Wall Thickness Residual Heat Removal Piping 12210-SPM-6

The inspection team was informed that Division Manager technical reviews were not part of the design verification process or the quality assurance audit program, and therefore they were not auditable. As a result the reviewer's comments and their resolutions have not been formally maintained. However, documentation in the form of informal memorandum were obtained for all of the scheduled reviews except the February 1984 technical review. Stone & Webster explained that the documentation of this review is not in the file because the review is ongoing and not complete.

The team examined the reviewer's comments to draw a conclusion as to the depth of the reviews and the technical quality of the comments made. The five technical reviews (3 engineering products per review) were performed by three different individuals. Of the 15 engineering products reviewed, the technical reviewers had no comments

A.A.2-1 (Detail) continued.

or indicated that the products were technically adequate and conformed to Stone and Webster technical and administrative procedures in 10 instances. The comments made on the other 5 engineering products were in general not substantive with three exceptions. These exceptions were comments dealing with inconsistencies and descrepancies between a calculation and referenced drawings, with the apparent incorrect use of General Electric interface information, and with the failure to include all information required by subsection NA-3250 for a ASME III design specification.

The sample of engineering products selected for review on or about February 1983 included Power Division's calculation PN-268 Rev. 1, "RHR System Pumps TDH and NPSHa." A later revision of this calculation was also reviewed by the inspection team. A comparison of the two reviews can be made, because the calculation revision reviewed by the inspection team had only a reference added as compared to that reviewed by the Division technical reviewer. The Division technical reviewer did not identify substantive findings nor the deficiencies which the inspection team found in the calculation (see Deficiency D2.3-1).

The team found that Power Division technical reviews were being performed on River Bend engineering products (i.e., drawing, specifications and calculations) at a rate of approximately 4.5 per year. The team found that the technical reviews in general did not produce substantial comments and were not totally effective in identifying existing deficiencies.

A.A.2-2 (Detail) Engineering Assurance Division Audits

During the inspection the team was informed that the Engineering Assurance Division had conducted several system level design audits recently. These audits were conducted to determine if project activities are performed in a manner that will ensure a quality product and to fulfill audit requirements of Stone & Webster's quality assurance plan. The team was informed that recent Engineering Assurance audits have been structured to examine a particular system and its related documents. The team was informed that the engineering audit teams were tasked to verify that applicable procedures were followed and to examine the technical adequacy of the particular system. To accomplish the latter task the Engineering Assurance audit teams were augmented with technical personnel.

The team examined 3 Engineering Assurance audit reports which were described as audits at a system level. These audits were on the liquid waste system, the battery system, and the reactor plant ventilation system. Each of the audit reports identified the listing of activities reviewed. Only the audit reports on the battery and reactor plant ventilation system appear to be system level reviews. Specifically the audit of the battery system reviewed specifications, supplier documents, supplier drawings, design criteria, design consistency, elementary diagrams (ESKs), one-line diagrams, cable block diagrams (CBDs), project drawings, and calculations. Similarly the audit of the reactor plant ventilation system reviewed design consistency and technical adequacy among FSAR sections, supplier drawings, flow diagrams (FSKs), elementary diagrams, logic diagrams (LSKs), production drawings, and calculations.

The Engineering Assurance audits identified documentation deficiencies and procedural violations; however, no significant design inadequacies were identified. The following are examples of more significant design related errors identified. In parenthesis are deficiencies and observations made by the inspection team that appear to be similar or related.

> Design details described in the FSAR are not consistent with the actual design per design calculations and procurement specifications (Deficiencies D2.5-4, D2.4-1, D3.4-4, D3.4-6, D3.5-2, D6.5-2, D4.3-3).
>
> Safety Category I calculations do not contain evidence of independent review (Observation O4.4-1).
>
> Some electrical cable designs were found not in compliance with generic criteria for sizing 600 volt cables. Exception to the generic criteria was granted through calculations associated with interoffice correspondence. These calculations did not fulfill the documentation requirements of EAP 5.3. In addition routed lengths of some cables exceeded the lengths authorized by the interoffice correspondence calculations (Deficiencies D2.5-5, D2.5-7, 5.11-1).

During the team's examination of the Engineering Assurance audit reports, it was noted that two battery sizing calculations reviewed by Engineering Assurance personnel were subsequently reviewed by the team during the inspection. Specifically calculations E-149 and E-150 were audited during September, 1983 by Stone & Webster personnel. As a result of this audit the calculations were revised in December, 1983. In spite of the Engineering Assurance audit observation and subsequent revision of the calculations, the team found design input (namely capacity rating factor) was not obtained from vendor or other suitable documents.

A.A.2-2 (Detail) continued.

The team also observed that only one Engineering Assurance audit appeared to be augmented with technical personnel. This audit was on the battery system where an engineer from the Electrical Division in Boston was used to provide technical expertise.

DETAIL A.A.2-3

Document Type	EAP	Responsible for Independent Objective Review	Method of Documentation
System Descriptions	3.7	Operational Design Review (ODR) Group, Operations Services Div.	Sign title page
Technical Topical Reports	2.6	Reviewer designated by EAP 2.6	Approve "Approval Slip" per EAP 2.6
Preliminary Safety Analysis Report (see note)	2.9 2.10	Division Licensing Representative	Approve Review/ Approval Slip per EAP 2.9, or Change Request
Conceptual Dwgs o Site Plan o Plot Plan o Gen Arrangements	5.17	ODR Group, Operations Services Division	Initial drawing
Flow Diagrams	5.9 5.16	ODR Group, Operations Services Division	Initial diagram
Logic Diagrams	5.10	ODR Group, Operations Services Division	Initial diagram
One Line Diagrams	5.13	Reviewer designated by Chief Engineer, Electrical Div.	Initial diagram
Electrical Design Criteria	5.21	Electrical Division Specialist	Sign title page
Structural Design Criteria	5.19	Reviewers designated according to EAP 5.19	Sign title page
Master Specifications	4.12	Reviewer designated according to to EAP 4.12	Per EAP 4.12
Project Specifications	4.13	Reviewer designated according to to EAP 4.13	Per EAP 4.13
Design Specifications for Structural Support and MC Components	4.7	Reviewer designated according to EAP 4.7	Per EAP 4.7
Calculations	5.3	Reviewer designated according to EAP 5.3	Per EAP 5.3

NOTE: The PSAR is a "key design document" only when it is the first documentation of the design inputs. In this case, the PSAR remains a "key design document" only until subsequent documents are issued to record this information.

D2.3-1 (Deficiency) RHR Pump Runout Calculation

BACKGROUND

The low pressure coolant injection mode of operation has a design requirement to not exceed a runout flow of 6060 gpm. This interface requirement is stated on the General Electric process diagram (Reference 1). The General Electric process diagram also recommends a resistance orifice be installed and sized. Stone & Webster performed calculation PN-268 to size the resistance orifice. The calculation determined a runout flow of 5575 gpm. It was thus concluded that since the calculated flow rate was less than 6060 gpm an orifice was not required to prevent runout.

DESCRIPTION

The Stone & Webster calculation contained two substantial errors:

1. The head curve used to determine the runout flow is approximately 40 ft of head less than the certified pump head curve (Reference 3). This results in runout conditions at lower flow when the system resistance curve is extrapolated to intersect the head curve.

2. The method used to determine the system runout flow was an incorrect graphical technique. The technique used would also lead to a lower value for runout flow.

In combination these two errors led to a calculated runout flow of 5575 gpm. This was well below the limit of 6060 gpm and appeared to indicate that a flow restricting orifice was not needed. However, using the correct head curve and proper graphical technique would have yielded a calculated runout flow of 5950 gpm with other factors remaining the same. For all practical purposes this is the same as the limit of 6060 gpm.

In addition, although the calculation was generally conservative for determining the pump head requirements it was nonconservative with respect to limiting runout flow in the following respects:

1. Piping bends in the suppression pool were apparently assumed for friction loss purposes to be 90 degree bends. The EP drawings (Reference 4) indicate the bends to be approximately 15 degrees.

2. Several fittings were assumed to be standard fittings for friction drop purposes, whereas the EP drawings require that long radius fittings be installed. The site walk through of the system revealed that indeed long radius fittings were installed.

3. A comparison between the system fittings assumed in the calculation, what is installed and what is called out on the EP drawings showed that more fittings were included in the calculation.

As discussed above, the calculated runout flow is essentially at the limit using the correct pump curve and the correct graphical technique. Thus, considering the non-conservative nature of the remainder of the calculation for runout purposes, there appears to be real potential for exceeding the 6060 gpm limit.

D2.3-1 (Deficiency) continued.

BASIS

The calculation performed is not adequate for evaluating runout flow in order to assure that an NPSH problem will not occur.

IMPACT ON DESIGN

The potential impact is the need to install resistance orifices in the three residual heat removal loops for the low pressure coolant injection mode in order to prevent excessive flow. Excessive flow could result in a NPSH required greater than that available and thus cavitation.

EXTENT

The errors of using the lower head curve and the incorrect graphical technique for extrapolating system resistance appear to be isolated. The nonconservative approach of using a total dynamic head calculational approach for determining runout flow appears to be systematic.

REFERENCES

1. GE Process Diagram 762E425AA, Sheet 1, Rev. 4.
2. S&W Calculation PN-268, RHR System Pumps TDH And NPSHa, Rev. 2.
3. Byron-Jackson Pump Curve T-36766-1.
4. S&W Drawings 12210-EP71 REV.5, Sheets A thru H, J, and K.

D2.3-2 (Deficiency) Failure to Prepare a Calculation that meets the Stated Objectives

BACKGROUND

In response to design concerns following the Three Mile Island accident, the owners of BWR-6 designs formed an Owners Group. This Owners Group identified various post Three Mile Island issues and addressed each generically. These issues were called Licensee Review Group II issues.

Licensee Review Group II Issue 5-RSB concerns the control of post-LOCA leakage to protect emergency core cooling systems and preserve suppression pool level. To satisfy this issue Licensee Review Group II participants must demonstrate that passive failures (i.e., leakage from the first isolation valve outside of the suppression pool) will be contained so that the suppression pool is not drained and that redundant emergency core cooling system equipment is not flooded out. In response to Licensee Review Group II Issue 5-RSB, the Power Division prepared a calculation PN-283, Revision 1. The calculation had two stated objectives. The first objective was to determine that suppression pool levels due to leakage into the Auxiliary Building pump cubicles are acceptable. The second objective was to determine that the leakage from the first isolation valve outside of the suppression pool will be contained so that the suppression pool is not drained, nor is emergency core cooling system equipment flooded out.

Calculation PN-283 is a Quality Assurance Category I nuclear safety-related calculation that has been checked and independently reviewed.

DESCRIPTION

The assumptions used to evaluate the consequences on suppression pool level from post-LOCA leakage do not support the intent of the calculation and may not be conservative. The basis for assuming a 100 gallon per minute leak into a pump cubicle is not identified. Since the pump cubicles contain sumps and sump pumps which automatically start, leakage rates lower than 100 gallons per minute may be more conservative considering the ability of the operator to recognize the leak. Although not explicitly stated the calculation also assumes that two non-safety-related sump pumps, each with a pumping capacity of 50 gallons per minute, are not functioning as designed. It appears that since the sump pumps were not safety-related their successful operation was ignored. The failure to consider the effects of proper functioning sump pumps may be a non-conservative assumption with respect to maximizing the loss of suppression pool level, considering the operator response times as discussed above.

With respect to the second objective, the calculation did not address the consequences of leakage from the first isolation valve outside of the suppression pool.

The calculation only addressed leakage into the residual heat removal pump cubicle. The first isolation valves outside of the suppression pool are located in emergency core cooling valve cresent area between the suppression pool and the separate watertight emergency core cooling system equipment rooms.

BASIS

The assumptions used to demonstrate sufficient suppression pool level following leakage into the largest emergency core cooling system equipment room were not selected to maximize loss of suppression pool water. The calculation did not address the consequences

D2.3-2 (Deficiency) continued.

of leakage from the first isolation valve outside the suppression pool as required by the objective of the calculation.

The Independent Reviewer of the calculation did not perform a review that meets the review requirements of EAP 5.3, Rev. 3, Attachment 3.0. Specifically, the Independent Reviewer did not ensure that all assumptions were uniquely identified as assumptions and adequately described; that all assumptions were reasonable; and that the results were applicable and appropriate to the objective.

IMPACT ON DESIGN

The failure to select assumptions which maximize the loss of suppression pool water through the leakage into an emergency core cooling system equipment room does not appear to have a specific impact on the design. It is expected that sufficient instrumentation is available to the operator to recognize the leak and take corrective action before jeopardizing suppression pool level. However, leakage from the first isolation valve outside the suppression pool into the emergency core cooling valve cresent area may have a design impact. The subject area is significantly larger than any of the emergency core cooling system equipment rooms. In addition, this area does not have safety-related level indication in the sumps or on the operating floor levels. The consequences of a leak are more severe, since all emergency core cooling system suctions from the suppression pool have isolation valves in this area.

EXTENT

This deficiency is an example of a weakness in the implementaton of the design verification process on River Bend Project. Taken collectively with other deficiencies, there appears to be a systematic weakness in the design verification process.

REFERENCES

1. LRG II Position Paper, 1-25-82, Control of Post-LOCA Leakage to Protect ECCS and Preserve Suppression Pool Level, Attachment to letter from Mr. D. L. Holtzscher (LRG-II Working Group) to Mr. H. J. Faulkner (Division of Licensing, U.S. Nuclear Regulatory Commission), 1-25-82.
2. S&W Calculation PN-283, Auxiliary Building ECCS, RHR, RCIC, Pump Cubicle Flooding and Associated Effects Upon the Suppression Pool Level, Rev. 1, 5-13-82.
3. Engineering Assurance Procedure (EAP) 5.3, Preparation and Control of Manual and Computerized Calculations (Nuclear Projects), Rev. 3, 1-31-79.
4. FSAR Section 6.3.

D2.3-3 (Deficiency) Failure to Consider Passive Piping Failures in Emergency Core Cooling System Suction Lines Post-LOCA

BACKGROUND

General Design Criterion 35 provides basic requirements for a system to provide abundant emergency core cooling. The criterion states that the emergency core cooling system shall perform its safety function assuming a single failure. ANSI Standard 58.9-1981 defines a single failure and separates the types of single failures into active and passive failures. Following a loss of coolant accident the postulated single failure can be either an active or passive failure. In the short term only a single active failure is postulated. However, in the long term a single passive failure is postulated. Stone and Webster has included the ANSI Standard's definition of single failure and its applicaton in technical procedure PTP-0.3.1-0.

DESCRIPTION

A quantitative analysis/evaluation has not been performed to determine the effect on the ability to maintain core cooling and safe shutdown assuming a passive piping failure in one of the emergency core cooling system suction lines post-LOCA. Specifically, the consequences of a passive piping failure in one of the emergency core cooling system suction lines between the first isolation valve outside of the suppression pool and its associated watertight emergency core cooling system equipment room were not analyzed.

BASIS

Stone & Webster's Power Division Technical Procedure Number PTP-0.3.1-0 requires that a passive failure be considered during the long term following a loss of coolant accident.

IMPACT ON DESIGN

The failure to consider a passive single failure in the emergency core cooling system suction lines post-LOCA may cause design changes. The emergency core cooling system valve cresent area is a large area containing containment isolation valves for each of the suppression pool suction lines. The cresent area is sufficiently large such that leakage from an unisolated leak can result in a loss of suppression pool water below minimum acceptable levels. Consequently, the leak must be isolated. However, determination of which emergency core cooling system suction line is leaking will be difficult. The cresent area does not have safety-related level indication in the sumps or on the operating floor levels. Given that the operator recognizes that there is a leak in the cresent area, he must isolate the leak prior to submerging the containment isolation valve motors (not environmentally qualified for submergence).

In lieu of design changes the applicant may elect to exempt the component from passive failure. Where such exemptions are taken a report with technical details supporting the exemption is required. Low stresses and augmented inservice inspection may be acceptable alternatives to design changes.

EXTENT

This deficiency is not considered to be systematic, because other passive failures were not identified by the inspection team which could prevent providing emergency

D2.3-3 (Deficiency) continued.

core cooling. However, the failure to recognize that a passive failure in the emergency core cooling system suction lines was not considered represents a weakness in the design verification process at Stone & Webster.

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 35.
2. ANSI/ANS 58.9-1981, Single-Failure Criteria for LWR Safety-Related Fluid Systems.
3. PTP-0.3.1-0, Single-Failure Criteria for Nuclear Plants, 6-23-80.
4. Standard Review Plan 6.3, page 6.3-8.

D2.3-4 (Deficiency) Lack of Documented Basis for Sizing Safety Class 2 Equipment

BACKGROUND

The emergency core cooling system has Safety Class 2 subsystems that are used to assure the discharge lines of the main system pumps are filled. This is accomplished by a single pump per subsystem. The design flowrate for each pump should be sufficient to counter leakage across boundary valves and have a head capable of delivering flow to the uppermost elevation of each emergency core cooling system.

DESCRIPTION

The pumps have been specified and purchased with a design flow of 50 gpm. The calculation (Reference 1) which established the head requirement at the design flow referenced the PSAR as the basis for the 50 gpm. Further investigation into the adequacy of the 50 gpm found that the actual basis was a "voided" calculation. The reason for the calculation being voided was said to be housekeeping. A review of the "voided" calculation revealed a non-system specific set of assumptions for both the leak rate per valve and the number of boundary valves in each emergency core cooling subsystem.

BASIS

The current situation is one of having a key parameter in the design of a Safety Class 2 system without documented basis or engineering justification.

IMPACT ON DESIGN

If the 50 gpm proves to be insufficient, there would be a major impact on the design. This is not expected to be the case.

EXTENT

All the emergency core cooling system fill pumps have been sized, purchased and installed without documentation design basis.

REFERENCES

1. S&W Calculation PN-048, Subsystem Fill Pump Sizing, Rev.1.

D2.3-5 (Deficiency) Insufficient LPCS Fill Pump TDH

BACKGROUND

Stone & Webster calculation PN-048 was performed to size the emergency core cooling subsystem fill pumps. The design basis was set at 50 gpm. The calculation was performed to determine the total dynamic head, NPSH and heat rejection at that flow rate.

DESCRIPTION

A number-by-number check of the calculation in question revealed that the total dynamic head calculation for the low pressure cooling system fill pump was in error. This error was carried forward to give a total dynamic head required for the low pressure cooling system fill pump of 97.4 feet. The correct total dynamic head using 24 feet 4 inches versus the erroneous value of 34 feet 4 inches was estimated to be approximately 102 feet. In checking the certified head curve to determine if the error was significant, it was found that the curve did not even satisfy the results of the original calculation. The certified head curve indicates a total dynamic head of 87.5 feet at 50 gpm. Thus the pump purchased based on the calculation has insufficient head.

BASIS

The design process has failed with respect to this calculation which provides the basis for the pump purchased. This situation also existed for the residual heat removal fill pump until the specification data sheet was later made to agree with the calculation by addendum. The review of vendor certified performance data should not have allowed this type of error to persist.

IMPACT ON DESIGN

The installed pump will not provide the design leakage flow of 50 gpm. The pump will deliver water to the highest point in the system assuming less leakage and a cooling recirculation flow of 10 gpm.

EXTENT

Since a similar situation was discovered with the residual heat removal fill pump, this situation should be considered as systematic for the emergency core cooling subsystem fill pumps.

REFERENCES

1. S&W Calculation PN-048, Subsystem Fill Pump Sizing, Rev. 1.
2. Goulds Pumps Certified Pump Head Curve A-28379-N767283, Rev. 0.

D2.3-6 (Deficiency) Preoperational Tests for Confirming Compliance with System Design Bases

BACKGROUND

Preoperational tests are intended to verify operability of a system and to confirm that the system has been designed to meet the design basis. The design basis for the low pressure coolant injection mode of the residual heat removal system was provided by General Electric. Gulf States Utilities is responsible for developing the various preoperational tests for this system.

DESCRIPTION

One of the preoperational tests for the residual heat removal system is to verify the pump head curve and system resistance for the low pressure coolant injection mode. From this information the minimum, rated and runout flows are verified to be within the functional limits set by General Electric. The preoperational test for each residual heat removal loop currently starts with a requirement to verify the sizing of the resistance orifice in a runout flow condition of 5950 gpm with the reactor pressure vessel injection valve wide open. When the responsible Gulf States Utilities Preoperational Test Supervisor was informed that these resistance orifices are not installed, he indicated there was a problem with that portion of the test program.

BASIS

It is apparent that the Gulf States Utilities staff did not develop the low pressure coolant injection portion of the preoperational tests in conjunction with the Stone and Webster design documents (in this case FSK's) and the actual installed system.

IMPACT ON DESIGN

Currently, this portion of the preoperational tests cannot be performed because the first step of the test is to size something that is not installed.

EXTENT

Gulf States Utilities staff is apparently working from General Electric supplied documents without checking with Stone & Webster design documents to verify that the tests are applicable to the installed system. This (elimination of a restricting orifice) is a portion of the residual heat removal system that Stone & Webster departed from standard General Electric practice.

REFERENCES

1. GSU Preop Test Section 7.9 - Hydraulic Resistance and System Flow Path Verification.

D2.3-7 (Deficiency) Preoperational Test to Verify LPCI NPSHa

BACKGROUND

The General Electric design basis (Reference 1) requires a certain NPSH available for runout flow in the low pressure coolant injection mode (A-2). The requirement states as follows:

"The NPSH available in mode A-2 at a reference location 2 feet above the pump mounting flange must equal or exceed 5 feet assuming saturation temperature of 212 degrees F. The NPSH available at the pump suction nozzle must equal or exceed this value plus the difference in elevation between the reference location and centerline of the pump suction nozzle."

DESCRIPTION

The review of the preoperational test data sheet that will determine the NPSH available at the conditions mentioned in the General Electric requirement revealed the following:

1. The requirement to adjust the value to the pump suction was not included.

2. The equation converting the NPSH available at test conditions (temperature and pressure) to 212°F saturated conditions is not correct.

BASIS

The conversion methodology reviewed contained terms for test condition barometric pressure and the vapor pressure for 212°F. Both of these terms are not necessary.

IMPACT ON DESIGN

This portion of the preoperational tests is important because Stone & Webster calculations were not able to verify that the system design met this requirement. Stone & Webster requested relief (to a subcooled condition of 210°F) from this requirement from General Electric. General Electric responded by accepting the change but reserving a final decision which will depend on the results of the preoperational tests.

EXTENT

Discussion with the Gulf States Utilities supervisor revealed some confusion and concern with how to make the conversion from test to design basis conditions.

REFERENCES

1. GE Drawing 762E425AA Rev. 4, RHR System Process Diagram Sheet 1 Note 8.
2. Gulf States Utilities Preop Test Data Sheet 9.18.

D2.4-1 (Deficiency) ADS Pneumatic Sizing

BACKGROUND

Stone & Webster performed design calculations (Reference 1) to verify adequate capacity of the penetration valve leakage control system to provide short and long term air supply to the safety relief valves. The penetration valve leakage control system is the safety grade source of air supply for the safety relief valves.

DESCRIPTION

The calculations for maximum air supply flow were performed based on one low-low set safety relief valve opening 100 times in 6 hours. A more limiting condition specified in the design basis documents is one low-low set safety relief valve opening two times per minute (Reference 2).

BASIS

The calculations to verify adequate penetration valve leakage control system capacity were performed using an assumption that was inconsistent with the appropriate design specifications for safety relief valve operation.

IMPACT ON DESIGN

This deficiency has no impact on design or analysis since the penetration valve leakage control system has sufficient margin to provide the higher air flow needed for safety relief valve operation. The analysis documentation should be corrected.

EXTENT

This deficiency is representative of improper implementation of the design verification process in that the independent reviewer of the calculation did not perform a review that meets the review requirements of EAP 5.3, Rev. 3. Attachment 3.0. Based upon a review of other areas this type of deficiency appears to be systematic.

REFERENCES

1. Stone & Webster, Calculation, Main Steam Safety Relief Valve Pneumatic System Piping Design Verification, PN-255 Revision 1.
2. GE, Nuclear Boiler System Design Specificaion Data Sheet, 22A4622, AT, Revision 1, July 18, 1983.

D2.4-2 (Deficiency) ADS Pneumatic Supply Adequacy

BACKGROUND

Stone & Webster performed design calculations to verify the adequacy of the safety grade air supply to the safety relief valves.

DESCRIPTION

This calculation was initially performed based on an air compressor providing 150 psig. Later it was revised based on an air compressor providing 120 psig. The air compressor actually purchased for River Bend will provide between 106-114 psig. However, the design calculation was not revised again to reflect of this lower pressure (106 - 114 psig).

BASIS

The design calculation is not consistent with the actual design.

IMPACT ON DESIGN

This error has no impact on design adequacy, since the compressors purchased are adequate to supply long term air requirements. The analysis and documentation need correction.

EXTENT

This deficiency does not appear systematic based on a review of other calculations.

REFERENCES

1. Stone & Webster, Calculation, Main Steam Safety Relief Valve Pneumatic System Piping Design Verification, PN-255, Revision 1.
2. Stone & Webster, IOC, Confirmation of Inputs/Assumptions, 5-1-84.

D2.4-3 (Deficiency) ADS Pneumatic Supply Specification

BACKGROUND

The automatic depressurization system pneumatic supply design specifications (References 1 and 2) require that a safety grade pneumatic supply of 150 psig minimum be available continuously during plant operation and emergency conditions.

DESCRIPTION

The safety grade pneumatic supply for River Bend Station is provided by the penetration valve leakage control system compressors which operate between 106-114 psig. T re is another air compressor that will supply 150 psig; however, it is not safety grade

The safety grade supply pressure is below the 150 psig design basis minimum specified by the design freeze specification agreed to by Stone & Webster and General Electric (Reference 3). Stone & Webster had not informed General Electric or requested a variance. Accordingly, the plant safety analyses performed by General Electric has not been reviewed in light of the reduced air pressure.

BASIS

The safety grade pneumatic supply pressure provided by the penetration valve leakage control system is below the minimum established by the design specifications and is not consistent with the plant safety analyses performed by General Electric.

IMPACT

This item is not expected to require extensive hardware changes, since the non-safety grade air compressors can keep the automatic depressurization system accumulators charged to 150 psig during normal operation and the penetration valve leakage control system can make up for consumption and leakage in the long term. It is likely that upgraded instrumentation for monitoring the non-safety grade air supply will be needed in order to meet the commitments to Regulatory Guide 1.97. The design specifications and plant safety analyses will need to be revised as appropriate.

EXTENT

This deficiency does not appear systematic based on a review of other systems.

REFERENCES

1. GE, Nuclear Boiler System Specification, 22A4622, Rev. 5, March 16, 1983.
2. GE, Plant Air Specification, A62-4180, Rev. 0, November 20, 1979.
3. S&W, Nuclear Boiler System Design Freeze, RBV-2054, March 30, 1984.
4. FSAR Section 6.3.3, ECCS Performance Evaluation.

D2.4-4 (Deficiency) ADS Accumulator Check Valve Specifications

BACKGROUND

General Electric Nuclear Boiler System Specification 22A4622 requires that the automatic depressurization system accumulator check valves be provided with "bubble tight" shut off. This is needed to assure that adequate pressure (about 150 psig) remains available since the safety grade compressors, which provide long term supply, only provide about 10、 - 114 psig.

DESCRIPTION

Stone & Webster purchased Velan check valves which were not required to be "bubble tight." These valves were required to pass a hydrostatic seat leakage test in accordance with Manufacturer's Standardization Society Procedure No. MSS-SP-61. Review of test data for one of these valves found that the valve was hydrostatically tested at a pressure of 1500 psig with no seat leakage. This test is not representative of the automatic depressurization system application, which is a pneumatic supply at approximately 100-175 psig. In addition to initial testing, periodic testing should be performed to assure that the check valves retain adequate accumulator pressure. Although the design documentation failed to identify the proper leakage function for these valves, Gulf States Utilities personnel identified this as an item to be resolved in preoperational testing and had not yet established periodic testing requirements.

BASIS

The valve purchase specification did not recognize the required leakage function and allowed a test not representative of the needed function.

IMPACT ON DESIGN

This item should not require hardware changes since proper initial testing and periodic inservice testing can assure that the valves will perform their function. Analysis to determine an allowable leakage rate for proper testing appears necessary.

EXTENT

This deficiency did not appear to be systematic based on review of other lines and systems.

REFERENCES

1. GE, Nuclear Boiler System Specification, 22A4622, Rev. 5, March 16, 1983.
2. MSS-SP-61, Hydrostatic Testing of Steel Valves, 1961 Edition.
3. GSU, Preoperational Acceptance Test Procedure, 1-PT-202, October 10, 1984.
4. GSU, River Bend Problem Report, No. 187, April 20, 1981.
5. IE BULLETIN 80-01, Operability of ADS Valve Pnuematic Supply, January 11, 1980.

D2.4-5 (Deficiency) ADS Pneumatic Supply Instrumentation

BACKGROUND

The automatic depressurization system air supply is measured by pressure instrumentation on each pneumatic supply system header outside the drywell. Control room annunciation of low pneumatic pressure is provided to alert the operator to pneumatic supply failure (References 1 and 2).

DESCRIPTION

This instrumentation is used to determine the availability of a pneumatic supply required for operation of safety related equipment. Accordingly, this is Regulatory Guide 1.97 Type D instrumentation which is subject to Category 2 provisions. Stone & Webster currently has this pressure instrumentation classified as nonsafety and has specified no quality assurance provisions or other Category 2 provisions. River Bend is committed to comply with Regulatory Guide 1.97 (Reference 3).

BASIS

The design is inconsistent with the licensing commitment to meet Regulatory Guide 1.97.

IMPACT ON DESIGN

This item is not likely to require significant hardware changes. However, analysis of the existing equipment is needed to make a determination whether or not hardware changes are needed to meet the licensing commitment. The evaluation should include the need to provide control room indication of automatic depressurization system pneumatic supply pressure.

EXTENT

This deficiency does not appear systematic based on a review of other systems.

REFERENCES

1. GE Nuclear Boiler System Design Specification, 22A4622, Revision 5, Item 4.3.2.4.b.
2. Stone & Webster, Calculation, Main Steam Safety Relief Valve Pneumatic System Piping Design Verification, PN-255, Revision 1.
3. NRC Regulation Guide 1.97, Rev. 3, Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, May 1983.

D2.4-6 (Deficiency) SRV Pneumatic Supply Adequacy

BACKGROUND

Stone & Webster designed the pneumatic supply for the safety/relief valves (SRV's) and automatic depressurization system valves. Six of seven automatic depressurization system valves must operate to depressurize the reactor pressure vessel following small pipe breaks as analyzed in the FSAR Section 6.3

DESCRIPTION

Stone & Webster performed design calculations (PN-255) and concluded that the penetration valve leakage control system was adequate to meet the maximum short and long-term flow requirements for the safety relief valve pneumatic supply system. Calculation PN-255 assumed that each division of the penetration valve leakage control system was available to provide makeup air for 8 safety relief valves. Calculations were not performed for the automatic depressurization system accumulators and no need was identified for use of these accumulators. This is incorrect in that one of the penetration valve leakage control system compressors would be inoperable due to the single failure of an emergency diesel generator as assumed in the accident analysis. In this case the automatic depressurization system accumulators are needed to provide pneumatic supply to the automatic depressurization system valves.

BASIS

The calculation is incorrect and inconsistent with the functional requirements of the accident analysis.

IMPACT ON DESIGN

Design calculations should be performed to demonstrate that the automatic depressurization system accumulators and check valves provide the required pneumatic supply when one penetration valve leakage control system compressor is unavailable. Hardware modifications are not expected to be necessary, pending confirmatory results of revised design calculations and leak rate testing for automatic depressurization system accumulators check valves.

EXTENT

This deficiency did not appear systematic based on review of other systems.

REFERENCES

1. S&W, Calculation, Main Steam Safety Relief Valve Pneumatic System Piping Design Verification, PN-255, Revision 1.
2. FSAR Sections 5.2, 6.3, and 9.3.
3. FSK's-32-8A, 8B, 8C, 8D, 8E.

D2.5-1 (Deficiency) Failure to Assure that Sources of Design Input are Identified in Design Analysis

BACKGROUND

The Engineering Mechanics Division performed detailed calculations to identify the jet load on each individual target in the jet path. These calculations are organized into different calculations according to the target classifications and the cubicle locations. A cubicle is defined as an arbitrarily chosen 90-degree region of the containment. For a given system, the analysis follows the order of the break location number such that the jet impingement affects due to jets initiated for each postulated break location are examined. Each calculation is prepared, checked, independently reviewed, and approved in accordance with EMAG-CH-41-2.

In order to calculate the jet load on a particular target, the jet "footprint" is drawn over a hand sketch of the various targets at a particular elevation. For example, in a jet impingement analysis of structural targets the sketch will include structural walls, structural steel, grating, and pipe restraint beams. These sketches are prepared from production drawings such as piping isometrics, framing drawings, etc.

Although these calculations are organized into different calculations according to the target classifications and the cubicle locations, there is duplication in the jet loads and in the potential targets. This occurs because of the layout of major piping runs such as main steam which is mirrored in another portion of the containment.

DESCRIPTIONS

The source of information used to prepare the sketch of various targets were not consistently identified on the sketch (i.e., the EP, ES, etc.). In general the reference section of these calculations did not reference the production drawings used in the calculation. Likewise, when information from another calculation was used the calculation page usually contained a reference to the source and page number. However, the reference section of the calculation did not consistently identify the source calculation. It was also found that when these calculations were revised (i.e., adding new pages and deleting outdated information) the list of references was not revised accordingly. The following are examples of these deficiencies:

Calculation CBA-1746 referenced calculation CBA-1736 on pages 106, 109, 110, 111, 114, 115, 118, and 119; but calculation CBA-1736 was not identified in the reference section of CBA-1746.

Calculation CBA-1766 identified drawings EP-71B-3, EP-870-5, and EP-74A-4 on target sketches; however, the list of references in the calculation did not identify these drawings.

For calculation CBA-1549 the sketches used in the calculation did not always identify the EP-drawings from which the sketch was prepared. Where references are provided they are not included in the list of references in the calculation.

For calculation CBA-1736 the target steel sketch on page 118 references drawing ES-54E; however, the list of references of the calculation does not identify any drawings.

D2.5-1 (Deficiency) continued.

For calculation CBA-1746 the target steel sketch on page 31 references drawing ES-54E; however, the list of references of the calculation does not identify any drawings.

On page 35b of calculation CBA-1671, Rev. 1 drawing EP-2A-4 is referenced on the target sketch as well as EP-2B-6; however, pages 18 and 19 of the reference section identified EP-2A-3 instead of EP-2A-4. The reference section identifies drawing EP-2B-6 correctly. Since page 35B was entered into the calculation by Revision 1, it appears that when the calculation was revised the list of references was not revised. The reference page was Revision 0 and was not included in the list of revised pages.

BASIS

Stone & Webster Engineering Mechanics Division's procedure EMAG-CH-41 requires drawings and status (including number, revision, and status) be listed in the reference section of calculations.

IMPACT ON DESIGN

This deficiency appears to have limited impact on the design, since it is expected that the production drawings have changed very little since the calculations were done. The impact is one of proper documentation. Failure to reference the drawings or calculation used within a calculation inhibits the ability to revise appropriate portions of the calculation based on revised information. The design traceability is lost.

EXTENT

This deficiency appears to be systematic because of the number of examples found in a limited sample size.

REFERENCES

1. S&W Procedure EMAG-CH-41-2, Preparation, Review, and Control of Manual and Computer Calculations by the Engineering Mechanics Division, 7-7-82. Revision 1, 10-15-79 and Revision 2, 7-1-77.

D2.5-2 (Deficiency) Incorrect Jet Impingement Loading on ASME Class 1 Low Pressure Coolant Injection Line

BACKGROUND

Engineering Mechanics Division has prepared calculations containing the jet load on individual targets in the jet path of a high energy line break. The calculations are organized into different calculations according to the target classification and cubicle location. A cubicle is defined as an arbitrarily chosen 90-degree region of the containment. For a given system, the analysis follows the order of the break location number.

In order to calculate the jet load on a particular target, the jet "footprint" is drawn over a sketch of the various targets at a particular elevation. The total jet force on any cross-section is assumed distance-invariant, with a total magnitude equivalent to the jet thrust force. All potential targets in the jet paths are identified and examined for jet impingement effects. Targets are classified into structural targets, piping and pipe support/restraint targets, equipment targets, and electrical cable tray targets. From these calculations all targets are identified for further evaluation by respective disciplines through the High Energy Line Break Coordinator. The following information is summarized by the target calculations: target designation including identification and general location, the high energy break and location, the thrust coefficient, jet intensity at the target, jet diameter and area of target, and target normal load.

Engineering Mechanics Division's target calculation for main steam line break 30 on loop B was reviewed in detail. At break location 30 both circumferential and longitudinal breaks are postulated. To determine the jet impingement targets on piping and pipe support/restraint Engineering Mechanics Division has prepared two calculations, CBA 1766 Rev. 1 and CBA 1549 Rev. 1. Calculation CBA 1549 supplements the results of CBA 1766 by adding new break locations and deleting other break locations. The results of CBA 1766 and CBA 1549 are summarized in the Jet Impingement/Pipe Rupture Data File. Each calculation is prepared, checked, independently reviewed, and approved in accordance with EMAG-CH-41.

DESCRIPTION

The Jet Impingement/Pipe Rupture Data File dated December 12, 1983 had an incorrect loading for line RHS-010-19-1-Z. This line is the ASME Class 1 line connected to the reactor pressure vessel and functions as a low pressure coolant injection line. Specifically the Jet Impingement/Pipe Rupture Data File indicates that the jet force and the jet impact area of the target are 18.45 kips and 5.6 square feet, respectively, for this particular line. However, the calculated jet impact area is 7.7 square feet which results in a jet force of 33.8 kips. To identify the source of the error calculations CBA-1766 and CBA-1549 were examined in detail as well as the documents used to transmit the calculational results. Calculation CBA-1766 differs from CBA-1549 in the location of the jet footprint. Specifically, calculation CBA-1766 assumed no displacement of the main steam piping connected to the reactor vessel, while calculation CBA-1549 assumed that the pipe moved 24 inches. In CBA-1766 the "footprint" of the jet intersects with only line RHS-010-19-1(Z); however, by moving the main steam line 24 inches the "footprint" also intersects line RHS-010-16-1(B). DEM-1663 documented the revision to the piping targets and the revised loads on lines RHS-010-016-1(B) and RHS-010-019-1(Z). It appears that the inconsistency between the calculated jet loading on line RHS-010-19-1(Z) and the value in the Jet Impingement/Pipe Rupture Data File was caused by incorrect input of data into the computer data base.

D2.5-2 (Deficiency) continued.

BASIS

The Jet Impingement/Pipe Rupture Data File had a jet impingement load on line RHS-010-19-1(Z) inconsistent with the value calculated in CBA-1549.

IMPACT ON DESIGN

The jet impingement load on line RHS-010-19-1(Z) is 45% higher than that identified in the Jet Impingement/Pipe Rupture Data File. This impingement load appears to have no affect on the physical design of line RHS-010-19-1(Z), because line RHS-010-16-1(B) has the same safety function and has a 4% higher jet impingement load.

EXTENT

The Jet Impingement/Pipe Rupture Data File identifies seven targets associated with breaks at location 30 of main steam line loop B. Four of these are pipe supports for which no loading is provided. Of the other three piping targets identified only RHS-010-19-1(Z) had an incorrect value for the jet impingement. Since the sample size is small, it is not possible to conclude that the deficiency is systematic.

REFERENCES

1. S&W Calculation CBA-1766, Rev. 1, Jet Impingement On Piping Targets Inside Drywell, dated September 29, 1981.
2. S&W Calculation CBA-1549, Rev. 1, Jet Impingement Loads On Piping Targets (This calculation supplements CBA-1766 and CBA-1777), dated August 19, 1982.
3. S&W Jet Impingement/Pipe Rupture Data File (Special Sort For Power Division) dated December 19, 1983, page 44.
4. S&W Interoffice Memorandum DEM-1663, Jet Load On Revised Piping Targets (Jet Loads on RHS-010-16-1(B) and RHS-010-19-1(Z) from MS Line B Break 30C), dated June 18,1982.

D2.5-3 (Deficiency) Field Configuration of Structural Components Differs from that used in Engineering Mechanics Division's Jet Impingement Target Calculations

BACKGROUND

The Engineering Mechanics Division has performed detailed calculations to identify the jet load on each individual target in the jet path. These calculations are organized into different calculations according to the target classifications and the cubicle locations. A cubicle is defined as an arbitrarily chosen 90-degree region of the containment. For a given system, the analysis follows the order of the break location number such that the jet impingement effects due to jets initiated from each postulated break location are examined. Each calculation is prepared, checked, independently reviewed, and approved in accordance with EMAG-CH-41-2.

In order to calculate the jet load on a particular target, the jet "footprint" is drawn over a hand sketch of the various targets at a particular elevation. For example in a jet impingement analysis of structural targets, the sketch will include structural walls, structural steel, grating, and pipe restraint beams. These sketches are prepared from structural production drawings.

Targets are identified from these calculations for further evaluation by respective disciplines through the High Energy Line Break Coordinator. The following information is summarized by the target calculations: target designation including identification and general location, the high energy break and location, the thrust coefficient, jet intensity at the target, jet diameter and area of target, and target normal load.

To assess Engineering Mechanics Division's accuracy in the preparation of target packages, Engineering Mechanics Division's target calculation for main steam line break 30 on loop B was reviewed in detail and a site survey conducted. At break location 30 both circumferential and longitudinal breaks are postulated. To determine the jet impingement targets on structural targets, Engineering Mechanics Division has prepared two calculations, CBA 1746 Rev. 1 and CBA 1860 Rev. 1. Calculation CBA 1746 is the jet impingement analysis performed to identify structural targets within cubicle 3 inside containment. Calculation CBA 1860 incorporates all jet impingement loads on structural components within one calculation.

DESCRIPTION

The structural steel configuration used in Engineering Mechanics Division's structural target calculations CBA 1746 Rev. 1 and CBA 1860 Rev. 1 do not reflect the actual installed condition within the drywell. The structural steel targets impinged by jets from postulated circumferential and longitudinal breaks at location 30 of main steam line loop B differ from those identified by Engineering Mechanics Division in their structural target calculations. The following are examples of these differences:

At elevation 118 ft-5/6 inches Engineering Mechanics Division's sketch of structural targets identify that only one structural member is impacted by the jet. The structural member is identified as a W 18x97. However, field inspection revealed that the structural member is actually a W 18x119 and that two other members are within the jet steam. These structural members are both W 10x45. The review of the latest structural drawing confirms the field observation. It appears that the sketch in the calculation was prepared from a revision of the structural drawing which has been superceded/revised since the calculation was prepared. Specifically, the target sketch at this particular elevation identifies ES-54A-sketch as the reference drawing, while the actual plant condition is reflected on drawing ES-54H-3.

D2.5-3 (Deficiency) continued.

At elevation 139 ft.-5 inches for a circumferential break, Engineering Mechanics Division's sketch of structural targets identify the following structural members within the jet stream: W 14x61, W 12x26, L 3 1/2 x 2 1/2 x 3/8 and a pipe restraint beam located under W 14x61. However, field inspection revealed that the structural members actually within the jet stream are as follows: W 14x90, W 14x82, MC 18x53, 2L 4 x 4 x 3/8, and a pipe restraint beam located under W 14x61. A review of the lastest structural drawing confirms the field observation. The sketch in the calculation was prepared from Revision 2 of ES-54A, while the current revision is Revision 4.

At elevation 139 ft.-5 inches for a longitudinal break, Engineering Mechanics Division's sketch of structural targets identify the following structural members within the jet stream: W 18x119, W 12x26, two L 3 1/2 x 2 1/2 x 3/8, and W 14x51. However, field inspection revealed that the structural members actually within the jet stream are as follows: W 18x119, W 24x162, and a W 14x90. A review of the latest structural drawing confirms the field observation. The sketch in the calculation was prepared from revision 2 of ES-54A, while the current revision is Revision 4.

Calculations of piping and equipment targets accurately reflected the as constructed condition. However, it was noted that for this particular break location the number of non-structural targets are few. (This comment does not pertain to targets such as small bore piping, instrument lines, and electrical conduit because these types of targets are not identified by Engineering Mechanics Division calculation. Instead a field walkdown is proposed to identify these types of targets). Based upon the discrepancies observed, it appears that when major changes occurred to the structural steel within the drywell sketches within the calculation were not revised.

Engineering Mechanics Division personnel were asked to explain why the structural target calculations do not reflect the as constructed condition. They indicated that they were aware of the structural steel changes because they are on distribution for all structural steel drawings. In explaining why the discrepancies exist, Engineering Mechanics Division personnel stated that they probably made an engineering judgement that the changes were not sufficient to warrant a revision to the calculation. Engineering Mechanics Division personnel also stated that changes to structural targets did not receive as high a priority as changes to piping and equipment targets.

BASIS

Calculation CBA 1746 states that "the objective of this analysis is to supply the jet impingement load and jet intensity on structures inside cubicle 3 that have changed from previous calculations and to list all loading on all structural targets for all postulated breaks for all piping systems." Calculation CBA 1860 has a similar objective. Since the structural steel arrangement used in these calculations differ from the as constructed configuration, these calculations do not fulfill their stated objective. In addition PMM-152 states that Engineering Mechanics Division shall transmit to the High Energy Line Break Coordinator a table and/or listing of the numbered break locations and associated pipe whip and jet impingement targets. It further requires that Engineering Mechanics Division provide the resulting pipe whip and/or jet impingement loads imposed on the target so that the structural integrity of the targets can be evaluated. Engineering Mechanics Division has provided the High Energy Line Break Coordinator a structural target identification package which was based on the results of calculations CBA 1746 and CBA 1860.

D2.5-3 (Deficiency) continued.

Because the structural targets identified by do not reflect the as constructed condition, Engineering Mechanics Division has not fulfilled the requirements of PMM-152.

IMPACT ON DESIGN

Although the structural steel arrangement in the calculations differs from the as constructed configuration, this item appears to have no impact on the design. This judgement is based upon our observation of how the Structural Division uses Engineering Mechanics Division's target identification package. Specifically the Structural Division does not use the targets or loads calculated by Engineering Mechanics Division, instead they use the "footprint" and jet intensity at a particular elevation to load the structural members within the jet path.

EXTENT

Although the team selected only one break location to examine in detail, the nature and extent of the discrepancies indicate that calculated structural targets within the containment do not reflect the as constructed condition. The team believes that the discrepancies found at this particular break location are systematic and indicate a failure to insure that design analysis fulfills its intended objective throughout the plant design and construction.

REFERENCES

1. S&W Procedure EMAG-CH-41-2, Preparation, Review, and Control of Manual and Computer Calculations by the Engineering Mechanics Division, dated July 7, 1982.
2. EMD Calculation CBA 1746, Rev. 1.
3. EMD Calculation CBA 1860, Rev. 1, dated February 24, 1983.
4. S&W Structural Drawing ES-54H-3.
5. S&W Structural Drawing ES-54A-4, dated January 12, 1982.
6. S&W Project Management Memorandum PMM-152, Rev. 2, High Energy Line Break (HELB) Evaluation Procedure, dated April 6, 1984.

D2.5-4 (Deficiency) Failure to Translate FSAR Commitments into High Energy Line Break Evaluation Procedure

BACKGROUND

Since January 1984 the Power Division of the Stone & Webster River Bend project has been performing evaluations to confirm that the plant can be brought to a cold shutdown condition following postulated high energy line breaks coincident with a loss of offsite power and a worst-case single active failure. The evaluation consists of a review of target structures and components impacted by a whipping pipe and/or jet of steam or water. Under the direction of the High Energy Line Break Coordinator the Power Division has been reviewing pipe whip and jet impingement targets with respect to the acceptance criteria documented in Project Management Memorandum (PMM) 152. With the acceptability of a target hit determined, the output of Power Division's evaluation is used by Engineering Mechanics Division to determine if more refined stress analysis is required to eliminate a postulated pipe break (i.e., the stress levels at the break location are below those requiring a break to be postulated). For acceptable targets and for unacceptable targets from break locations which cannot be eliminated, Engineering Mechanics Division performs analysis of the affects of calculated jet impingement loading and pipe whip impacts to determine the consequences.

DESCRIPTION

Prior to the inspection cutoff date (i.e., February 17, 1984) the High Energy Line Break Evaluation Procedure did not identify the following requirements:

The procedure did not include Containment Isolation piping and equipment as a system requiring protection from high energy line breaks.

The procedure did not include pipe supports of safe shutdown and containment isolation systems as needing protection from high energy line breaks.

The procedure did not include a requirement to consider the potential for another high energy line to be ruptured as a consequence of the postulated high energy line break.

BASIS

FSAR section 3.6A describes the protection against effects associated with the postulated rupture of piping within the Stone & Webster scope of supply. In this section the Applicant stated the objectives for protection against pipe failure effects. These objectives are:

To assure that the reactor can be shut down safely and can be maintained in a safe cold shutdown condition or that the consequences of a LOCA can be mitigated.

To assure that containment integrity is maintained.

To assure that a pipe break which is not a loss of reactor coolant does not cause a loss of reactor coolant.

To assure that the radiological doses resulting from a postulated piping failure remain below the limits of 10CFR100.

D2.5-4 (Deficiency) continued.

To assure that the consequences of the postulated piping failure can be mitigated considering a single active component failure.

In summary the High Energy Line Break Evaluation Procedure as documented by PMM-152 Rev. 0 was insufficient to meet the FSAR commitments of Section 3.6A.

IMPACT ON DESIGN

This item appears to have little impact on design, since the High Energy Line Break Evaluation Procedure was revised to incorporate the missing requirements on March 12, 1984. However, if Revision 0 of PMM-152 was used to evaluate targets affected by postulated high energy line breaks, it is possible that unacceptable target/break interactions may have been overlooked.

EXTENT

This deficiency in the High Energy Line Break Evaluation Procedure could have resulted in evaluations which would not have fulfilled the Applicant's commitments in the FSAR. However incorporation of the missing requirements coupled with a verification that unacceptable target/break interactions were not overlooked should minimize the potential for errors.

POST CUTOFF WORK

On March 12, 1984 PMM-152 was revised to incorporate the missing requirements. The deficiency in the High Energy Line Break Evaluation Procedure was identified by the Stone & Webster and corrected prior to the inspection team's arrival at the River Bend site.

REFERENCES

1. S&W Project Management Memorandum PMM-152, Rev. 0, High Energy Line Break (HELB) Evaluation Procedure, dated January 4, 1984.
2. FSAR Section 3.6A, Protection Against Affects Associated With The Postulated Rupture Of Piping (S&W Scope of Supply).
3. S&W Project Management Memorandum PMM-152, Rev. 1, High Energy Line Break (HELB) Evaluation Procedure, dated March 12, 1984.

D2.5-5 (Deficiency) Errors and Inconsistencies in HELB Evaluations

BACKGROUND

Since January 1984 the Power Division of Stone & Webster's River Bend project has been performing evaluations to confirm that the plant can be brought to a cold shutdown following postulated high energy line breaks coincident with a loss of offsite power and a worst-case single active failure. The evaluations consist of a review of target structures and components impacted by a whipping pipe and/or jet of steam or water. Under the direction of the High Energy Line Break Coordinator the Power Division has been reviewing pipe whip and jet impingement targets with respect to the acceptance criteria documented in Project Management Memorandun (PMM) 152.

The Engineering Mechanics Division is responsible for transmitting to the High Energy Line Break Coordinator a listing of the high energy line break locations and associated pipe whip and jet impingement targets. The targets identified are structures (e.g., structural steel, grating, stairs, walls, pipe restraint beams, etc.), equipment, valves, dampers, piping, ventilation ductwork, electrical cable trays, and instruments. For each target Engineering Mechanics Division provides the resulting pipe whip and/or jet impingement loads imposed on the target. These targets are grouped into target identification packages.

The High Energy Line Break Coordinator is responsible for transmitting a copy of the target identification packages to Electrical, Power (Building Services), and Structural Divisions. For each high energy line break location each Division is responsible for evaluating the acceptability of a target/break interaction using the guidance of PMM-152 for those identified targets within its scope of responsibility. For identified mechanical equipment and instrument targets, the High Energy Line Break Coordinator is responsible for evaluating the acceptability.

Once the unacceptable target/break interactions have been identified, the High Energy Line Break Coordinator is responsible for performing a preliminary review to determine whether other acceptable means are available to achieve a safe shutdown for each pipe break.

Although not called out by PMM-152, Engineering Mechanics Division is also examining whether the break can be eliminated for those situations where unacceptable target/break interactions are identified. For piping and pipe supports targets impacted by jet impingement, the High Energy Line Break Coordinator has completed his evaluation in accordance with PMM-152 and has transmitted this information back to Engineering Mechanics Division. If the break cannot be eliminated, then Engineering Mechanics Division intends to perform analysis to ensure that the impacted piping targets can withstand the jet impingement without the loss of functionality. For acceptable target/break interactions, Engineering Mechanics Division intends to perform analysis to ensure that the impacted piping target does not fail (e.g., for high energy line targets, rupture and add to the hostile environment) or become a missile (i.e., capable of withstanding the jet impingement but functionality not assured).

The inspection team examined the results of the High Energy Line Break Coordinator's evaluation of the pipe and pipe support jet impingement target/break interactions to evaluate the quality of the work performed to date and to assess the adequacy of the guidance of PMM-152.

D2.5-5 (Deficiency) continued.

DESCRIPTION

On March 27, 1984 Power Division provided their evaluations of jet impingement on piping and pipe supports by informal interoffice correspondence to Engineering Mechanics Division (see Deficiency D2.5-7). This transmittal included a marked up copy of the Jet Impingement/Pipe Rupture Data File dated December 12, 1983. Additions and deletions from data sheets were transmitted from Engineering Mechanics Division to Power Division by informal interoffice correspondence on March 15, 1984 (see Deficiency D2.5-7). The purpose of this information transmittal was to identify to Engineering Mechanics Division the break locations which result in unacceptable jet impingement interactions on piping or pipe supports required for safe shutdown. The team was informed by Stone & Webster staff that PMM-152 was used to determine whether a pipe or pipe support was unacceptable. The informal interoffice correspondence of March 27, 1984 identified 19 high energy lines for which postulated breaks result in unacceptable target interactions with piping or pipe supports having a safe shutdown function.

The team examined the evaluations concerning the pipe and pipe supports impacted by a jet from break location 30 of main steam line loop B to assess their technical adequacy. This particular line was not identified in Power Division's summary of high energy lines for which postulated breaks result in unacceptable target interactions with piping or pipe supports with a safe shutdown function. The Jet Impingement/Pipe Rupture Data File used by Power Division identified 6 piping and pipe support targets impacted by a jet due to a circumferential break of main steam loop B at location 30 inside the drywell. All 6 were identified as unacceptable targets based upon the criteria established in PMM-152. In addition, Power Division augmented their assessment to indicate whether the particular target was required for safe shutdown, containment isolation, and/or environment. The latter category reflects that the impacted target is a pipe or pipe support associated with another high energy line whose failure could worsen the harsh environment caused by the postulated break. The following is a listing of these targets and Power Division's assessment of each:

Target I.D.	Target Type	Remarks
RDS Lines	Piping	Safe shutdown
RHS-010-16-1-B	Piping	Environment, Containment Isolation
RHS-010-19-1-Z	Piping	Environment, Containment Isolation
H101B	Pipe Support	Environment
S101B	Pipe Support	Environment
S102B	Pipe Support	Environment

The RDS lines are control rod drive lines which are required for safe shutdown. The pipe support targets are supports for main steam line loop A within the drywell and do not perform a safe shutdown or containment isolation function. However, the two Reactor Heat Removal System lines are ASME Class 1 low pressure coolant injection lines which are an integral part of the emergency core cooling system. For a postulated main steam line break inside the drywell, assuming a loss of offsite power and single failure of Division I diesel, at least one of these two impacted lines is required to meet the minimum requirements prescribed by General Electric for safe plant shutdown.

D2.5-5 (Deficiency) continued.

The following summarizes the deficiencies found in the evaluation performed by Power Division:

Reactor heat removal system piping required to remain functional to meet minimum safe shutdown requirements were not properly categorized.

Main steam line loop B was not identified as a high energy line which had an unacceptable target/jet interaction, even though RDS lines were correctly identified as being required for safe shutdown.

Power Division's evaluation of the same target in other break locations are not consistent. For example at break location 35E of feedwater line FWS-012-36-1-B inside the drywell, pipe supports for RHS-010-16-1-B and RHS-010-19-1-Z are identified as jet impingement targets. In this instance these targets are categorized as required for safe shutdown; however, they are not also categorized as required for containment isolation or environment as they were for the main steam line break.

Based upon the deficiencies found in Power Division's evaluations, it is the inspection team's assessment that the High Energy Line Break Evaluation Procedure is deficient in the following areas:

PMM-152 does not provide sufficient guidance to individuals performing evaluations to correctly categorize unacceptable target/break interactions into groups such as required for safe shutdown, containment isolation, environment, etc.

The evaluations performed by Power Division were not checked or reviewed, even though another design group was using these evaluations to conduct additional design work.

PMM-152 states that the High Energy Line Break Coordinator will review the unacceptable items in the target identification sheets and perform a preliminary review to determine whether other acceptable means are available to achieve a safe shutdown for each pipe break. PMM-152 does not identify what criteria the High Energy Line Break Coordinator will used to make this determination. It is the team's assessment that the current acceptance criteria of PMM-152 is very conservative and will result in numerous unacceptable target/break interactions. Consequently it is expected that the High Energy Line Break Coordinator will be required to review a large number of plant faulted conditions in an effort to determine whether the plant is designed to mitigate the consequences of those conditions. As a minimum PMM-152 should reference the minimum requirements outlined in General Electric's design specification on mechanical equipment separation for engineered safety features.

BASIS

Gulf States Utilities has committed to follow ANSI N45.2.11 which requires that design activities be prescribed and accomplished in accordance with procedures of a type sufficient to assure that applicable design inputs are correctly translated into specifications, drawings, procedures or instructions. It further requires that design activities be documented in sufficient detail to permit verification and auditing.

D2.5-5 (Deficiency) continued.

Because the evaluations currently being performed are verifying that the River Bend plant is designed to withstand high energy line breaks, these evaluations should be classified as design analysis. As such ANSI N45.2.11 requires that they be performed in a planned, controlled and correct manner. In addition, the standard requires that design analysis be in a form suitable for reproduction, filing and retrieving and be sufficiently detailed as to purpose, method, assumptions, design input, references such that a person technically qualified in the subject can review and understand the analyses and verify the adequacy of the results without recourse to the originator.

IMPACT ON DESIGN

This item appears to have no impact on design because the evaluation of target/break interactions only began in February of 1984. However, the evaluations performed thus far may have to be redone and will require proper documentation.

EXTENT

The type of errors found in the evaluation of jet impingement target/break interactions coupled with the lack of formal documentation and checking imply that this evaluation as a whole is systematically deficient. The procedural deficiencies are unique to the High Energy Line Break Evaluation Procedure.

REFERENCES

1. S&W Interoffice Correspondence from J. Zaccaria to H. Kuo, Subject: Jet Impingement/Pipe Rupture Data Sheets, March 27, 1984.
2. S&W Jet Impingement/Pipe Rupture Data File (Special Sort For Power Division), December 19, 1983.
3. S&W Interoffice Correspondence from Engineering Mechanics Division to Power Division, March 16, 1984.
4. PMM-152, Rev. 2, High Energy Line Break (HELB) Evaluation Procedure, April 6, 1984.
5. ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants.
6. Regulatory Guide 1.64 Rev. 2, Quality Assurance Requirements for the Design of Nuclear Power Plants, June 1976.
7. General Electric Design Specification, Document No. 22A7193 Rev. 0, Mechanical Equipment Separation For Engineered Safety Features, September 30, 1980.

D2.5-6 (Deficiency) Design Control of Input to High Energy Line Break Evaluation

BACKGROUND

Since January 1984 the Power Division of the Stone & Webster River Bend project has been performing evaluations to confirm that the plant can be brought to a cold shut-down condition following postulated high energy line breaks coincident with a loss of offsite power and a worst-case single active failure. The evaluation consists of a review of target structures and components impacted by a whipping pipe and/or jets of steam or water.

DESCRIPTION

Under the direction of the High Energy Line Break Coordinator, the Power Division has been reviewing pipe whip and jet impingement targets with respect to the acceptance criteria documented in Project Management Memorandum (PMM) 152. To compare the acceptance criteria to the postulated high energy line break effects, the pipe whip and jet impingement targets are categorized into safe shutdown configurations based upon diesel-load configurations. These safe shutdown configurations, including associated essential support systems, can be categorized as:

a. 1 Low Pressure Coolant Injection loop (with heat exchanger) and the low pressure core spray system connected to a single diesel generator.

b. 2 additional Low Pressure Coolant Injection loops (1 loop with heat exchanger) connected to a single diesel generator.

c. The high pressure core spray system connected to a single diesel generator.

The Stone & Webster system flow diagrams are used to identify the proper safe shutdown configuration of each targeted component. Given the target category and break location, a determination is made as to the acceptability of losing the identified targets. The sources of input information for the ongoing evaluations are not being recorded. Specifically, the date/revision number of applicable system flow diagrams used to determine the safe shutdown configurations are not being recorded.

BASIS

Gulf States Utilities has committed to follow ANSI N45.2.11 which requires that design activities be prescribed and accomplished in accordance with procedures of a type sufficient to assure that applicable design inputs are correctly translated into specifications, drawings, procedures or instructions. It further requires that design activities be documented in sufficent detail to permit verification and auditing as required by the standard. Because the evaluations currently being performed are verifying that the River Bend plant is designed to withstand high energy line breaks, these evaluations should be classified as design analyses. As such ANSI N45.2.11 requires that they be performed in a planned, controlled and correct manner. In addition, the standard requires that design analyses be in a form suitable for reproduction, filing and retrieving and be sufficiently detailed as to purpose, method, assumptions, design input, references such that a person technically qualified in the subject can review and understand the analyses and verify the adequacy of the results without recourse to the originator.

D2.5-6 (Deficiency) continued.

IMPACT ON DESIGN

This item had no impact on existing design or analyses. It is a documentation problem that will render it difficult or impossible to maintain the high energy line break evaluation correct and current as design changes are made.

EXTENT

The deficiency appears to be systematic based upon review of the evaluations currently being performed by the Power Division and discussions with responsible personnel.

REFERENCES

1. S&W Project Management Procedure PMM-152, Rev. 2, High Energy Line Break (HELB) Evaluation Procedure, April 6, 1984.
2. ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants.
3. Regulatory Guide 1.64 Rev. 2, Quality Assurance Requirements for the Design of Nuclear Power Plants, June 1976.

D2.5-7 (Deficiency) Failure to Control the Flow of High Energy Line Break Design Information between Internal Design Groups

BACKGROUND

The Power Division has been reviewing pipe whip and jet impingement target with respect to the acceptance criteria documented in Project Management Memorandum (PMM) 152 since January 1984.

Input for this evaluation has come from Engineering Mechanics Division. This input identifies the postulated break locations and the targets affected by either jet impingement or pipe whip. The input was prepared by Engineering Mechanics Division by summarizing results of reviewed and approved jet impingement and pipe whip target calculations. In the case of pipe and pipe support/restraints targets, the jet impingement information was input into a jet impingement/pipe rupture data file. The input process is a controlled process to insure that calculation results are correctly reflected by the data base.

Under the direction of the High Energy Line Break Coordinator, the Power Division used the Engineering Mechanics Division input to compare the pipe whip and jet impingement targets by break location to the acceptance criteria documented in Project Management Memorandum (PMM) 152. With the acceptability of a target hit determined, the output of Power Division's evaluation is used by Engineering Mechanics Division to determine if a more refined stress analysis is required to eliminate a postulated pipe break (i.e., the stress levels at the break location are below those requiring a break to be postulated). For acceptable targets and for unacceptable targets from break locations which cannot be eliminated, Engineering Mechanics Division performs analysis of the affects of calculated jet impingement loading and pipe whip impacts to determine the consequences.

DESCRIPTION

The following deficiencies were noted in the internal interfaces between Power Division and Engineering Mechanics Division:

The High Energy Line Break Coordinator, a Power Division engineer, received a jet impingement/pipe rupture data file sort from Engineering Mechanics Division by informal means about the second week of January 1984. This copy of the data file sort was used by Power Division to evaluate the degree of acceptability of the jet impingement targets.

On March 12, 1984, Engineering Mechanics Division included a revised jet impingement/pipe rupture data file sort as an attachment to an informal interoffice correspondence. This copy of the data file was used by Power Division to supplement the information received previously from Engineering Mechanics Division.

For pipe whip targets and jet impingement targets other than piping and pipe supports, Engineering Mechanics Division gave the High Energy Line Break Coordinator a copy of their original books summarizing break and targets. This information was provided by informal means.

On March 16, 1984, Engineering Mechanics Division included revised break and target information for pipe whip and jet impingement other than piping and pipe supports as an attachment to an informal interoffice correspondence.

D2.5-7 (Deficiency) continued.

On March 27, 1984, Power Division provided their evaluations of jet impingement on piping and pipe supports by informal interoffice correspondence to Engineering Mechanics Division. As indicated in the background Engineering Mechanics Division was using that information to direct further design analysis.

BASIS

Gulf States Utilities has committed to follow ANSI N45.2.11 which defines internal design interface as the relationship between design groups or organizations within a company. This standard requires that design information transmitted from one organizational unit to another be documented and controlled. It further requires that the status of the design information or document be identified. Where necessary incomplete items which require further evaluation, review or approval are to be identified. When it is necessary to initially transmit design information orally or by other informal means, the transmittal should be confirmed promptly by a controlled document. The oral and informal interoffice correspondence internal interfaces between Engineering Mechanics Division and Power Division did not fulfill the requirements of ANSI N45.2.11. In addition, the interoffice correspondence used by Engineering Mechanics Division and Power Division did not comply with the requirements of River Bend Procedure 6.14-0. This River Bend procedure defines the minimum requirements which are imposed on memoranda (Interoffice Correspondence, Interoffice Memorandums, etc.) used by project Engineering Department personnel to transmit/request design/engineering data either between groups within one division or between divisions. Engineering Mechanics Division and Power Division did not comply with the requirements of the procedure concerning requesting and providing design/engineering data.

IMPACT ON DESIGN

This deficiency in the control of design information between Engineering Mechanics Division and Power Division can result in inadequate design provisions to mitigate the consequence of a high energy line break inside and outside of containment. Specifically the use of informally obtained design information in design analysis affecting quality can result in the use of erroneous input and assumptions.

EXTENT

The deficiency in the control of design information between Engineering Mechanics Division and Power Division is systematic. The incidents of informal transmittal of design information between Engineering Mechanics Division and Power Division represent a significant weakness in the design control process associated with the analysis of high energy line breaks. The incidents listed represent all the informational exchanges between the internal design groups from mid-January 1984 through late March 1984.

REFERENCES

1. S&W Interoffice Correspondence from Engineering Mechanics Division to Power Division, March 12, 1984.
2. S&W Interoffice Correspondence from Engineering Mechanics Division to Power Division, March 16, 1984.
3. S&W Interoffice Correspondence from J. Zaccaria to H. Kuo, Subject: Jet Impingement/Pipe Rupture Data Sheets, March 27, 1984.

D2.5-7 (Deficiency) continued.

4. ANSI N45-2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants.
5. Regulatory Guide 1.64 Rev. 2, Quality Assurance Requirements for the Design of Nuclear Power Plants, June 1976.
6. S&W River Bend Procedure RBP 6.14-0, Procedure for Communication of Design/Engineering Data, August 20, 1979.

D2.7-1 (Deficiency) Interim Problem Report

BACKGROUND

Stone & Webster has established a problem reporting system for the review of operating reactor experience data and the resolution of problems which may affect Stone and Webster projects. Potential problem areas are identified in Interim Problem Reports issued by Engineering Assurance to the applicable projects for resolution.

DESCRIPTION

Interim Problem Report 50978 was issued by Stone & Webster Engineering Assurance on IE Information Notice 83-26, "Failure of Safety/Relief Valve Discharge Line Vacuum Breakers." This Interim Problem Report was issued December 13, 1983 and requested that the River Bend Project respond by January 12, 1984 with planned resolution of vacuum breaker problems. During the inspection, Stone & Webster engineers determined that the requested response had not been provided, but Stone & Webster had corresponded with Gulf States Utilities on the adequacy of these vacuum breakers. The vacuum breaker failures noted in IE Information Notice 83-26 had not been addressed. Gulf States Utilities had entered IE Information Notice 83-26 into its Problem Reporting System and was tracking resolution. However, prior to the inspection cutoff date, the River Bend vacuum breakers were not analyzed in detail to account for the failures noted in IE Information Notice 83-26. In 1982, Gulf States Utilities had stated that these valves would be monitored in service to ensure their operability. During discussions with Gulf States Utilities and Stone & Webster personnel no plans were found to implement this monitoring. An inspection of these valves found no special design features which would cushion the loading to the hinge pin area during valve operation.

BASIS

The response requested by Interim Problem Report 50978 was not provided.

IMPACT ON DESIGN

The design adequacy of the Velan check valve vacuum breakers has not been demonstrated by a detailed stress analysis for all applicable loadings. Adequacy of these valves is questionable considering failures which have occurred to date, lack of stress analysis, and lack of test data applicable to in plant service conditions. Failure of these valves could result in the valves sticking either open or closed. If the valves stick closed, they could not perform their safety function. If the valves stick open, leakage of steam into the drywell could complicate plant events with safety relief valve operation.

EXTENT

This deficiency does not appear systematic. Stone & Webster and Gulf States Utilities have established programs for evaluating and resolving problems identified by the NRC and the Institute of Nuclear Power Operations. One other deficiency was found in Stone & Webster's response to IE Bulletin 80-01, "Operability of ADS Valve Pneumatic Supply" (see Deficiency D2.7-2).

POST CUTOFF WORK

Since the inspection cutoff date, both Stone & Webster and Gulf States Utilities have documented an evaluation to resolve IE Information 83-26 (References 4 & 5).

D2.7-1 (Deficiency) continued.

Both the Stone & Webster and Gulf States Utilities evaluations are deficient in that they considered only failures of valves manufactured by GPE. As stated in the notice, valves manufactured by Anderson Greenwood have also failed. Also the Gulf States Utilities and Stone & Webster evaluations consider only hinge pin diameter and bearing adequacy. Velan submitted rough calculations for stresses from loads due to disk slamming. A broader evaluation of the valve adequacy was not performed.

REFERENCES

1. S&W Interim Problem Report 50978, Failure of Safety/Relief Valve Discharge Line Vacuum Breakers, December 13, 1983.
2. IE Information Notice 83-26, Failure of Safety/Relief Valve Discharge Line Vacuum Breakers, May 3, 1983.
3. GSU letter from J. Glazer to J. Kirkebo, Stone & Webster, RBG-12, 494, April 23, 1982.
4. GSU letter from R. Helmrick to J. Kirkebo, Stone & Webster, RBG-17,638, April 19, 1984.
5. S&W Interoffice Memorandum, April 30, 1984, response to IRP-50978.
6. S&W letter from J. Kirkebo to W. Cahill, GSU, December 15, 1981.
7. S&W letter from J. Kirkebo to J. Glazer, GSU, March 24, 1982.
8. S&W letter from J. Kirkebo to R. Helmrick, December 15, 1983.

D2.7-2 (Deficiency) IE Bulletin 80-01

BACKGROUND

The team reviewed Stone & Webster and Gulf States Utilities responses to IE Bulletin 80-01, "Operability of Automatic Depressurization System Valve Pneumatic Supply." This bulletin describes misapplication of a check valve and lack of periodic leak testing for the automatic depressurization system accumulators. This bulletin was selected because it related directly to automatic depressurization system design and also provided a check on the use of the problem reporting systems established by Stone & Webster and Gulf States Utilities.

DESCRIPTION

The team was informed that Stone & Webster had issued an Interim Problem Report for information but had taken no other action in response to IEB 80-01. We found that Gulf States Utilities had obtained a copy of IEB 80-01 from an information service because IEB 80-01 was not issued to plants under construction. Gulf States Utilities had not taken or planned to take any action in response to IEB 80-01. Gulf States Utilities personnel in pre-operational and startup testing had identified the need to leak test the check valves during pre-operational testing. The team discussed the question of NRC Bulletins which were not sent to construction plants with the NRC Resident Inspector-Operations. He indicated that a review of all bulletins was being performed for applicability. For bulletins applicable to River Bend Gulf States Utilities would be requested to consider the issues involved regardless of whether or not the particular bulletin was sent to Gulf States Utilities.

BASIS

The problem reporting systems established by Gulf States Utilities and Stone and Webster did not identify the need to evaluate an IE Bulletin directly applicable to the automatic depressuration system design for the River Bend Project. The design process did not identify the need to periodically test the automatic depressuration system accumulator check valves.

IMPACT ON DESIGN

This deficiency has no direct impact on hardware but Stone & Webster should evaluate IEB 80-01 to ensure that the automatic depressuration system design does not involve misapplication of a check valve and to establish leak testing requirements and criteria for the automatic depressuration system accumulators.

EXTENT

This deficiency does not appear systematic based on additional review of Stone and Webster and Gulf States Utilities response to other IE Bulletins. This additional review shows that Stone & Webster and Gulf States Utilities have established problem reporting systems. The Gulf States Utilities problem reporting system was recently modified and improvement is expected.

D2.7-2 (Deficiency) continued.

REFERENCES

1. NRC, IE Bulletin 80-01, Operability of ADS Valve Pneumatic Supply, January 11, 1980.
2. GSU, River Bend Problem Report, No. 187.
3. GSU, Letter to B.G. Schultz (S&W), RBG-11,010, August 18, 1981.

D3.3-1 (Deficiency) Control of Ball Joint Rotation

BACKGROUND

Three ball joints are installed in each of the sixteen safety/relief valve discharge lines to provide greater flexibility to the discharge line and to reduce the safety/relief valve nozzle loads to acceptable levels. The ball joints in the safety/relief valve discharge lines have a limited range of flexural movement (7.5° of arc) within which they behave as flexible joints. Controls must be established for design evaluation and construction to assure this range is not exceeded. Without controls the limit of flexural movement could be exceeded and excessive bending moments could be applied to the connected safety/relief valves.

DESCRIPTION

The transmittal from power engineering (Reference 1) did not provide the piping stress analysis engineer any reference to ball joint information on limits of angular movement of the ball joints. Thus there was no request for feedback of the predicted movements. Construction Specification No. 228-160, also from power engineering, permits misalignment of 2.5° in the joints during erection. This leaves 5° for predicted movement. The piping analysis (Reference 3) did not include evaluation of the predicted movement relative to any limit.

BASIS

A significant limit on capability of ball joints in piping was not subject to adequate control in design evaluation.

IMPACT ON DESIGN

Initial attempts at evaluation of the range of ball joint rotation indicated that the allowable rotation is exceeded. Any resulting increase in rigidity caused by exceeding the limit could have significant impact on design of the discharge lines.

EXTENT

This deficiency is presumed to extend to all 16 safety/relief valve discharge lines which contain 48 ball joints.

POST CUTOFF WORK

Some reanalysis has been performed with improved modeling. The revised predictions were not reviewed during the inspection.

REFERENCES

1. S&W I.O.C. DP-760, Boothby to SHAH, 4-8-80.
2. S&W Field Fabrication and Erections Spec No. 228.160.
3. S&W Calculation No. 12210-AX-2E-1, 7/20/82.

D3.3-2 (Deficiency) Flow Meter Weight

BACKGROUND

Stress analysis procedures require that accurate weights of components be used in modeling of piping systems for stress analysis for dead weight loads and dynamic loads. Where accurate weights are not known a value may be assumed. In that case this assumption is to be included in the list of assumptions in the calculation and flagged as requiring verification. When any assumption requiring verification is included in a calculation, an indication is required on the sign off sheet that confirmation is required.

DESCRIPTION

Flow meter IE12*FE in residual heat removal piping Loop B was assumed as weighing 450 lbs in a listed assumption in the pipe stress calculation AX-71AE-Z. This assumption was not flagged for verification. The sign off page of the analysis package was marked as not requiring verification.

BASIS

This was a violation of the Stone & Webster stress analysis procedure.

IMPACT ON DESIGN

This item appeared to have negligible impact on design.

EXTENT

Other items were observed where assumptions were not listed for verification.

REFERENCES

1. S&W Calculation AX-71AE-Z.
2. S&W CHOC-EMDM-82-12, Stress Analysis.

D3.3-3 (Deficiency) Safety/Relief Valve Nozzle Loads

BACKGROUND

The main steam lines are within the General Electric scope of supply out to the second isolation valve. These lines are connected to the safety/relief valve discharge lines supplied by Stone & Webster with the interface at the outlet of the safety/relief valves, which are supplied by General Electric. General Electric has provided input (Reference 1) on the maximum forces and moments allowed at the inlet and outlet flanges of the safety/relief valves. Pipe stress analysis on the entire system is performed by both General Electric and Stone & Webster. The analysis performed by Stone & Webster includes both the steam line and the safety/relief valve discharge lines (with ball joints) to the first anchor. One objective of the analysis is to evaluate the forces and moments on the flanges to assure that the limits are satisfied. Valve nozzle loads do not usually require special consideration by the pipe stress engineer.

DESCRIPTION

The pipe stress calculation which addresses main steam system loop C includes six connected safety/relief valve discharge lines. The input to this calculation (Reference 3) includes the interface requirements and flags the nozzle load limits on the safety/relief valves. No limits are shown in the calculation. The transmittal of results should reference a transmittal of calculated loadings relative to nozzle loading limits but no reference is shown. The equipment nozzle load summary in the calculation does not mention nozzle loads on the safety/relief valves except to note that they will be qualified by General Electric. This calculation was reviewed. The reviewers indicate that a comparison with similar calculations and a number by number check was performed in the review. A major reason for performing the analysis was overlooked in that the analysis did not address interface limits as required.

BASIS

A significant equipment interface control was provided but was not included in the evaluation of the results of the analysis.

IMPACT ON DESIGN

This item could affect design. Substantial reanalysis is needed. Results from the calculations appear to show that the allowable inlet nozzle loads are exceeded by as much as 50% while outlet nozzle loads are well within limits. The impact on design can be significant, but cannot be estimated at this time.

EXTENT

It is anticipated that the above situation will also be the case for the other three main steam lines.

POST CUTOFF WORK

Some reanalysis has been performed with improved modeling. The results of the reanalysis again predict excessive loads on the valves. These results were not reviewed during this inspection.

D3.3-3 (Deficiency) continued.

REFERENCES

1. S&W Interface Control Drawing (IDC) (GE 105D5229-6), SWEC File No. 0222-212-00-001G.
2. S&W Stress Calculation No. 12210-AX-2E-1, 7/20/83.
3. S&W I.O.C. DP-760 Boothby to Shah, 4/8/80.

D3.3-4 (Deficiency) Transmittal of Valve Acceleration Data

BACKGROUND

As part of the stress analysis effort, calculated accelerations of valves and their operators are transmitted to the Equipment Qualification Section. This is only done when certain threshold values of predicted acceleration are exceeded. These transmittals are controlled correspondence as part of the stress package.

DESCRIPTION

The transmittal of calculated accelerations of valves in the residual heat removal-low pressure coolant injection Loop B transmitted inaccurate information. Calculated accelerations for valve F042B and valve F037B were transposed in transmittal.

BASIS

The transmitted information on calculated accelerations of two valves in residual heat removal-low pressure coolant injection Loop B is inconsistent with the results of the calculations.

IMPACT ON DESIGN

There is no impact on design anticipated since the analysis is to be repeated with revised loadings and the difference in magnitude of transposed loadings was not great.

EXTENT

This did not appear to be systematic. Valve accelerations from other calculations were reviewed with no evidence of such a problem.

REFERENCES

1. S&W Calculation 12210-AX-71-K-2, Attachment E.

D3.4-1 (Deficiency) Valve Modeling

BACKGROUND

Stone & Webster procedures (Reference 1) define the valve modeling procedure to obtain data for equipment qualification. This procedure points out that variation in the modeling procedure has also been found to affect the stress response of piping elements adjacent to the valve. The interface control drawing for the safety/relief valve defines limits of moments (stress) to be applied to the valves. The procedure prescribes that valves with an extended structure be modeled as two masses connected by a flexible yoke, one mass for the valve and one mass for the actuator.

DESCRIPTION

In the calculation of stress in main steam line C, six safety/relief valves and two isolation valves (supplied by General Electric) are modeled with one eccentric lumped mass. One isolation valve supplied by Stone & Webster is modeled as prescribed by procedures. The modeling procedure for the General Electric valves is not listed as an assumption subject to verification. The analysis was reviewed and approved.

BASIS

This was a violation of Stone & Webster Engineering Corporation procedures involving dynamic analysis (Reference 1).

IMPACT ON DESIGN

The impact could not be determined during the inspection. Its effect need not be studied in isolation from reanalysis required to correct Deficiency D3.4-2.

EXTENT

It is anticipated that this would extend to all four steam lines and all sixteen safety/relief valves.

POST CUTOFF WORK

Reanalysis was performed with a two mass model for the main steam isolation valve.

REFERENCES

1. Stone & Webster CHOC-EMDM-81-28, 12/11/81.
2. Stone & Webster Calculation No. 12210-AX-2E-1, 7/20/82.

D3.4-2 (Deficiency) Ball Joint Modeling

BACKGROUND

The ball joints in the safety/relief valve lines are modeled in the calculation for stress analysis of main steam line C. The joints are defined in the model as having a lumped mass which is to be added to the equivalent pipe length to represent their total weight.

The length between ball joint centers is important in determining the amount of rotation of the joint that is needed to accommodate a given amount of movement of the pipe. Three different specified joints are purchased to accommodate different end connections. A 300 lb flange is welded to one joint and a 600 lb flange is welded to another joint.

Ball joints behave in a non-linear fashion, similar to a soft steel which would be relied on to yield and deform plastically with no increase in load. Since the computer code used for analysis of pipe stress, NUPIPE, is based on linear elastic behavior of pipe system elements, special modeling procedures have to be employed. Consideration must be given to (1) differences in behavior between bending and twisting of the joints, (2) trial and error analysis repetitions to assure reasonable loading predictions are achieved, and (3) ways of bounding the combined loadings and deflections. Superposition of loads and deflections for various load conditions (algebraic summation) is not valid since the ball joints yield (breakaway) at low loads and do not require additional load for additional motion in the same direction.

DESCRIPTION

A concentrated weight of 360 lbs is stated as an assumption for the analytical model to simulate the added ball joint mass in excess of the pipe weight. The assumption was not flagged for verification. The vendor's drawing shows a weight of 260 lbs total (not including the flanges) for the ball joint. The center of the ball joint adjacent to the safety/relief valve nozzle is modeled as being at the safety/relief valve nozzle flange. This is an error in location of the joint of about 10 inches. The calculation was reviewed and approved without detecting these errors.

The ball joints were modeled with the same moment resisting properties in each of the three principal directions. Trial and error repetitions had not been continued until reasonable estimates of loading had been achieved in all three directions. No evidence was shown of special consideration of combined deflections or loadings.

BASIS

The treatment of assumptions was a violation of Stone & Webster procedures. There was inadequate engineering consideration of a component with non-linear behavior in a linear based analysis.

IMPACT ON DESIGN

The error in weight is expected to have negligible effect on the design. The error in ball joint location could have a significant effect on safety/relief valve nozzle loadings since the safety/relief valve inlet flange limit is already exceeded by the

D3.4-2 (Deficiency) continued.

predictions of the analysis. The error in modeling the flexural characteristics of the ball joints indicated that predicted loads could increase even further. The lack of consideration of combinations of loading and deflections resulted in overlooking the limit on rotation of the joints and the result that the limit would be exceeded by those predictions.

EXTENT

It is assumed that this applied to ball joints in all sixteen safety relief valves discharge lines.

POST CUTOFF WORK

The system was remodeled and reanalyzed during the inspection but the results were not reviewed by the team.

REFERENCES

1. S&W Calculation No. 12210-AX-2E-1, 7/20/83.
2. Aeroquip Drawing BB31137-70-11, Rev. B., S&W File No. 0228-165-084-587.
3. CHOC-EMDM-82-12.

D3.4-3 (Deficiency) Dimensional Discrepancy

BACKGROUND

Piping systems are generally broken down into subsystems for purposes of stress analysis. The geometry of the subsystem is derived from the EP-series piping drawings and BZ-series pipe support drawings prepared by the Stone & Webster Power and Pipe Support groups.

DESCRIPTION

The elevation of pipe support 1RHS-PSA2134A2 for the vertical leg of pipe 1-RHS-014-39-2 is given as 87.5 ft. (node point 200) in stress package AX-71AE-2. However, the centerline dimension is given as 91.0 ft. on 1-BZ-71EP-CD-1 (the controlling drawing), and is also detailed as 91.0 ft. on the following pair of drawings, which are for information only:

- 12210-EZ-71ZE-1
- 1-RHS-039-CD-B, Rev. 8

BASIS

The basis of this deficiency is an out-of-tolerance dimensional discrepancy in excess of the allowable tolerance of 12 in. for class 2 pipe.

IMPACT ON DESIGN

The impact on design is probably minor.

EXTENT

This is considered a random deficiency.

POST CUTOFF WORK

Stone & Webster indicated that this discrepancy was noted after Revision 2 of stress package AX-71AE was issued and that this discrepancy will be corrected either before or during the as-built verification.

D3.4-4 (Deficiency) Pipe Support Stiffness

BACKGROUND

ASME NF design requirements as a function of design condition category are tabulated in FSAR Table 3.2-4. These ASME documents, as well as applicable ANSI B31.1 documents, are reiterated in the Reference 1 and 2 Stone & Webster pipe support and snubber specifications.

There is no comparable FSAR commitment to pipe support stiffness criteria. Stiffness criteria for restraints and anchors are initially elaborated in the Stone & Webster pipe support specification, which tabulates translational and rotational stiffnesses for supports as a function of nominal pipe size, and snubber spring rates as a function of rated load capacity. Although not explicitly indicated therein, the minimum spring rate criteria are invoked solely to the NF boundary. This assumption is accurate for pipe supports affixed to concrete slabs or walls. However, for supports affixed to structural steel, the additional flexibility of the primary structural steel can substantially decrease the overall support stiffness. As a consequence, the actual spring rates for such supports can be substantially lower than the spring rates employed in the pipe stress analysis.

DESCRIPTION

Hanger No. 1RHS *PSST2094A2, detailed on the Reference 3 drawing, is supported by a W16X45 beam which is shown in plan on the Reference 4 drawing. Pages 19 and 20 of the Reference 5 computation calculate a total support stiffness to the NF boundary of 4.49E6 lbs/in. If the flexibility of the W16X45 beam is additionally taken into account, the total support stiffness is then 0.18E6 lbs/in.

BASIS

There is a discrepancy between the as-built and calculated support stiffnesses.

IMPACT ON DESIGN

The impact on design is minor to moderate.

EXTENT

This appears to be a systematic deficiency.

REFERENCES

1. S&W Specification No. 228.310, Design and Fabrication of Power Plant Piping Supports, Rev. 1, Add. 1, dated 9/11/81.
2. Design and Fabrication of Mechanical Snubbers for Nuclear Power Plant Piping Supports ASME III, Code Class 1.
3. S&W Dwg. No. 1-BZ-71CW, Rev. 1, 2/18/82.
4. S&W Dwg. No. 12210-ES-66G-9, Rev. 9, 1/14/82.
5. S&W Calculation No. Z-71-20[illegible], Rev. 0, 12/09/80.

D3.4-5 (Deficiency) Percent Critical Damping

BACKGROUND

Table 3.7A1 of the FSAR specifies percent of critical damping for Operating Basis Earthquake and Safe Shutdown Earthquake seismic events as a function of pipe diameter. For piping greater than 12", the respective damping values are two and three percent. For piping equal to or less than 12", the respective damping valves are one and two percent.

DESCRIPTION

The segment of Loop B of the Low Pressure Coolant Injection mode of the RHR system contained in stress package AX-71P is primarily composed of piping greater than 12" in diameter, however, there is some 8" pipe. The input seismic response spectra have respective damping values of two and three percent for Operating Basis Earthquake and Safe Shutdown Earthquake, which is unconservative for the 8" diameter pipe.

BASIS

The basis for this deficiency is a failure to meet an FSAR commitment.

IMPACT ON DESIGN

The impact on design is probably minor.

EXTENT

This appears to be a random error.

POST CUTOFF WORK

The six stress packages comprising Loop B of the Low Pressure Coolant Injection mode of the Residual Heat Removal System were reviewed on 5/2/84. Only stress package AX-71P employed unconservative damping values for the input seismic response spectra.

D3.4-6 (Deficiency) Added Mass for Trapeze Hangers (BOP)

BACKGROUND

Trapeze hangers provide vertical support for a horizontal run of pipe, but no lateral support. In fact, for a seismic event occurring in the horizontal plane, the pipe drives the support mass. Depending upon the pipe size and the support configuration, added mass may have to be lumped at the pipe node located at the support point.

DESCRIPTION

There are three trapeze hangers contained in pipe stress package AX-71C:

- 1 RHS-PSSH3093A1
- 1 RHS-PSSH3092A1
- 1 RHS-PSSH3088A1

The Reference 1 Stone & Webster memorandum details a procedure to be used to determine whether there is a need to re-evaluate the piping system due to the added mass from the spring hanger assembly. This procedure was not employed.

BASIS

The basis of this discrepancy is the failure to invoke the memorandum.

IMPACT ON DESIGN

The impact on design is minor.

EXTENT

This deficiency appears systematic for any stress analyzed piping subsystem containing trapeze hangers.

POST CUTOFF WORK

A reanalysis of pipe stress package AX-71C was performed on May 30, 1984 which included the weights of the spring hanger trapezes. The resulting variations in the piping system frequencies were found to fall within the allowable variation permitted in the memorandum.

REFERENCES

1. S&W EMDM 81-04, Effect of the Spring Hanger Assembly Weight on Piping Systems.

D3.4-7 (Deficiency) Pipe Functionality Criteria

BACKGROUND

Section 3.9.1.4.2A.4 of the FSAR commits to the functional capability requirements detailed in NEDO-21985. These are contained in the Stone & Webster Technical Procedure EMTP-12.9.12.

DESCRIPTION

For purposes of stress analysis, Loop B of the Low Pressure Coolant Injection mode of the RHR system is broken down into six subsystems: 71AK, 71P, 71C, 71K, 71AF and 71AE. Although the procedure to evaluate functionality is referenced in several of these stress packages, functionality criteria are not addressed, nor is this flagged as an open item.

BASIS

The basis of this deficiency is the failure to meet an FSAR commitment.

IMPACT ON DESIGN

The impact on design cannot be assessed at this time.

EXTENT

This deficiency appears to be systematic.

POST CUTOFF WORK

The status of the functional capability check for ASME Code Class 1, 2 and 3 piping is addressed in an Interoffice Correspondence dated May 13, 1984.

REFERENCES

1. S&W Tech. Proc. No. EMTP 12.9.12-0, Procedure to Evaluate Functional Capability of BWR Essential Piping.

D3.4-8 (Deficiency) Time Limits on Drawing Revisions

BACKGROUND

Section 3.0 of the reference 1 Stone & Webster procedure details two time constraints limiting the elapsed time permitted before a drawing is revised to incorporate outstanding Engineering and Design Coordination Reports:

1. when a total of six outstanding Engineering and Design Coordination Reports is reached, a revision is to be issued within two months after the sixth Engineering and Design Coordination Report is issued;

2. at least annually.

DESCRIPTION

Revision 4 of the reference 2 Stone & Webster drawing was issued on 5/8/81. The sixth of a total of ten outstanding Engineering and Design Coordination Reports was issued on 3/15//82. Revision 5 of that drawing was issued on 10/24/83. Per the reference 1 procedure, revision 5 of that drawing should have been issued on 5/8/82 (annual) or on 5/15/82 (two months after the issue date of the sixth Engineering and Design Coordination Report). However, a written request was made to the Project Engineer on 9/14/82 for a waiver of the time constraints contained in the reference 1 procedure, and this was granted. It is noted that:

1. the ability to waive a requirement of the procedure by a request to the Project Engineer in writing is not a part of that procedure, although the procedure was originally approved by the Project Engineer;

2. the Interoffice Correspondence drafted to request the waiver of time constraints is not a controlled document.

BASIS

The basis of this deficiency is a failure to revise the drawing in accordance with the time limits stipulated in the procedure.

IMPACT ON DESIGN

This did not appear to have an impact on the design.

EXTENT

A number of other drawings were handled in similar fashion.

POST CUTOFF WORK

No specific post cutoff work is required.

REFERENCES

1. Stone & Webster Procedure RBP 12.0-12, dated 7/13/81, Attachment E.
2. Stone & Webster Drawing No. EP-71B.

D3.4-9 (Deficiency) Interoffice Correspondence Control

BACKGROUND

Stone & Webster Procedure River Bend Procedure 6.14 details the manner in which interoffice correspondence containing substantive technical information is to be controlled. Such documents are to be numbered and indexed and a response by the recipient is required.

DESCRIPTION

During a review conducted at the site to track Engineering and Design Coordination Reports written against the Reference 1 drawing, four Interoffice Correspondence's that were subsequently incorporated into Engineering and Design Coordination Reports were not numbered and indexed in accordance with the requirements of River Bend Procedure Procedure 6.14. References 2-5 list these Interoffice Correspondence's and the numbers of the corresponding Engineering and Design Coordination Reports.

BASIS

Failure to control interoffice correspondence containing substantive technical information in accordance with the prescribed Stone & Webster Procedure.

IMPACT ON DESIGN

This item had no impact on design or analysis.

EXTENT

Based on other deficiencies, occasional lapses in control of Interoffice Correspondence's appears to be systematic.

REFERENCES

1. Stone & Webster Dwg. No. 12210-EP-71B, Rev. 5, dated 10/24/83.
2. IOC from H. C. Lin/K. Basu to R. H. Bates, dated 10/1/81 (E&DCR P-11,129).
3. IOC from D. Bray to D. Koszowski/D. Nelson, dated 12/4/81 (E&DCR P-11,213).
4. IOC from H. K. Lee/K. Basu to D. Nelson/D. Koszowski, dated 2/17/82 (E&DCR P-11,372).
5. IOC from Y. J. Wu to D. Koszowski, dated 8/12/83 (E&DCR P-12,287).

D3.5-1 (Deficiency) Incomplete Pipe Support Location Plan

BACKGROUND

The design pipe support drawings issued by the Stone & Webster Engineering Mechanics Division Pipe Support Group are subsequently issued to the Site Engineering Group, where they are re-worked to become the controlled drawing to be used for the fabrication and installation of the pipe support.

DESCRIPTION

Pipe support 1 RHS-PSR3030A2, detailed on the Reference 1 drawing, lacks a dimension on the location plan which locates the pipe support with respect to an adjacent elbow.

BASIS

The key plan detailed on the pipe support drawing defines the location for that support. The location of the pipe support shown elsewhere is given for information only.

IMPACT ON DESIGN

The impact on design is minor.

EXTENT

This is a random deficiency.

REFERENCES

1. S&W Dwg. 1-BZ-7MV-CD (1 of 2), Rev. 3, dated 3/12/84

D3.5-2 (Deficiency) Small-Bore Seismic Piping Maximum Support Spans

BACKGROUND

Table 3.7A-8 of the FSAR details the maximum allowable support spans for small-bore seismic piping as a function of pipe size and building location. No span tolerance is allowed for the tabulated spans.

DESCRIPTION

The 3/4 in. line (1-RHS-750-198-2) shown on the Reference 1 drawing spans a distance of 10.0 ft. between supports 1-RHS-198*PSR-12 and -198*PSR-13. The line designation table shows this line to be uninsulated. According to Table 3.7A-8, the maximum allowable span length for an uninsulated 3/4 in. line located outside of the Reactor Building is 8.0 ft. The actual span therefore exceeds the maximum allowable span length by 2.0 ft.

BASIS

The basis of this deficiency is a failure to meet Stone & Webster design criteria as summarized in the FSAR.

IMPACT ON DESIGN

It is expected that the FSAR will be amended.

EXTENT

This is a random deficiency.

POST CUTOFF WORK

Stone & Webster indicated that the referenced span had been reworked by the Site Engineering Group in accordance with Stone & Webster Engineering Mechanics Division 83-13 and 84-03, which permit span lengths greater than the span lengths detailed in Table 3.7A-8 of the FSAR.

REFERENCES

1. S&W Dwg. No. 12210-EP-305P-2, Small Bore Piping Auxiliary Building El 70'-0" & 95'-0", Rev. 2, 1/25/83.

D3.5-3 (Deficiency) Dynamic Coupling

BACKGROUND

FSAR Subsection 3.7.3.4.1A details relative frequency and mass criteria to be employed to evaluate the potential for dynamic coupling between Seismic Category I equipment and supporting structure. Equipment and supporting structure meeting these criteria enable the rational application of amplified response spectra, which presupposes no dynamic coupling.

The team selected the four Residual Heat Removal System heat exchangers and exchanger support configurations to sample Stone & Webster compliance with FSAR Subsection 3.7.3.4.1A. This equipment is General Electric supplied and the support configuration is designed to General Electric specified stiffness criteria by the Stone & Webster Engineering Mechanics and Structural Divisions. Since responsibility for the support configuration rests with Stone & Webster, the validity of the dynamic as well as the static aspects of the heat exchanger support design must be verified.

DESCRIPTION

There are four RHR heat exchangers (1E12*EB001A-D) that are supported on the structural steel detailed on the Reference 1-3 drawings. This steel is designed to stiffness criteria established by General Electric, as detailed in the Reference 4 calculations. The seismic analysis performed was quasi-static, employing peak spectral values and dynamic load factors. However, no frequency analysis was performed, and the decoupling criteria detailed in Section 3.7.3.4.1A of the FSAR were not invoked in the Reference 4 calculation. It is therefore noted that the magnitudes of the peak spectral values employed in the static analysis, whose use presupposes an uncoupled dynamic system, have not been adequately verified.

BASIS

The basis of this deficiency is a failure to meet a Stone & Webster design commitment as summarized in the FSAR.

IMPACT ON DESIGN

The impact on design can be negligible to substantial, on a case by case basis.

EXTENT

This deficiency is systematic for major equipment mounted on steel.

REFERENCES

1. S&W Dwg. 12210 EV 155A-6, 1/6/84
2. S&W Dwg. 12210 EV 155B-7, 1/6/84.
3. S&W Dwg. ES-66V-1, 5/5/80.
4. S&W Calc. No. 221.900 HBA 1699, Rev. 4, 8/31/83.

D3.6-1 (Deficiency) ASME Code Edition for Valves

BACKGROUND

Each specification for components procured to ASME Code Section III establishes the applicable ASME Code Subsection, and the Edition and Addenda to which the component is to be constructed. This is stated in the text of the specification and is shown on the cover sheet with a stamped certification by a registered professional engineer as to compliance with the code in accordance with procedures.

DESCRIPTION

The Stone & Webster specification for carbon steel valves 2 1/2 inches and larger presently calls out the 1974 Edition on the cover sheet and in the text calls out the 1974 Edition with the Summer 1974 Addenda. The change in the body of the specification was in response to a change request. The document was approved by all reviewers.

BASIS

The code specified in the body of the specification is inconsistent with that of the cover sheet.

IMPACT ON DESIGN

None. The valves were constructed to the 1974 Edition with the Summer 1974 Addenda and stamped accordingly as noted in Reference 3. This later code is considered to be an improved code.

EXTENT

This did not appear to be systematic.

REFERENCES

1. RBP 2.001, General Procedures for the Preparation and Maintenance of Project Specifications.
2. S&W Specification RBS 228.211, Addendum 6, 7/7/83.
3. E&DCR No. P-1045.

D3.6-2 (Deficiency) Ball Joint Qualification

BACKGROUND

Main steam line C has six safety/relief valve discharge lines attached at the safety/relief valve discharge nozzles. Each of the discharge lines are anchored to the dry well wall with about 15 feet of piping between the nozzle and the anchor. To accommodate the thermal movement of the steam line relative to these anchors and limit the reactions at the safety/relief valve nozzles, three ball joints are incorporated in each of the discharge lines. The most significant characteristics of the ball joints are (1) adequate pressure retaining capacity (2) the available range of movement, (3) the breakaway flexing moment and torque required to flex and twist the joints as needed, and (4) their ability to resist all applied forces and movement throughout a service cycle without leaking excessively.

DESCRIPTION

The specification which procured these joints required a maximum flexural torque of 6500 ft lbs at 570 psig steam pressure and also specified the following tests: (1) Low Pressure Leakage Test, (2) Radiation Life Tests, (3) Endurance Cycle Life Test, (4) Vibration Test, (5) Shock Test, and (6) Temperature Cycling Tests. Stress Analysis and Design Calculation Reports were also required. No requirement is specified for analysis or test reports to justify the breakaway moment or torque requirements.

Reports which were provided by the vendor were reviewed and approved by Stone & Webster. Some of these reports do not meet the requirements of the specification and no documented resolution of the inconsistencies was recorded, as noted in the following:

1. Low Pressure Leakage Test Report, Section 5.2 Reference 2

   a. Specification requires testing with steam at 5 psig for 30 days with leakage less than 2 cc of water/minute.

   b. Reference 2 reported a test with steam at 6 psig for 60 days with acceptable leakage.

The report was accepted without comment since the imposed test condition obviously exceeds the specified requirements. This is not considered an inadequate review.

2. Radiation Life Test, Section 5.3 and Appendix 1 of Reference 2, plus Reference 7

   a. Specification requires:

      1. Integrated Gamma dose of 4.5E07 rads
      2. Neutron dose 1.8E15 neutrons/sq.cm

   b. References 2 and 7 reported a test on seal plus lubricant:

      1. 2.6 thru 7.0E07 rads gamma
      2. no neutron irradiation

D3.6-2 (Deficiency) continued.

No significant change in tensile properties or deterioration was reported. The report was approved without comment and did not meet the requirements for neutron irradiation. This is not considered an inadequate review.

3. Endurance Life Cycle, Section 5.4 of Reference 2

   a. Specification required cycling through the allowable motion (+/-7.5° arc) for 8000 cycles at 570 psig and 480°F steam. Acceptance is based on leakage less than 2 cc of water/minute.

   b. Reference 2 reported a test of three ball joints in a specific assembly with the movement applied at one end of the assembly as follows:

      1) 8000 cycles of .03" movement at 500 psig, 470°F in the axial direction.

      2) 500 Cycles of 3.0" movement in the transverse direction at 5 psig, 227°F without unacceptable leakage.

The report was approved without comment regarding the endurance test. No relation is shown between end motion and angular rotation, and the test did not meet requirements for angular movement. The test results indicate slight surface adhesion between the ball and the gasket. Such adhesion could significantly affect breakaway torque values. No torque values are reported with these test results. This is considered an inadequate review.

4. Vibration Test, Section 1.7 of Reference 2 refers to Reference 3 for qualification by similarity.

   a. The specification requires testing at a frequency of 25 to 125 Hz., with an amplitude of 0.05 to 0.005 inches for 170 hours in each direction, 510 hours total, with any resonances recorded.

   b. Reference 3 presents test data from a resonance search on 5" and 14" joints in assemblies of 3 joints. Several resonance frequencies are apparent in the data between 40 and 100 Hz.

Testing for 170 hours in each direction at 33 Hz. and 0.06" amplitude was reported.

Reference 3 is a stand alone document which, while presently available at Cherry Hill, has no supplier's document data form, is not a part of the official records at Stone & Webster, and there is no evidence presented that it was reviewed as part of the review of Reference 2. The qualification report Reference 2 was reviewed without comment regarding the Vibration Test or Reference 3. The test can be judged to satisfy the specified requirements although the reported resonant frequencies can be on a peak of the hydrodynamic loading response spectrum. This review is considered inadequate in that Reference 3 is not an official record.

5. Shock Test, Section 1.7 of Reference 2, refers to Reference 3 for qualification by similarity.

   a. The specification requires testing from 0 to 625 psig for 8000 cycles at 30 cycles per minute with no leakage.

D3.6-2 (Deficiency) continued.

b. Test reports of Reference 3 offer test results from tests performed to meet MIL-5-901C (Navy) specification on 5 inch and 14 inch ball joints specimens. The test consisted of 3 hammer blows in each of 2 directions (hammer heights of 1, 3 and 5 feet). Post test inspection indicated a hair line crack in a gasket on a 5 inch joint.

The report, Reference 2, was approved without comment regarding the shock test. The review of Reference 2 does not mention Reference 3. The shock test did not meet requirements of the specification. This is considered an inadequate review.

6. Temperature Cycling Tests, Section 1.7 of Reference 2, refers to Reference 3 for qualification by similarity.

a. The specification requires 100 cycles of rapid steam heating from 135^{O}F to 480^{O}F with slow cooling and no unacceptable leakage.

b. Reference 3 offers results of tests on 5 inch and 14 inch joints with 100 cycles of measured temperature change between 460^{O}F and 210^{O}F with no leakage. Steam at 625 psig was used for cycling the 5 inch joint and 450 psig on the 14 inch joint.

Reference 2 was reviewed and approved without comment regarding the cycling tests or Reference 3. The test can be judged as meeting the specification. This is not considered an inadequate review.

The specification requires a pressure design report. Aeroquip Reports (4) and (5) were submitted and are judged to satisfy this requirement. The report was approved and reviewer comments justifying acceptance are included.

The specification requires an operating instruction manual to be submitted and that the manual address the frequency and type of lubrication. A manual was submitted which does not address frequency nor type of lubrication. Qualification of lubrication is reported in Reference 7, therefore, it is assumed that lubrication is needed. The report was approved with no comments and does not appear to meet the specification. This is considered an inadequate review.

The requirements for qualification and acceptability of the submitted documents is confused by the contents of three letters (References 8, 9, and 10). B. F. Shaw, the piping supplier responded to the preliminary copy of the ball joint specification with clarifications. Clarification item 4 states that certain documents (References 2, 4, 5, 7, and 11) are to be accepted by Stone & Webster as demonstrating that the Barco joints satisfy all the design conditions and qualification tests as stated in specification No. 228.150, Section 5. It stated that "NO ADDITIONAL TESTING FOR QUALIFICATION OR VERIFICATION OF DESIGN CALCULATIONS ARE INCLUDED IN THIS QUOTATION" with emphasis as shown here. Stone & Webster transmited the B. F. Shaw letter to Gulf States with a cover letter which states that Stone & Webster concurs on the proposed clarifications of acceptability of the referenced reports. Gulf States concurred with the Stone & Webster recommendations for accepting the B. F. Shaw proposal and stated an understanding that Stone & Webster will delete requirements to verify the pressure design calculations and qualification test reports but will review reports prior to issuance of the purchase orders. Reference 11 was rejected on review with reviewer comments and was voided.

D3.6-2 (Deficiency) continued.

No change in the specifications was made as a result of this correspondence. The hand written notations on Reference 9 suggest a need for such changes.

The specification was approved as issued in accordance with the Engineering Assurance procedures which require review by a specialist. The specification appears to have been modeled after the General Electric specification for ball joints. Most Aeroquip reports appear to have been originally submitted to General Electric as part of a previous design effort. Since ball joints appear to be relatively new to the Stone & Webster piping specifications and personnel, significant attention should have been given to this area by a specification specialist. The specialist was not interviewed as part of this inspection.

BASIS

Specification content and subsequent actions are not consistent with Stone & Webster procedures in that the specification did not contain requirements for documented verification of values of breakaway torque, and some test reports do not meet specified requirements. Correspondence appears to show a need for change to the specification in order to accept the reports which were submitted in order to make their approval consistent with the contents of the specification.

IMPACT ON DESIGN

Initial reevaluation of the results of calculations of steam line C, using catalog values of torque, predicts that ball joints would exceed the allowable movement of 5° arc (7.5° arc less 2.5° arc allowed for erection misalignment) and the loads on the safety/relief valve nozzles exceed those allowed. If the catalog values of torque cannot be verified and higher values must be used, the valve nozzle loads may increase. If lower values of torque are used, the predicted amount of rotation increases.

EXTENT

This can be expected to apply to safety/relief valves and ball joints on all four steam lines.

POST CUTOFF WORK

The following work has been performed by Stone & Webster.

Change notice No. D-12,830 was issued stating the need to revise the specification to reflect actual documentation submittal.

Reanalysis has been initiated which continues to use catalog values of torque.

A request to Aeroquip for substantiating data concerning the 6,500 ft lb torque requirement produced a letter (Reference 15), which enclosed a graph which is submitted as torque test data for perusal. The graph appears to be from an Aeroquip Report TR101.116A.

D3.6-2 (Deficiency) continued.

REFERENCES

1. S&W Pipe Specification No. 228.150, Rev. 1, Addendum 8.
2. S&W File No. 6228-150-084-0061, Qualification Test Report Aeroquip (Barco) Report 40764-2.
3. Aeroquip Report 101,116C.
4. Aeroquip Report No. 122019, Design Justification 10" Ball Joint, S&W File No. 6228-150-084-001A.
5. Aeroquip Report No. 122015, Burst Pressure Test, 10" Ball Joint, S&W File No. 6228-150-084-004A.
6. Aeroquip Report ACES 2216 Rev. 0, Installation and Maintenance Manual for 10" Flex Ball Joints, S&W File No. 3328-150-084-002A.
7. Aeroquip Report No. 122021, Radiation Life Test for 11-N Mat'l., S&W File No. 6628-150-084-005A.
8. Letter, B. F. Shaw Co. to S&W, BFS No. 2035, with handwritten notation, October 21, 1981.
9. Letter, S&W to GSU, RBS-7103, November 23, 1981.
10. Letter, GSU to S&W, RBG-11,769,5-6,025, December 14, 1981.
11. Crosby Valve and Gage Co. Report No. 3701, Ball Joint Cyclic Loading Test, S&W File No. 6228-150-084-003A, (Rejected on review and voided).
12. S&W EAP 4.13, paragraph 4.2.7, Processing of Project Specifications.
13. S&W EAP 3.1, Verification of Nuclear Power Plant Designs.
14. S&W RBP 2.0-1, General Procedures for the Preparation and Maintenance of Project Specifications.
15. Letter, Aeroquip (R. L. Chellevald) to S&W (R. Tate), dated 5/1/84.
16. General Electric Document No. 21A3536, Rev. 1.

D4.3-1 (Deficiency) Calculation of Lumped Masses

BACKGROUND

A computer program called 'MASS' was used to calculate the masses and mass eccentricities to be used in the seismic analysis of the Reactor Building. The input and output of this program verifies the masses included for the seismic analysis.

DESCRIPTION

The input and output information was not contained within the calculations.

BASIS

Engineering Assurance Procedure 5.3 requires that input and output of computer analysis be a part of the calculations.

IMPACT ON DESIGN

This item was a documentation problem and had no impact on design or analysis.

EXTENT

This deficiency did not appear systematic, since the mass calculations for the Auxiliary Building were properly documented.

POST CUTOFF WORK

Stone & Webster personnel located the computer input & output during the inspection.

REFERENCES

1. S&W Calculation, 201.130.085, Verification of Reactor Building Seismic Analysis, Revision 0, 9/25/80.
2. S&W EAP 5.3, Preparation and Control of Manual and Computerized Calculations, Revision 3, 1/31/79.

D4.3-2 (Deficiency) Concrete Strength of Drywell

BACKGROUND

The concrete strength of the drywell in the Reactor Building above elevation 90' - 0" was changed from 3000 psi to 4000 psi. At a later date when the seismic analysis of the Reactor Building was performed, the changed concrete strength was not utilized.

DESCRIPTION

In calculating the stiffness properties of the drywell above elevation 90'-0", a concrete strength of 3000 psi was used instead of 4000 psi.

BASIS

The seismic analysis was inconsistent with actual design.

IMPACT ON DESIGN

This item should have minimal impact on the analysis of the Reactor Building.

EXTENT

The deficiency is not systematic and is limited to the calculation of stiffness properties of the drywell.

REFERENCES

1. S&W E&DCR #C-30013 dated 7/10/80.
2. S&W Calculation, 201.120.124, Seismic Analysis of Reactor Building with Concrete Fix, 1/22/82.

D4.3-3 (Deficiency) Effects of Torsional and Rocking Modes

BACKGROUND

During the development of the seismic amplified response spectra curves for the Reactor Building the torsional and rocking modes of vibration were excluded.

DESCRIPTION

The seismic amplified response spectra curves do not include the horizontal and vertical components of torsional and rocking modes of motion.

BASIS

River Bend Station FSAR Section 3.7.2.5A was violated.

IMPACT ON DESIGN

This item should be evaluated to determine the impact on the amplified response spectra for each mass point.

EXTENT

The deficiency is systematic since review of the Auxiliary Building analysis revealed the same approach.

REFERENCES

1. S&W Calculations, 201.120.124, Seismic Analysis of Reactor Building with Concrete Fix, 1/22/82.
2. RBS FSAR Section 3.7.
3. S&W Calculations, 201.130.010, Seismic Analysis of Auxiliary Building, Revision 1, 3/19/76.

D4.3-4 (Deficiency) Vertical Frequencies of Floor Systems

BACKGROUND

The vertical seismic amplified response spectra curves for the Reactor Building were developed without considering the vertical frequencies of the floor systems.

DESCRIPTION

The floor systems were assumed to be rigid in the vertical direction. Such an assumption does not consider the amplifications in the vertical response spectra for equipment to be installed on the floor systems.

BASIS

The flexibility of the floor systems can effect the vertical amplified response spectra.

IMPACT ON DESIGN

This item should be evaluated to determine the impact on the amplified response spectra for each mass point.

EXTENT

The deficiency is systematic because the same approach was used for the seismic analysis of the Auxiliary Building.

REFERENCES

1. S&W Calculation 201.120.124, Seismic Analysis of Reactor Building with Concrete Fix, 1/22/82.
2. S&W Calculation 201.130.010, Seismic Analysis of Auxiliary Building, Revision 1, 3/19/76.

D4.4-1 (Deficiency) Discrepancy between Calculations and Structural Criteria

BACKGROUND

During the construction of the drywell the minimum compressive strength of concrete has been changed from 5,000 psi to 4,000 psi.

DESCRIPTION

There is an apparent discrepancy between the Structural Criteria and the calculations with regard to the compressive strength of concrete in the drywell wall. This discrepancy is specified below.

	Criteria	Drawing	Calculations
El. 70' to 90'-3"	5,000 psi	5,000 psi	5,000 psi
Above 90'-3"	4,000 psi	4,000 psi	3,000 psi

BASIS

The calculations should be in conformance with the criteria.

IMPACT ON DESIGN

Since the lower value of concrete strength was used in the calculations, there should be no impact on design.

EXTENT

This deficiency was confined to the drywell wall.

REFERENCES

1. S&W Criteria, Structural Design Criteria for River Bend Station, Document No. 200.010, Rev. 2, 7/8/83.
2. S&W Calculations, Drywell Design Calcs., N.12210.201.120-105 Sh. 20, Rev. 0, 3/82.
3. S&W Drawing, Dwg #12210-EC-30A-12, Rev. 12, 5/9/83.

D4.4-2 (Deficiency) Identification of Reviewed Pages

BACKGROUND

When more than one reviewer was engaged in review of design calculations the portion reviewed by the individual reviewers should be traceable.

DESCRIPTION

Title Sheets for the Primary Shield Wall, Weir Wall Design and Reactor Pressure Vessel Pedestal calculations did not have identification of who reviewed the individual pages. There were several reviewers in each case.

BASIS

This is in violation of Engineering Assurance Procedure 5.3.

IMPACT ON DESIGN

The impact on design is unknown.

EXTENT

This item appears to be systematic.

REFERENCES

1. S&W EAP-5.3 Preparation and Control of Manual and Computerized Calculations, Rev. 3, January 31, 1979.
2. S&W Calc. No. 201.120-068, Primary Shield Wall Supplementary Calculations, Rev. 0, January 10, 1980 and April 9, 1984.
3. S&W Calcs. No. 201.120-048, Weir Wall Design, Rev. 0, April 30, 1980, June 6, 1980, March 12, 1981 and March 12, 1981.
4. S&W Calcs. No. 201.120-070, Design of Reactor Pressure Vessel Pedestal Supplementary Calculations, Rev. 0, June 17, 1980.

D4.4-3 (Deficiency) Mistaken Cross-reference in Drawings

BACKGROUND

Reinforced concrete drawings have been detailed by Trans-American Steel Company. This company is a sub-vendor of Kelly's Associates, the principal contractor for the detailing of reinforcing bars for reinforced concrete structures at River Bend Station.

DESCRIPTION

During the review of Trans-American Steel Company reinforced concrete shop drawings for the reactor building, it was found that a note on drawing RC19A incorrectly referenced drawing RC59.

BASIS

The reference was erroneous.

IMPACT ON DESIGN

This deficiency does not appear to have any impact on design.

EXTENT

This item did not appear systematic based on the check of other drawings.

REFERENCES

1. S&W Drawing, Dwg RC19A - Shield Wall, Reinforcing If, Horiz. & Vert., 2 Lift El. 140'-0" to 200'-0", South Half, Rev. 1, 6/11/80.
2. S&W Drawing, Dwg RC 59 - Drywell Wall #18 Add'l Vert. Bars #18 & #14 Horiz. Bars, 9/22/80.

D4.4-4 (Deficiency) Incorrect Capacity Factor

BACKGROUND

Load combination equations for the design of structural steel are listed in the beginning of the calculations. In these load combination equations factors are applied which increase the allowable stresses.

DESCRIPTION

In one of the load combination equations, an incorrect factor of 1.7 was noted instead of the correct factor of 1.6.

BASIS

This is inconsistent with Section 3.8.3 of the FSAR and the value used in the calculations.

IMPACT ON DESIGN

Since the correct factor was used in the calculations it does not have an effect on design. However, this error was not corrected by the checking or verification process.

EXTENT

This appears to be an isolated case.

REFERENCES

1. S&W Calculation No. 12210-201.120-068, Primary Shield Wall p. 4A, February 23, 1984.

D4.4-5 (Deficiency) Mistakes in Review of Weir Wall Calculations

BACKGROUND

Design calculations for weir wall were reviewed for their accuracy and consistency.

DESCRIPTION

During the review of the calculations the following errors were noted:

a) On page 57 of Reference 1 the pressure of 170 psi was marked as 170 deg. F.

b) On page 236 the cross-reference identifying the maximum force on cadweld splice at the bottom of the wall was listed as page 277. It should state page 180.

BASIS

Although these mistakes have not been carried out in the analysis they indicate some carelessness in the review program and violation of the provisions of Engineering Assurance Procedure 5.3.

IMPACT ON DESIGN

This particular item had no impact on design or analysis.

EXTENT

A general discussion of errors in calculations is discussed in Report Appendix A-1.

REFERENCES

1. S&W Calc. No. 12210.201-048, Weir Wall Design, Rev. 0, 12/19/83.
2. S&W EAP 5.3, Preparation and Control of Manual and Computerized Calculations, Rev. 3, January 31, 1979.

D4.6-1 (Deficiency) Inadequate Anchorage of Radial Shear Reinforcing

BACKGROUND

The shield building, the drywell, weir wall, and reactor pedestal contains stirrups inclined at 45^{o} for the purpose of resisting radial shears. The so called "Z bar" consists of a diagonal portion and two vertical tails.

DESCRIPTION

The stirrups were not anchored in accordance with ACI 318-71 which was committed by FSAR. For the shield building, the designer noted that the anchorage was not explicitly specified by the ACI code, but that this type of anchorage was previously used. The embedment length was calculated from the mid point of the wall which therefore includes one half of the diagonal portion and a vertical tail.

The other cylindrical structures of the reactor building were designed differently in that only the vertical legs were considered in the calculation of the required embedment length. A comparison of the two methods used clearly indicates that the method used for the shield building is much less conservative. In both methods of design the embedment lengths required were reduced by applying a factor of 0.8. The team questioned the application of this reduction factor in this application.

BASIS

Paragraph 12.13 of ACI 318-71 requires a standard hook plus embedment of 0.5 times the embedment length or bending around longitudinal reinforcement at least 180^{o} in the tension zone. The method used in the design does not meet these requirements.

For the shield building, the design was based upon "previous use" only and was subsequently considered satisfactory after checking and review even though no references were listed, as required by Engineering Assurance Procedure 5.3, Attachment 1.4. For the other structures the anchorage provided by the vertical tails meets neither the requirements for a tension splice for bent up bars nor the hooks required for stirrups.

IMPACT ON DESIGN

The new containment loading generated by General Electric eventually led to the filling of concrete in the annulus between the shield and containment structures. This eliminates the need for the radial shear reinforcing placed in the shield building and therefore there is no impact upon the design of the shield building. The anchorage in other buildings is considered acceptable.

EXTENT

The same type of inclined stirrups are used in the drywell wall, the weir wall, and the primary shield wall; however, none were found by the team in other structures.

POST CUTOFF WORK

Stone & Webster has interpreted the requirements of 12.13 of ACI 318-71 to include a method of anchorage which requires that one half of the anchorage requirement be met in the length of stirrup from mid point of the wall to the point of tangency at the vertical tail. An additional one half is to be provided from the point of tangency

D4.6-1 (Deficiency) continued.

through the vertical leg.

Stone & Webster has stated that the shear reinforcing calculations will be revised for the cylindrical structures of the reactor building.

Stone & Webster has performed additional analyses of the Shield Building which account for a lower thermal effect and the fact that concrete fill has been placed in the annulus between the Shield Building and the Containment Vessel. The analyses indicate that no radial shear reinforcing is required in the Shield Building.

REFERENCES

1. S&W Calculation No. 201.120-067, Analysis and Design of Shield Building, Supplementary, Rev. 0, 1/30/81.
2. S&W Calculation No. 201.120-020, Shield Building, pages 209, 210 and 278.
3. S&W EAP 5.3, Preparation and Control of Manual and Computerized Calculations, Revision 3, 1/31/79.
4. ACI 318-71, Building Code Requirements for Reinforced Concrete.

D4.6-2 (Deficiency) Calculation Procedures

BACKGROUND

The supplementary calculations (Reference 1) were prepared under Engineering Assurance Procedure 5.3 and partially under Structural Technical Procedure 11.5-2. The former procedure does not require page-wise notation of reviewers as does the latter, subtier (divisional), procedure. Therefore a consistent approach in placing reviewers names on individual sheets of the calculations was not required or used. Many inconsistencies between the calculation sheets and the cover sheet were found.

DESCRIPTION

The following procedural inconsistencies were noted on calculations.

1. Calculations were internally documented as prepared by GSB, L.C. Shiau, JKS; however, these individuals were not indicated as preparers on the calculation cover sheet.

2. The initials of one of the preparers (SKA) was placed on the calculation cover sheet by his supervisor.

3. Since there are two names on the "reviewed by" line and many sheets are unsigned, there is no record of what sheets were checked and by whom.

4. The checker on page 227 (SKA) was not listed as checker on the cover sheet.

BASIS

Engineering Assurance Procedure 5.3 requires that the individual reviewer and preparer be identified for each portion of a calculation.

IMPACT ON DESIGN

This item has no impact upon the design and analysis.

EXTENT

Other calculations were identified with similar deficiencies in the identification of reviewers and preparers (see Appendix A-1).

REFERENCES

1. S&W Calculation No. 201.120-067, Analysis and Design of Shield Building (Supplementary Calculations), Rev. 0, 1/30/81.
2. S&W Technical Procedure 11.5.-2, Control of Structural Division Calculations, Rev. 2, 11/30/82.
3. S&W EAP 5.3, Preparation and Control of Manual and Computerized Calculations, Rev. 3, 1/31/79.

D4.7-1 (Deficiency) Inadequate Anchorage of Vertical Reinforcing Bars Into Mat

BACKGROUND

The mat of the reactor building has many vertical reinforcing bars which extend into the mat.

DESCRIPTION

Some of the required embedment lengths of vertical reinforcing bars were based upon concrete strengths f'_c = 5000 psi.

BASIS

The minimum specified concrete strength in the mat was required to meet only f'_c = 4000 psi.

IMPACT ON DESIGN

There is a possibility of inadequate anchorage because the 5000 psi strength was only used apparently when 4000 psi was insufficient.

EXTENT

This deficiency did not appear to extend beyond the Reactor Building mat calculation.

POST CUTOFF WORK

Actual test results of concrete are being obtained.

REFERENCES

1. S&W Calculation No. 201.120-096, Verification of Reactor Building Mat Design Rev. 0, 3/12/82.

D4.11-1 (Deficiency) Pump Shaft Casing Moments

BACKGROUND

The Residual Heat Removal, Low Pressure Core Spray and High Pressure Core Spray pump shafts extend from 15' to 20' below the bottom of the Auxiliary Building mat. These shafts are enclosed by cylindrical pipe-like concrete structures. The concrete housings for the pump shafts develop moments in the mat. These moments should be considered in the design of the mat.

DESCRIPTION

The shears in the mat due to the pump shaft casing moments were calculated; however, they were not added to the other appropriate loads in checking the adequacy of the design of the mat.

BASIS

Shear forces due to the pump shaft casing moments were omitted from design calculations by mistake.

IMPACT ON DESIGN

The omitted loads should be added to the other loads to determine whether the design is adequate.

EXTENT

This item (omission of shear forces) did not seem systematic, based upon review of other calculations.

POST CUTOFF WORK

Stone & Webster performed calculations to include these shears into the analysis of the foundation mat on 5/10/1984.

REFERENCES

1. S&W Calculation C66.201, Auxiliary Building Foundation Mat Analysis & Design, Revision 2, 4/16/82.

D4.11-2 (Deficiency) Mat Shear Design

BACKGROUND

The Auxiliary Building foundation mat was analyzed using the STRUDL finite element computer program. In this analysis thin shell elements were used to represent the foundation mat. However, in the design some code requirements for deep beam theory were used which were not consistent with the analysis. The shear capacities used and the sections checked for critical shear were not in accordance with the code requirements. A published paper which proposed an alternate method to the code (Reference 3) was used to determine the adequacy of the mat for shear.

DESCRIPTION

The ACI Code has design requirements for shear reinforcing for deep beams and also for slabs and footings. The design of the mat should be consistent with the analysis performed and the requirements of the ACI Code.

BASIS

The design was not consistent with analysis performed. The code requirements for checking shear at two critical locations were not met. ACI 318-71 section 11.10.1 requires shear strength to be checked at two locations: 1) A distance 'd' away from the support point with an allowable shear stress of 2 fc'; 2) A distance 'd/2' away from the support point with an allowable shear stress of 4 fc'. Stone & Webster checked for an allowable shear of 4 fc' at a distance of 'd/2' or 0.15ln (clear span between support points), whichever was more limiting. As a consequence the mat was not checked for shear at a distance of 'd' from the support point. This situation may be the most limiting in certain situations. Distance 'd' is the distance from center of gravity of reinforcing to the outermost fiber of concrete in compression (otherwise known as the effective depth of the member).

IMPACT ON DESIGN

A reanalysis of the foundation mat for shear is needed to determine whether the mat is structurally adequate.

EXTENT

This deficiency does not seem to be systematic since the Reactor Building mat does not show the same problems.

POST CUTOFF WORK

Stone & Webster has performed calculations checking the shear in the mat using section 11.10.1 of the ACI code. In this reanalysis higher strengths of concrete than specified in design were used. These calculations were not reviewed by the team.

D4.11-2 (Deficiency) Mat Shear Design

REFERENCES

1. S&W Calculation, C36.201, Auxiliary Building Foundation Mat Analysis & Design, Revision 2, 4/16/82.
2. ACI 318-71, Building Code Requirements for Reinforced Concrete.
3. Design of Containment Base Mats, paper by Anil K. Kar, Specialty Conference on Structural Design of Nuclear Plant Facilities, 12/17/73.

D4.12-1 (Deficiency) Lack of Calculations for Nelson Studs

BACKGROUND

In connection with the audit of calculations pertaining to the floor at Elev. 114'-0" the team reviewed the pertinent drawing. During this review the team noted that Nelson studs have been welded to the sides of the beam webs.

DESCRIPTION

The calculations did not reflect the addition of the Nelson studs and appropriate calculations could not be found. The apparent reason for the studs was to provide connection between the slab and the supporting steel beam. In view of lack of calculations there is no assurance that number of the studs is sufficient.

BASIS

(1) The drawings should reflect and be consistent with the design calculations.

(2) Calculations are needed to provide the basis for the information contained on the engineering drawings.

IMPACT ON DESIGN

The impact on design is unknown.

EXTENT

From the review of several drawings and calculations for the floors, it appears that this deficiency is systematic.

REFERENCES

1. S&W Calculations, Design Check of Elev. 114'0" Floor Framing, Containment, Reactor Bldg (SW Quadrant), Rev. 0, 12/16/83.
2. S&W Drawing, Dwg #12210-ES-53-J-8, Misc Framing Details Reactor Bldg Sh. 4, Rev. 8, 10/13/82.

D4.12-2 (Deficiency) Qualitative Elimination of Load Combinations

BACKGROUND

The load combination equations pertaining to floors at El. 114'-0" and 118'-3" have been narrowed down to several factors, or sometimes even one, which is qualitatively considered to be the critical one. These equations were used in the design of the members.

DESCRIPTION

The few equations selected as the governing design have been selected without any evidence of quantitative analysis. It appeared that some were selected on the basis of judgement or qualitative considerations.

BASIS

Since the allowable capacities of structural members vary with different loading conditions, qualitative determination of the most critical equation may lead to an error.

IMPACT ON DESIGN

The impact on design is unknown.

EXTENT

This practice appears to be widespread in all of the structural elements reviewed.

REFERENCES

1. S&W Calculations, Steel Floor Framing at El. 118'-3" No. S53-500, Rev. 1, 2/17/84.
2. S&W Calculations, Design Check of El. 114'-0" Floor Framing No. S53-500, Rev. 0, 6/3/83.
3. S&W Calculations, Weir Wall Design Calcs., No. 201-120-048, Rev. 0, 12/83.
4. S&W Calculations, Primary Shield Wall Calcs., No. 12210-210.720-068, Rev. 0, 2/83.

D4.12-3 (Deficiency) Lack of Calculations for Support Angles and Shear in the Concrete Slab

BACKGROUND

Drawings have been reviewed in connection with the inspection of the floor at El. 114'-0". The team noticed that in some cases the slab is supported by a beam by means of a structural angle. There was no evidence that this angle was analyzed for the load delivered by the slab, or that the shear in the slab at the point of support by the steel beam was examined.

DESCRIPTION

The structural drawings should be supported by calculations which would demonstrate that the members shown on the drawings are appropriate.

BASIS

Drawings should be supported by design calculations in conformance with the requirements of EAP 3.1, Attachment C.3.

IMPACT ON DESIGN

The impact on design is unknown.

EXTENT

The above situation was noted in several cases and appears to be systematic.

REFERENCES

1. S&W Calculations, Design Check of El. 114'-0", Floor Framing, Containment, Reactor Bldg (SW Quadrant), Rev. 0, 12/16/83.
2. S&W Drawing, Dwg #12210-ES-53-5-8, Misc. Framing & Details, Reactor Bldg Sh.4, Rev. 8, 10/13/82.
3. EAP 3.1, Verification of Nuclear Plant Designs, Rev. 2, 7/8/81.

D4.15-1 (Deficiency) Checking Procedure Violations, Cable Tray Supports

BACKGROUND

Cable tray supports are designed from standards which have been prepared by Stone & Webster's Cherry Hill office. In calculation ES 4000, pages ES 4026 and ES 4039 were checked by means of alternate calculations in Attachments 1 and 2. The original calculations found a single degree of freedom frequency which considered the flexibility of the structural tube cantilever support but ignored the flexibility of the cable tray. The checker apparently realized the error and performed a two degree of freedom analysis which resulted in a fundamental frequency almost half of the original.

DESCRIPTION

The alternate calculation was in fact not a check but rather a new calculation. However, since alternate calculations need not be checked, certain errors in the alternate calculations were not corrected. The calculated frequencies of the two calculations were 50 Hz and 31 Hz. The checker apparently reasoned that even though there was a wide disparity in frequencies the dynamic responses would be the same. In the new calculations an incorrect formulation was used to determine the frequency response.

BASIS

In Structural Technical Procedure 11-5-2, paragraph 5.1.3, it is stated that "alternate calculations need not be checked provided there are no changes in results obtained by the preparer." Since the original calculation was in error and the calculated fundamental frequency differed, Structural Technical Procedure 11-5-2 was violated.

IMPACT ON DESIGN

The calculation is being corrected. The increase in acceleration is in the axial direction of the support member which would probably not be controlling.

EXTENT

This item does not appear to be systematic. In general, alternate calculations were not used for checking.

REFERENCES

1. S&W Calculation No. ES 4000, Reactor Building Cable Tray Supports Attached to Steel Containment, Rev. 0, 10/19/83.

D4.16-1 (Deficiency) Unverified Program PIPERUP

BACKGROUND

During the inspection of Reactor Controls, Inc., Pipe Rupture and Pipe Whip Evaluation (Appendix C to SA-932-DAO) was reviewed.

DESCRIPTION

The computer program PIPERUP has been included in the "Pipe Rupture and Whip Evaluation" criteria. However, it was not included in the computer verification program.

BASIS

The computer programs to be used in the analysis should be verified according to the provisions of the Quality Assurance Manual.

IMPACT ON DESIGN

Reactor Controls, Inc. stated that this program will not be used in this analysis, therefore, it will not affect any design.

EXTENT

The deficiency was not systematic.

POST CUTOFF WORK

The PIPERUP program was removed from the Pipe Rupture and Whip Evaluation Criteria during the inspection.

REFERENCES

1. RCI Submittal, Pipe Rupture and Pipe Whip Evaluation CRD System, Rev. 3,9/19/83.
2. RCI Submittal, Appendix D to SA-932-DAO, Verification Descriptions of Computer Programs Used on the River Bend Project, Rev. 0, 5/27/83.
3. RCI Quality Assurance Manual, Second Edition, Rev. 9, 9/21/83.

D4.16-2 (Deficiency) Document Control of Reactor Controls, Inc.

BACKGROUND

The entire scope of work performed by Reactor Controls Inc. has been subdivided into a series of tasks. Each task is documented in a Task File which contains information supplementary to computer input/output and is a document essential to understanding the details of the computerized analysis. They are a counterpart of calculations files in an engineering design office.

DESCRIPTION

Task Files are kept as private notebooks by individual engineers performing the particular tasks. Quality Assurance Instructions, Section 3, Paragraph 4.3.2.2, states that computer input data should contain sufficient cross-reference to trace back to the source (e.g., Task File). Without proper documentation and control of the task files, traceability of computer input/output information may be impossible or may lead to erroneous conclusions.

BASIS

This is a violation of Quality Assurance Instructions, Section 3.

IMPACT ON DESIGN

Adverse impact on design was not noted during the inspection but this practice could lead to an error in design.

EXTENT

This item appears to be systematic.

REFERENCES

1. RCI Task File SA-5259-C-1, p. 2, Rev. 0, dated Oct. 1983.
2. RCI QA Instructions, Document QAI-3-1, Section 3, Design Control, Para. 4.3.2.2., Rev. 2, 4/24/84.

D5.3-1 (Deficiency) Unqualified Motor Operated Valve Space Heaters

BACKGROUND

Loop A and B Low Pressure Coolant Injection valves 1E12*F042A and B, located in the Reactor Building, have 120 volt ac space heaters within the Limitorque motor actuator limit switch compartment. These heaters are fed from Class 1E distribution panels and the circuits are classified as safety-related. For example, the 1E12*F042B space heater is fed from Class 1E panel 1SCV*PNL2B1 (References 1, 2, 3, 4 and 7) via Division II cable 1RHSNBC508. Numerous other motor operated valves located inside containment also have space heaters fed from Class 1E power supplies (Reference 1). Space heaters for several Limitorque motor actuators within an electrical division are connected in parallel and are fed from the same power distribution circuit.

DESCRIPTION

The vendor, (Limitorque), has contended that these motor actuator space heaters provide no safety function; they are only provided to help reduce condensation on electrical components during storage. These devices are not qualified by the Limitorque Environmental Test Program. The present design incorporates Class 1E power feeds to these unqualified space heaters. Failure of the space heaters during exposure to accident environmental conditions could potentially cause circuit trips and interrupt power to Class 1E loads or cause the motor actuator to malfunction. Discussions with Stone & Webster engineers revealed that the subject of Limitorque space heater qualification was raised by the Stone & Webster Equipment Environmental Qualification Group in 1980 based on computer sorts of divisional electrical loads. Subsequently, Stone & Webster learned from Limitorque that the subject space heaters were not qualified under the Limitorque qualification program (Reference 6). Stone & Webster's sole action with regard to this issue was to hold a telephone conversation with General Electric and request General Electric to include the space heaters in the GE Phase III equipment qualification program (Reference 8).

BASIS

The Final Safety Analysis Report Table 1.8-1 cites Regulatory Guide 1.75, Revision 2 as the principal regulatory commitment for electrical separation. Stone & Webster Design Criteria 12210-240.200 cites IEEE-384 (1974) as the principal industry design standard. The Stone & Webster design has unqualified Limitorque motor actuator space heater loads supplied by Class 1E power distribution panels in violation of IEEE-384 (1974) and Regulatory Guide 1.75 Rev. 2 commitments for maintaining electrical and physical independence of redundant safety-related equipment. The design was inadequate because it had not been demonstrated that the Limitorque actuator space heaters would not degrade the motor operated valve or Class 1E circuits below acceptable levels when exposed to accident environments. An analysis has not been conducted by Stone & Webster to justify the design and demonstrate that failure of the space heaters will not degrade safety-related equipment below acceptable levels. Based on our review, the team concluded that this design deficiency was not adequately identified and tracked on a project basis by Stone & Webster in a manner that would achieve a technically adequate resolution. We found no worksheets or problem tracking log for this deficiency. In addition, the Limitorque qualification documentation review by Stone & Webster failed to identify this particular problem area. The Stone & Webster telephone call to General Electric does not represent a formal attempt to obtain a resolution of this problem (Reference 8).

D5.3-1 (Deficiency) continued.

IMPACT ON DESIGN

The alternatives available to resolve this deficiency appear to be to implement a design change, qualify the space heater, or perform an analysis to justify the present design.

EXTENT

Unqualified Limitorque motor actuator space heaters are installed in numerous safety-related motor operated valves located inside the containment.

POST CUTOFF WORK

On April 5, 1984 Stone & Webster determined that these space heater circuits would be administatively opened, specifically, the circuit breaker would be placed in the OPEN position after plant operation (Reference 9).

REFERENCES

1. SWEC- Control Circuit Cable Block Diagram, 12210-CBD-7RHS01-1D, Rev. 1C, 8-10-83.
2. SWEC- One Line Diagram, Stby bus A&B Low Voltage Distribution System, 12210-EE-1ZC-1, Rev. 1, 6-23-83.
3. SWEC- Power Distribution, Panelboard Schedule, 12210-CBD-SCV2B1-2, Rev. M2, 9-20-83.
4. SWEC- 480V Wiring Diagram 1EHS*MCC2K Auxiliary Building, 12210-EE-9PG-2, Rev. 2, to be issued.
5. Limitorque Wiring Drawing 15-477-4071-3, Rev. I, 11-6-79.
6. Letter of J. Kirkebo (SWEC) to W. Cahill (GSU); Meeting Notes of December 1, 1983 on Environmental Qualification, RBS-T-13922, 1-9-84.
7. SWEC - Motor Load List, 12-14-83.
8. Telecon of A. Blum (SWEC) to W. Davis (GE), GE Phase III Equipment Qualification Program, 12-22-83.
9. SWEC - Meeting Notes of Environmental Qualification Meeting File 245.000, April 5, 1984.
10. Limitorque Motor Operated Valve Actuator Qualification Report B0058, 1-11-80.
11. SWEC Electrical Independence Design Criteria, 12210-240.200, Rev. 2, 9-10-79, and Rev. 3, preliminary.
12. USNRC Regulatory Guide 1.75, Physical Independence of Electric Systems, Rev. 2, 9-78.
13. IEEE STD-384, IEEE Trial - Uses Standard Criteria for Separation of Class IE Equipment and Circuits, 1974.

D5.4-1 (Deficiency) Inadequate Calculation Assumptions

BACKGROUND

Calculation E-137 defines the criteria, assumptions, and analyses used on River Bend to justify the selection and installation of power cable. The calculation is based on allowable ampacity, voltage drop, and limitation on conductor temperature rise due to short circuit current. In reviewing 600 V branch circuit power cabling 1RHSBBK017 and 18 (Cabling from motor operated valve 1E12*F042B to motor control center 1EHS*MCC2K), we found one documentation discrepancy and one error.

DESCRIPTION

Calculation E-137 assumptions reference project engineering memorandum ETG-IV-4-1 as the bases for using a cable current limitation of 55% of Locked Rotor Amperes for 480V ac motor operated valves based on thermal overload relay settings. The team considers this assumption to be conservative. However, Engineering Memorandum EDVM CHOC 83/18-1, issued after the initial calculation, revised motor operated valve thermal overload relay settings for valves which have their overloads bypassed during design basis accidents to 125% - 140% of Full Load Amperes. We concluded that when calculation E-137 was later reviewed and revised (on September 15, 1983 and March 1, 1984), EDVM CHOC 83/18-1 should have been referenced in the assumption for complete documentation; and rationale for use of 55% Locked Rotor Amperes instead of 125%-140% of Full Load Amperes should have been presented.

Calculation E-137 page 5 assumption (C) states that fault clearing time for circuit breaker protective trips is based on "GE breaker curves TD-4999 and 8088" (References 2 and 3). We found that these curves are from Gould, not GE. Furthermore, these curves are not applicable for this application. They apply to Gould type HE, HL, and LL molded case circuit breakers which are used for loads such as resistive heaters and dry type transformers. They are not used in motor operated valve applications. 1EHS*MCC2K Cubicle 3D consists of a Gould trip coil model A80J10 and size 1 circuit breaker model A821C. The vendor drawing for 1EHS*MCC2K (Reference 7) shows that cubicle 3D contains a full voltage reversing starter with a A822 circuit breaker. The Gould circuit breaker model number (A822) was clarified (Reference 8) as being an A821 full voltage non-reversing starter plus a type A10 reversing contactor. We therefore determined that the engineer used the incorrect time/current trip curves (References 2 and 3) for estimating circuit breaker clearing time. The engineer should have used the Gould time/current curve for the A80 series trip coils (Reference 9). However, the assumption of 0.02 seconds for circuit breaker clearing time remains conservative because Reference 9 shows that the clearing time at 25000 amperes is 0.01[illegible] seconds.

BASIS

The assumption in Calculation E-137 of a cable current limitation of 55% of Locked Rotor Amperes for motor operated valves had incomplete references to a related project Engineering Memorandum (Reference 5) and did not have complete engineering rationale provided. For assumption (C) in calculation E-137 the engineer used the incorrect time/current trip curves for estimating circuit breaker clearing time.

D5.4-1 (Deficiency) continued.

IMPACT ON DESIGN

These two discrepancies did not affect the overall validity of the calculation results or the cabling design and installation.

EXTENT

This type of discrepancy did not appear systematic based on review of other calculations. However, these types of errors, which are considered to be obvious, were not detected by two independent design reviews.

REFERENCES

1. SWEC 600V Cable Sizing Calculation, 12210-E-137, Rev. 2, 3-1-84.
2. Gould Curve TD-4999 Sheets 1 & 2, Time Current Curves, HE Circuit Breaker, Rev. 3, 12-8-76.
3. Gould Curve TD-8088, Time Current Curves, HL & LL Circuit Breakers, Rev. 0, 12-9-76.
4. SWEC Interoffice Memorandum ETG-V-2-3, Selection of Motor Running Overcurrent and Locked Rotor Current Protection, 1-10-75.
5. SWEC Engineering Division Memorandum EDVM CHOC 83/18-1, Clarification of ETG-V-2-3, 7-7-83.
6. SWEC Interoffice Memorandum ETG-IV-4-1, Cable Sizing Criteria.
7. Gould Drawing 84-51380-23 Sheet 1 of 9, Stby MCC 480V ac 1EHS*MCC2K, MCC Layout, SWEC #0242-562-082-113G, 2-9-84.
8. SWEC E&DCR No. C-22187, Clarification for Material Identification.
9. Gould Unitized Circuit Breaker, Instantaneous Time/Current trip curve for A80 Series Trip Coils, SWEC #7242-562-082-002A, June 7, 1982.

D5.5-1 (Deficiency) Cable Installation Routing Error

BACKGROUND

The team selected motor operated valve 1E12*F042B, located inside containment, and reviewed the cable route design documents (Reference 1 and 2) associated with the respective branch circuit power cable (1RHSBBK017). We performed a field walkdown of the installed cable and found that the cable was not installed in a specific tray as required by the design.

DESCRIPTION

Power cable 1RHSBBK017 is installed in raceways in the containment between tie points 1E12*F042B and electrical penetration cabinet 1RCP*TCR14A. The cable and raceway report EC-38 requires the routing of this division II cable to be run in vertical tray 1TK501B prior to top entry into the penetration cabinet. The EC-6 raceway list also shows that cable 1RHSBBK017 is required to be installed in tray 1TK501B; the EC-6 report indicates that the percent tray fill for 1TK501B is 64%. The cable pull ticket (Reference 3) for 1RHSBBK017 shows that this cable was installed in tray 1TK501B; the pull ticket was signed by the installer and was Quality Control approved. We found during our field walkdown of this cable route that cable IRHSBBK017 was not intalled in tray 1TK501B as required by the design documents. The cable was installed such that it completely bypassed vertical tray 1TK501B; the cable dropped directly from horizontal tray 1TK502B into 1RCP*TCR14A as a cable in "free air."

BASIS

Cable 1RHSBBK017 was not installed in tray 1TK501B as required by the pull ticket (Reference 3) and the design documents (References 1 and 2). The pull ticket was Quality Control approved thereby validating an incorrect cable installation.

IMPACT ON DESIGN

References 1, 2, and 3 are incorrect because these design documents list tray 1TK501B as a raceway for cable 1RHSBBK017. Reference 2 has an error which is in the conservative direction in that the percent tray fill for tray 1TK501B includes cable 1RHSBBK017. As a result, design documents now contain errors. The team considers this item to be a construction error.

EXTENT

This item does not appear systematic based on our site inspection activities.

REFERENCES

1. SWEC EC-38 Report, Cable and Raceway Installation Status, Pages 4573 and 4574, Issue 7 , 4-19-84.
2. SWEC EC-6 Report, Raceway List, Page 2199, Issue 82, 5-3-84.
3. SWEC Cable Pull Ticket 1RHSBBK017, Advance Copy, Issue 2A, Revised via markup on 4-24-84, 4-19-84.

D5.8-1 (Deficiency) Drawing Error

BACKGROUND

Gulf States Utilities Nuclear Plant Engineering personnel had compiled Stone & Webster's long standing review comments on General Electric elementary diagrams and incorporated these comments into a Field Deviation Disposition Request in order to expedite General Electric approval and incorporation of these comments into General Electric revised drawings. As a result of this effort, Field Deviation Disposition Request LDI-925 Revisions 0 and 1 were developed and issued.

DESCRIPTION

Stone & Webster incorporated revision 0 and revision 1 of Field Deviation Disposition Request LDI-925 (which was unapproved by General Electric San Jose) onto Stone & Webster vendor prints of the General Electric elementaries and reissued these drawings on a project approved risk basis. However, Stone & Webster inadvertently missed changes to Division I and Division I Associated separation group identifiers on the elementaries in two cases. On drawing 828E535AA Sheet 9, Stone & Webster incorporated revision 0 comments but did not incorporate revision 1 comments. Similiarly, on drawing 828E535AA sheet 5, Stone & Webster incorporated revision O comments but did not incorporate revision 1 comments.

BASIS

The Stone & Webster revised drawings were not consistent with Field Deviation Disposition Request LDI-925 Revision 1.

IMPACT ON DESIGN

This item had no effect on design or analysis. The drawing errors are considered to be minor. Because Stone & Webster proceeded to revise drawing on a risk basis, we believe that these errors would have been detected upon receipt and review of the General Electric approved version of the Field Deviation Disposition Request.

EXTENT

These types of drawing errors did not appear systematic based on our extensive review of drawings during the inspection.

REFERENCES

1. GE FDDR No. LDI-925, Rev.0, 8-8-83.
2. GE FDDR No. LDI-925, Rev.1, 10-14-83.
3. GE Elementary Diagram 828E535AA, Low Pressure Coolant Injection, Sh.9, Rev.4.
4. GE Elementary Diagram 828E535AA, Low Pressure Coolant Injection, Sh.5, Rev.3.

D5.8-2 (Deficiency) Drawing Error

BACKGROUND

Stone & Webster revised analog wiring diagram 12210-AWD-1-1.67 as a result of main control room changes associated with Change Request Form CF-0300. This modification incorporated addition of temperature recorders 1RHS*TR47A and B to Residual Heat Removal panel inserts on main control board 1H13-P601 in compliance with Regulatory Guide 1.97 Revision 2 recommendations.

DESCRIPTION

Stone & Webster revised drawing 12210-AWD-1-1.67 to incorporate Division I and II power supplies for safety-related temperature recorders 1RHS*TR47A and B respectively. Initially the drawing contained no divisional circuits. However, after addition of the divisional power supplies, the drawing safety-related identifier was not revised as required. The drawing (Reference 1) was issued with the identifier remaining unchanged "QA CAT II BLACK". The identifier should have been changed to "QA CAT I RED and CAT I BLUE".

BASIS

The safety-related identifier on drawing 12210-AWD-1-1.67 was incorrect.

IMPACT ON DESIGN

This item had no effect on design or analysis. This item is considered to be a minor documentation discrepancy.

EXTENT

These types of drawing errors do not appear systematic based on our extensive review of drawings during this inspection.

REFERENCES

1. SWEC Analog Wiring Diagram, Instrumentation Power Supply and Distribution, Panel 1H13-P869, 12210-AWD-1-1.67, Rev. 6, 1-6-84.
2. SWEC Change Request Form CF-0300, 11-9-83.

D5.8-3 (Deficiency) Field Deviation Disposition Request Error

BACKGROUND

Stone & Webster engineers prepared main control room Change Request Form CF-0288. This change modified the design of existing 2-position control switches by incorporating 2-position "spring-return-to-center" control switches on panel H13*P870 to accomplish throttling control for valves 1E12*FO68A and B (Heat Exchanger Service Water Discharge Valves). In response to Change Request Form CF-0288, General Electric field engineers developed Field Deviation Disposition Request LDI-1331, which provided details of wiring, device list, and assembly drawing changes to incorporate this modification. Stone & Webster site personnel approved implementation of the Field Deviation Disposition Request by issuing Engineering and Design Coordination Report C-51164.

DESCRIPTION

We found several errors on sketches and marked-up drawing sections attached to Field Deviation Disposition Request LDI-1331. The sketch of General Electric drawing 386X954-364A was in error. The switch contact diagram detail for 1-1SWPA37 and 1-1SWPB37 was listed as "detail AA" instead of detail "AP" as required by the Stone & Webster elementary ESK-6SWP38. This error resulted in reference to the incorrect switch configuration. In addition, the mark-up of General Electric assembly drawing 944E730 Sheet 2, H13-P870 insert 55C, was incorrectly labled "insert 55B" instead of "insert 55C" as required. This mark-up also did not show item "P5" as required because this portion of the drawing was apparently omitted during reproduction.

BASIS

The sketches and marked up drawings attached to Field Deviation Disposition Request LDI-1331 contained several errors.

IMPACT ON DESIGN

These errors appeared to have no impact on design or analysis. We determined that the correct type of control switch had been procured (Reference 4 and 5). We believe that the remaining portions of the Field Deviation Disposition Request were sufficiently clear to allow field personnel to correctly implement the required modification not withstanding the discrepancies.

EXTENT

This error does not appear systematic based on our review of several Field Deviation Disposition Request documents.

POST CUTOFF WORK

Engineering and Design Coordination Report C-51164 was issued on March 17, 1984 to accommplish the required modifications.

D5.8-3 (Deficiency) continued.

REFERENCES

1. SWEC PGCC Change Request Form CF-0288, Rev. 1, 8-3-83.
2. GE FDDR No. LDI-1331, Rev.0, 2-13-84.
3. SWEC E&DCR C-51164, 3-17-84.
4. GE Electrical Device List EDL PL442X897, Panel H13-P870 insert 55C, Rev.2, Sheets 2 and 3.
5. GE Electrical Device List EDL PL442X899, Panel H13-P870 insert 56C, Rev.2, Sheets 2 and 3.
6. GE drawing 386X954-364A, (SWEC Elementary 12210-ESK-6SWP38, Rev.7).
7. SWEC Elementary Diagram 12210-ESK-6SWP38, Rev.7, 10-13-83.
8. GE Assembly Drawing 944E730, Sh. 2, 1H13-P870 Insert 55C.

D5.10-1 (Deficiency) Documents Inconsistent

BACKGROUND

During the site inspection the team noted that General Electric local instrument rack H22-P002, located inside containment, contained both safety-related and non safety-related instrumentation. We observed that Reactor Pressure Vessel Head Transmitter E31*N092 had a red Division I separation group nameplate; all other instruments on H22-P002 had non safety-related (black) nameplates. After review of General Electric design documents conducted as part of our evaluation of main control room field changes, the team found several inconsistencies with respect to the safety classification of transmitter E31*N092.

DESCRIPTION

The team noted several inconsistencies with respect to classification of transmitter E31*N092, therefore we reviewed the General Electric master parts list (Reference 2) to determine classification status definitions for equipment within General Electric scope of supply. Reference 2 provides these definitions:

Equipment Classification Codes (Reference 2 Column "EC") -

A - essential item, automatic active safety function
N - non-essential item
P - essential item, passive safety function

Code Classification (Reference 2 Column "CC") -

I - Instrument or electrical item not having a pressure boundry class
E - Mechanical or structural items not covered by code class groups A, B, C, D.

We noted that General Electric elementary diagrams (References 9 and 10) show that E31*N092 provides an alarm function only. E31*N092 is wired to the Low Pressure Coolant Injection System circuitry as a "Division I Associated" device (Reference 9).

We determined that an analysis was not performed to demonstrate that failure of this transmitter will not degrade the safety-related circuit below acceptable levels, therefore qualification is required. We also determined that General Electric intends to environmentally qualify E31*N092 under the General Electric Phase III qualification program (Reference 11). We therefore concluded that E31*N092 should be classified as an essential item that has a passive safety function (i.e., the device must not fail in a manner which could degrade the safety-related circuitry to which it is connected). We believe that the equipment classification code should be "P" as defined in Reference 2.

Because the E31*N092 process line is connected between the inner and outer reactor vessel head seal, we concluded that the code classification should be seismic Catgory I and pressure boundary should be maintained. We determined the following with respect to instrument code classification. General Electric design specification for process instrumentation 22A3137 applies to pressure boundary instruments; Section 4.1 defines instruments connected to process lines or the reactor vessel via tubing or piping (i.e.., E31*N092) as "remote" instruments. General Electric Design Specification 22A3137 is the governing standard for pressure integrity of remote instruments; Section

D5.10-1 (Deficiency) continued.

4.3.4.1 requires that instrument bodies in contact with process fluid shall comply with the material requirements for Group D (ASME Section 8 or ANSI B31.1) components in the General Electric Pressure Integrity Specification A62-4030. In addition, General Electric Product Safety Standards 22A8400 Figure 1, defines rack mounted instrumentation (i.e., E31*N092) as Code Classification "I" (non-code). Section 2.4 of 22A8400 requires a Certificate of Conformance which confirms that General Electric purchase specification requirements are met. Material certification is not required. However, when instrumentation is classified safety-related, Quality Assurance per 10CFR50 Appendix B, Seismic Category I and environmental qualification requirements must be satisfied. Pressure integrity must be ensured if the instrument forms part of the pressure boundry. We concluded that Transmitter E31*N092 is a remote instrument with Code Classification "I" in accordance with 22A8400. As stated above, E31*N092 is a "Division I Associated" device which requires environmental qualification. Therefore, in accordance with Section 2.4 of 22A8400, 10CFR50 Appendix B Quality Assurance requirements, Seismic Category I requirements, and environmental requirements must be met.

The tabulation presented below reflects the inconsistencies in equipment classification for E31*N092 in General Electric documents, References 3, 1, 7, 4, and 8.

Document	Equip. Class	Code Class	Design Responsibility
Reference 3	N	I	GE RBS
Reference 1	A	I	GE RBS
Reference 7	P	I	GE (Standard Plant)
Reference 4	A	I	GE RBS
Reference 8	P	I	GE RBS
Reference 6	Div. I	-	GE RBS
Reference 10	Div. I	-	GE RBS
Reference 9	Assoc. Div. I	-	GE RBS
Reference 5	Safety-related	-	SWEC

In summary, General Electric documents (References 3, 1, 7, 4 and 8) reflect inconsistencies in the equipment classification of E31*N092 as defined in Reference 2. We believe that the equipment classification for E31*N092 should be "P" (i.e., essential item, passive safety function).

BASIS

The equipment classification code for E31*N092, as defined in Reference 2, is inconsistent within General Electric documents (References 3, 1, 7, 4, and 8). The team believes that the equipment classification should be "P" (i.e., essential item, passive safety function). In addition, the team believes that the code classification "I" for E31*N092, as defined in Reference 2, is correct. Because the device is safety-related and environmental qualification is required, the device should be seismic Category I and maintain a pressure boundry in accordance with Reference 13. Therefore, an equipment classification of "N" is incorrect in our judgement.

IMPACT ON DESIGN

This item has no affect on the design or analysis. The inconsistencies with respect to equipment classification are considered to be minor documentation discrepancies within General Electric design documents for E31-N092. The General Electric Phase

D5.10-1 (Deficiency) continued.

III qualification program will address the teams concern on the code classification of E31-N092, therefore, this is also considered a documentation discrepancy.

EXTENT

These discrepancies did not appear to be systematic based on our limited review.

REFERENCES

1. GE Elementary Diagram Device List, LPSI System, DL828E53AA, Sheet 9, Rev. 7.
2. GE Master Parts List (MPL), River Bend Station, 18NS06B04, Rev. 8, 5-25-83.
3. GE Master Parts List (MPL), Leak Detection System, MPL 283M283AA Sheet 1, Rev. 16.
4. GE River Bend Instrument Data Work Sheet E31-N092, 11-11-83.
5. SWEC Instrument Bill of Material, page 99, 1-31-84.
6. GE Connection Diagram 147D8397, Rev. 0, Local Panel H22-P002, SWEC #0222-440-000-082A, 7-14-81.
7. GE Elementary Diagram Device List (EDDL), DL828E535BA, Rev. 11, Sheet 9.
8. GE Device List EDL 368X542BA, Rev. 3.
9. GE Elementary Diagram 828E535AA Sheet 10, Rev. 7.
10. GE Elementary Diagram 828E535AA Sheet 9, Rev. 4.
11. Letter of H. D. Powell (GE) to A. Blum (SWEC), Environmental Qualification Program GSS-4114, 2-22-84.
12. GE IED Leak Detection System 762E293AA Sheet 4, Rev. 1.
13. GE Product Safety Standards, Safety Criteria, Generic Boiling Water Reactor Plant, 22A8400, Rev. 1, 11-1-83.
14. GE Process Instrumentation Design Specification 22A3137, Rev. 6, 8-16-83.

D5.10-2 (Deficiency) Conduit Sealing

BACKGROUND

During the site inspection, the team observed that General Electric supplied safety-related local instrument rack H22-P002, located in-containment, had a divisional rack mounted NEMA 4 junction box which contained terminal blocks for termination of safety-related field cables. We determined that General Electric requires the conduit penetrations on this junction box to be completely sealed by the Architect/Engineer to eliminate the containment post accident environment inside the box, thereby maintaining qualification of the terminal blocks. We found that Stone & Webster did not provide technical details to the electrical installer to conform to General Electric criteria. We also found that conduit was installed in violation of General Electric design criteria.

DESCRIPTION

We reviewed the assembly drawing, parts list, and purchased part drawings (References 3 thru 6) for General Electric local instrument rack H22-P002. We determined that the divisional junction box for termination of safety-related field cabling was a NEMA 4 design steel enclosure with a solid neoprene cover gasket and General Electric type CR151B terminal blocks. Eleven safety-related local instrument racks located in-containment have similar divisional junction boxes (i.e. H22-P004 and H22-P005).

We determined at our meeting at General Electric Valley Forge that General Electric Boiling Water Reactor Equipment Environmental Interface Data Specification 22A6926, Section 5.2.1, requires that the Architect/Engineer seal conduit penetrations of enclosures on General Electric supplied local instrument racks and electronic panels. The purpose of the seal is to protect internals from harsh containment environments following an accident. The General Electric design philosophy was to maintain an environmental seal on the enclosure (i.e. rack mounted junction box) to protect and thus qualify the internal devices (i.e. terminal blocks).

Stone & Webster's method of complying with General Electric conduit sealing requirements at divisional rack mounted junction boxes was to reference the General Electric Specification 22A6926 in Section 2.21 of the Electrical Installation Specificaton 248.000. We reviewed the actual details provided to the electrical installer by Stone & Webster to terminate cabling and conduit on the divisional junction box on H22-P002 in compliance with General Electric requirements. We determined that wiring diagram EE-4B-1 showed cable 1CSLNRX408 (for transmitter E31-N092) connected to the terminal block within the Division I junction box at H22-P002 (References 7 and 8). We also determined that conduit drawing EE-460AC-4 and support detail EE-450-AM-2 showed conduit approach, seismic support, and standard flexible conduit entry into the subject junction box. We found that specific technical details on conduit sealing to meet General Electric requirements were not provided to the field by Stone & Webster in Installation Specification 248.000 or the drawings.

During our site inspection, we observed that conduit had not yet been connected to the divisional junction box at rack H22-P002. However, American BOA type B1-0 stainless steel unbraided corrugated leak tight metal hose was installed at Division I junction boxes on in-containment General Electric rack H22-P004 (conduit 1CX504RB and 1CX504RP). Similarly, Anaconda sealtite N.W.C. flexible conduit (1CX5400A and 1CX5400F) was installed at Division III junction box on in-containment General Electric rack H22-P005. We determined that these flexible conduits were procured as non-engineered items (Reference 11) with no specific sealing requirements or

D5.10-2 (Deficiency) continued.

qualification documentation. Discussions with Stone & Webster engineers confirmed that both these conduit installations and the conduit sealing design do not comply with the General Electric conduit sealing requirement.

We concluded that the Stone & Webster design did not adequately incorporate General Electric conduit sealing requirements.

BASIS

Section 5.2.1 of General Electric Specification 22A6926 requires that Stone & Webster environmentally seal conduit penetrations on divisional junction boxes mounted on General Electric supplied local instrument racks located in-containment. The General Electric philosophy was to protect the terminal blocks from harsh post accident environments.

We found that Stone & Webster Installation Specification 248.000 cited General Electric Specification 22A6926 as a requirement. However, specific technical details on junction box sealing to meet the stringent General Electric requirements were not provided to the field in Specification 248.000 or the design drawings. We also found that actual conduit installations at H22-P004 and H22-P005 violated the General Electric requirements.

IMPACT ON DESIGN

The electrical design for conduit entry and sealing onto divisional junction boxes mounted on General Electric instrument racks violates General Electric Specification 22A6926; therefore a design change, analysis or testing is required.

EXTENT

A total of 11 safety-related General Electric local instrument racks, located in containment contain divisional NEMA 4 enclosures with General Electric CR151B terminal blocks connected to safety-related circuits.

REFERENCES

1. GE BWR Equipment Environmental Interface Data 22A6926, Rev. 0, SWEC # 0224-50300-0002B, 12-27-79.
2. SWEC Electrical Installation Specification 248.000, Rev. 7, 12-30-83.
3. GE Drawing, Reactor Water Clean-up Local Panel H22-P002, 184C5463, Assembly Drawing, Rev. 0, 3-8-81.
4. GE Parts List H22-P002, PL184C5463, Sh. 2, Sec. A, Rev. 1, 8-20-81.
5. GE Assembly Drawing, Terminal Box, 164C5999, Sh. 1, Rev. 4, 4-8-78.
6. GE Enclosure, Drawing Purchased Part Drawing, 272A7074, Rev. 1.

D5.10-2 (Deficiency) continued.

7. GE Connection Diagram, H22-P002, 147D8397, Rev. 0, SWEC # 0222-440-000-082A, 6-12-81.
8. SWEC Wiring Diagram 1H22*PNLP002, Reactor Building, 12210-EE-4B-1, Rev. 1, 6-15-83.
9. SWEC Seismic Conduit Installation Plan, Elev. 141' Reactor Building, 12210-EE-460AC-4, Rev. 4, 2-13-84.
10. SWEC Seismic Conduit Support Standard Details, 12210-EE-450AM-2, Rev. 2, 1-25-84.
11. SWEC Non-Engineered Item Data Sheet Page 1009A, Rev. 10, SWEC Specification 211.161.

D5.10-3 (Deficiency) Terminal Block Qualification

BACKGROUND

During the site inspection the team observed that General Electric supplied local instrument rack H22-P002, located inside containment, had a Division I rack mounted NEMA 4 junction box which contained terminal blocks for termination of safety-related field cables. Eleven General Electric local instrument racks, located inside containment, contain safety-related equipment wired to terminal blocks within rack mounted NEMA 4 divisional junction boxes. We learned at a meeting at General Electric Valley Forge on April 26, 1984 that the GE type CR151B terminal blocks on H22-P002 were generic for all General Electric racks. Therefore, we reviewed the environmental qualification for these terminal blocks to determine their ability to provide sufficient insulation characteristics during and after exposure to accident environments. We found that adequate qualification documentation did not exist for these devices.

DESCRIPTION

The team discussed local instrument rack terminal block qualification with General Electric at their San Jose and Valley Forge offices. We reviewed the assembly drawings, part lists, and purchased part drawings for local instrument rack H22-P002 (References 1 thru 5, and 11). The subject junction box was a NEMA 4 design steel enclosure with a solid neoprene cover gasket and two 12 point General Electric CR151B2 terminal blocks. We determined from the Design Record File for H22-P002 that the subject rack was shipped to the site in mid-1981. General Electric Product Quality Certification (Reference 6) indicated that there were no non-conformances concerning procurement quality requirements. The rack was released for conditional shipment due to deferred verification of the elementaries (Reference 7).

We reviewed the testing and analyses performed on the General Electric CR151B terminal block to substantiate qualification under accident environments. Report A00-794-2 was an accelerated aging test of CR151B2 terminal blocks and other devices to simulate a 40 year qualified life for Class 1E passive devices which are located in the main control room. Although the report concluded that the terminal block and all other devices were qualified for 40 years in a main control room environment, we found that technical details were not provided concerning the basis and justification for the selection of temperature and test duration relative to materials and normal ambient conditions. The report was not applicable to in-containment applications. The General Electric Bloomington Report (Reference 13) was a thermal aging and irradiation test of General Electric CR151B terminal blocks. Again, technical details were not provided to justify accelerated aging conditions; and the irradiation test dose (1E06 Rad) was less than the postulated accident condition in-containment at River Bend (1.7E07 Rad).

General Electric Memorandum 994-79-008 references a General Electric design specification 22A3033. This specification describes a test plan of a NEMA 4 junction box with CR151 terminal blocks and Anaconda flexible conduit and fittings to demonstrate pressure integrity of the junction box and terminal block qualification. The test plan required both ends of a single test conduit to terminate on the junction box; the energized circuit was to be connected at a separate epoxy sealed fitting. General Electric memorandum 994-79-008 states that the test was conducted for 28 days at 220°F and 17 psig. The team noted that the actual test results were not provided in this package and there was little distinction between the test plan and the actual test results; however, the reported test environment enveloped the River Bend in-containment environment

D5.10-3 (Deficiency) continued.

with the exception of radiation. Details with respect to leakage current to ground and test results were not provided. This package was considered by the team to be incomplete; documented evidence of qualification was not presented in an auditable form.

General Electric Boiling Water Reactor Equipment Environmental Interface Data 22A6926, Section 5.2.1, states that the Architect/Engineer shall seal conduit penetrations of enclosures to General Electric supplied safety-related instruments to protect the internals from containment environments. We were told at our meeting with General Electric at Valley Forge that General Electric initially believed that maintaining an environmental seal on the enclosure would protect and thus qualify the terminal blocks. However, this design philosophy proved inadequate as Architect/Engineer's found it impossible or impractical to maintain a sealed conduit system at the enclosure. In July, 1982, General Electric issued a memorandum to projects requesting that Architect/Engineer's review the installation design as it interfaces wih local racks (Reference 8). On August 4, 1982 Stone & Webster was advised of this problem (Reference 9). Subsequently, the task of qualification of General Electric enclosures with General Electric CR151B terminal blocks was added to the General Electric phase III qualification program (Reference 15).

A non-conformance report concerning the lack of auditable documentation of the General Electric CR151B terminal blocks should have accompanied the rack at shipment. However, it appears that these terminal blocks were procured commercial grade and were not considered safety-related because the parts list and purchased part drawing do not indicate that these components are safety-related (References 4 and 5). Discussions at General Electric Valley Forge substantiated this contention.

BASIS

Although General Electric had performed various environmental qualification tests of terminal blocks (References 12, 13, and 14) prior to and at the time of release of instrument rack H22-P002, these tests were separate (unrelated), incomplete, and not compiled into a complete package which satisfactorily demonstrated all relevant aspects of environmental qualification for in-containment applications. We judged the qualification documentation for the General Electric CR151B terminal block for use in-containment to be inadequate.

Section 3.11.2.1 of the Final Safety Analysis Report requires that safety-related equipment for River Bend be qualified to the Criteria of NUREG-0588 for Category I plants (Construction permit SER dated July 1, 1974 or later), the guidelines presented in Regulatory Guide 1.89, and the requirements of IEEE-323 (1974). We concluded that fabrication, shipment and installation of safety-related in-containment instrument racks using unqualified terminal blocks violates this FSAR commitment because a non-conformance report was not developed to track and resolve this problem.

IMPACT ON DESIGN

Qualification is not substantiated for these terminal blocks. A design change analysis or testing is required.

D5.10-3 (Deficiency) continued.

EXTENT

A total of eleven General Electric local instrument racks located in-containment contain General Electric type CR151B terminal blocks connected to safety-related circuits of redundant electrical divisions.

REFERENCES

1. GE Drawing Reactor Water Clean-up Local Panel H22-P002 184C5463, Assembly Drawing, Rev. 0, 3-8-81.
2. GE Parts List H22-P002, PL184C5463, Sh. 2, Sec. A, Rev. 1, 8-20-81.
3. GE Assembly Drawing, Terminal Box, 164C5999, Sheet 1, Rev. 4, 4-8-78.
4. GE Parts List, Terminal Box, PL164C5999, Sh. 2, Sec. A, Rev. 6, 8-27-81.
5. Drawing - Terminal Board, Purchased Part 174B9231, Rev. 8, 11-25-80.
6. GE Product Quality Certification, H22-P002, 7-15-81.
7. GE Engineering Instructions, WZ-00-2, Page 107, 5-7-81.
8. GE Memo, Environmental Qualification Interface Requirements, File 3.65, F. E. Hatch, 7-9-82.
9. Letter of H. D. Powell (GE) to A. Blum (SWEC), Environmental Qualification Interface Requirements, 8-4-82.
10. GE BWR Equipment Environmental Interface Data 22A6926, Rev. 0.
11. GE Enclosure Drawing, Purchased Part Drawing, 272A7074, Rev. 1.
12. GE Qualified Life Test Report to IEEE-323(74) of Selected Passive Devices, A00-794-2, Rev. 00, 3-27-80.
13. GE Qualification Report to IEEE-323(74), CR151B Terminal Blocks, GE Bloomington, 11-19-81.
14. GE Memo. Production Engineering Memorandum 994-79-008, Test Report Evaluation of Terminal Box in LOCA Environment, 6-4-79.
15. Letter of H. D. Powell (GE) to A. Blum (SWEC) Environmental Qualification Program, 2-22-84.
16. River Bend Final Safety Analysis Report, Environmental Qualification Document (EDQ), Amendment 12, 3-29-84.
17. NRC Regulatory Guide 1.89, Qualification of Class IE Equipment for Nuclear Power Plants, 11-74.
18. IEEE STD 323, Qualifying Class IE Equipment for Nuclear Power Generating Stations, 1974.
19. USNRC NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Regulated Electrical Equipment, Rev. 1, 7-81.

D5.11-1 (Deficiency) Use of Undocumented Data

BACKGROUND

Stone & Webster Engineering Assurance Procedure 5.3, attachment 1.4, requires listing of all documents related to or supporting the calculation. The listing should be in the reference section of the calculation.

DESCRIPTION

Standby Batteries IENB*BAT01A & B size verification calculations E-149 and E-150 did not list the Gould (vendor) document from which an important data item (capacity rating factor) used in the calculation was taken. The engineer stated that he received this information by telephone. No document could be identified (phone memo, instruction leaflet, etc.) to verify the data. Document control did not have an instruction leaflet from Gould which would verify the capacity rating factor values used in the calculation.

BASIS

Both Revision O and Revision 1 of the calculation were reviewed and independently reviewed by the same person, as allowed by the procedure (Reference 1). They were then distributed to the electrical division and the specialist. The review process did not identify the deficiency which might have resulted in equipment inadequacy. The Electrical Control Group identified an uncontrolled Gould instruction leaflet that was not in the document control file at the time of inspection, stating that the values from the leaflet may substantiate the data used in the calculation.

IMPACT ON DESIGN

The documentation needs to be changed. Vendor data from Gould should be used to determine whether the capacity rating factor used in the calculation was correct.

POST CUTTOFF WORK

Stone & Webster has since revised this calculation to include the manufacturer data reference, battery discharge characteristic table and graph, and the tables of the battery discharge characteristics from the manufacturer's instruction leaflet, including that for Gould's NCX-2100, supplied to the River Bend project. We have reviewed this calculation (Reference 5.11.8) and observed no discrepancy.

EXTENT

This deficiency did not appear to be systematic based on review of other documents.

REFERENCES

1. SWEC Procedure, Engineering Assurance Procedure, EAP 5.3, Rev. 3, January 31, 1979.
2. SWEC Calculation E-149, Verification of Battery Size, IEN*BAT01A, Rev. 2, December 23, 1983.
3. SWEC Calculation E-150, Verification of Battery Size, IENB*BAT01B, Rev. 2, December 12, 1983.

D6.3-1 (Deficiency) GE "Design Bases" Inconsistency with the FSAR

BACKGROUND

A General Electric design bases document (Reference 1) lists general design criteria, Regulatory Guides, and IEEE Standards applicable to the River Bend project. For certain items in the list there is an entry which prohibits River Bend General Electric design from using those guides and standards unless an Engineering Change Authorization or Engineering Change Notice from the project manager directs the General Electric design to comply with the requirements.

DESCRIPTION

River Bend FSAR Chapter 7 and 8 includes those Regulatory Guides and Standards which are listed with the requirement of Engineering Change Authorization or Engineering Change Notice in General Electric's design bases document (Reference 1). No Engineering Change Authorization or Engineering Change Notice was processed to require General Electric's scope of River Bend design to comply with these guides and standards. General Electric maintained that their design complies with the requirements of these guides and standards as per the FSAR.

IMPACT ON DESIGN

None

EXTENT:

Out of a large number of documents in Residual Heat Removal and Automatic Depressurization Systems control and instrumentation design that were reviewed by the team, one additional item was noted to have inconsistent information on General Electric documents (see Deficiency D6.3-2). This does not appear to be a systematic deficiency as the sample was a small part of the General Electric scope of River Bend design.

POST CUTOFF WORK

As a result of the team's identification of this error, General Electric issued a change notice (Reference 2) revising the design basis document (Reference 1) to remove the requirement of an Engineering Change Notice or Engineering Change Authorization for compliance to the specific regulatory guides and standards.

REFERENCES

1. GE Specification, Design Bases, Regulatory Requirements and Industrial Standards Document No. 22A52'6 MFL #A42-4070, Revision 2, 10/19/81.
2. GE Documents, ECN No. NH 19140, Revision 0, 5/25/84.

D6.3-2 (Deficiency) Incorrect Information on General Electric Drawing

BACKGROUND

Stone & Webster letters (References 1 and 2) requested General Electric to make a correction on a General Electric Residual Heat Removal System elementary diagram. The letters indicate the same item for correction in both requests and that General Electric had agreed to make the change.

DESCRIPTION

The Residual Heat Removal System Elementary diagram was revised on April 2, 1984; however, the requested change was not incorporated in the drawing. This constituted inconsistency between Stone & Webster sketch (reference 4) which includes the correct information and the General Electric drawing with incorrect information.

IMPACT ON DESIGN

None

EXTENT

Of a large number of documents in Residual Heat Removal System and Automatic Depressurization System control and instrumentation design that were reviewed by the team, only one additional item was noted to have inconsistent information (see Deficiency D6.3-1). This does not appear to be a systematic deficiency.

POST CUTOFF WORK

The drawing was revised on April 2, 1984 and did not include this change.

REFERENCES

1. S&W Letter to GE - RBV-1255, 8/17/79.
2. S&W Letter to GE - RBV-2063, 4/18/84.
3. GE Drawing, Elementary Diagram 828E534AA, Ch. 8, Revision 11, 4/2/84.
4. S&W Drawing, Sketch #ESK-715C01, Revision 6, 9/22/83.
5. GE Document, FDDR No. LOI-1614, Revision 0, 5/4/84.

D6.4-1 (Deficiency) Standby Service Water Manual Valves for High Pressure Core Spray D/G Cooling

BACKGROUND

General Electric design criteria for the General Electric-supplied High Pressure Core Spray and integral diesel generator require that the system be capable of operating independently of normal auxiliary AC power, plant service air, or the emergency cooling water system. The FSAR also states that the dedicated High Pressure Core Spray diesel generator is operable as an isolated system independent of electrical connection to any other system.

To satisfy this independence requirement, the High Pressure Core Spray ordinarily includes an engine-driven pump to provide water flow to its diesel generator jacket cooler. However, this pump was eliminated in the River Bend design.

The High Pressure Core Spray diesel generator purchase specification states that the engine can supply full load for at least two minutes without cooling water from a standby condition. Consequently, the off-site and on-site AC power systems must fulfill this requirement to satisfy this dependency on the emergency cooling water system.

A design change was requested by Stone & Webster in early 1974 so that the redundant Standby Service Water System loops could provide cooling water to the dedicated High Pressure Core Spray diesel generator jacket cooler rather than use a separate diesel driven pump normally provided by General Electric. As a result, operation of the Division 3 High Pressure Core Spray core cooling system is no longer independent of the Division 1 and 2 emergency AC power sources (see Observation O6.4-1).

In each connecting coolant pipe between the Standby Service Water System and the High Pressure Core Spray diesel generator, one check valve and one normally-open manual motor-operated valve are provided. These valves isolate the redundant Standby Service Water loops for a postulated coolant pipe through-wall leakage crack. They can also be used to isolate the High Pressure Core Spray diesel generator for maintenance. During maintenance, all four valves will be closed whereas only one set of two particular valves need be closed under the pipe crack condition.

To assure the capability to isolate a postulated return line break, the inlet valve from one Standby Service Water loop is powered by one redundant Class 1E source and the return valve for that same loop is powered by the other redundant Class 1E source. This arrangement requires that either off-site power or both Divisions 1 and 2 AC power sources be available should the valves need to be reopened to restore High Pressure Core Spray diesel generator cooling flow from either Standby Service Water loop.

DESCRIPTION

Under postulated Loss-Of-Coolant-Accident conditions, the four manual MOVs are not automatically aligned to their full open position. Special administrative controls to preclude inadvertent closure prior to a postulated LOCA, such as keylocked main control board switches, racked-out motor control centers, or chain-padlocked manual valve wheels, are not provided in the design. In addition, the operator is not informed

D6.4-1 (Deficiency) continued.

at the system level that the High Pressure Core Spray is rendered inoperable (bypassed) when these manual MOVs are not fully open.

The majority of High Pressure Core Spray "operability" indications provided to the operator, such as High Pressure Core Spray diesel generator cooling flow, High Pressure Core Spray cooling water temperature, and various High Pressure Core Spray and Standby Service Water annunciator alarms, use Quality Assurance Category II non-safety-related equipment. However, position lights for the High Pressure Core Spray cooling water valves are designed as Quality Assurance Category I items. High Pressure Core Spray diesel generator cooling prior to the onset of an accident depends on operator administrative control to ensure the valves remain open.

BASIS

10CFR50 Appendix A General Design Criterion 20 states "The protection system shall be designed ...to sense accident conditions and to initiate the operation of systems and components important to safety." In addition, General Design Criterion 22 states that "The protection system shall be designed to assure that the effects of ....maintenance....on redundant channels does not result in loss of the protective function." In this situation, a LOCA signal has not been implemented to assure correct valve alignment for High Pressure Core Spray diesel generator cooling.

IMPACT ON DESIGN

Hardware and documentation changes are anticipated, but re-analysis should not be required.

EXTENT

This particular item appears to be confined to the High Pressure Core Spray diesel generator jacket cooler and High Pressure Core Spray room unit coolers.

REFERENCES

1. FSAR Figure, 9.2-1a, Service Water P&ID.
2. FSAR Page 8.3-15, HPCS Power Supply, page 8.3-15.
3. GE Purchase Spec., 21A9236, HPCS D/G, page 9, rev. 4, 7/9/75.
4. Stone & Webster Flow Diagram, FSK-9-10F, Service Water System, Revision 2, 5/6/74, Revision 3, 2/15/77, and Revision 4, 5/2/78.
5. FSAR Page 8.3-60, HPCS Diesel Generator Unit.
6. NRC Branch Technical Position PSB-2, Revision 0, 7/81.

D6.4-2 (Deficiency) Logic Sketch Error in Valve Assignments

BACKGROUND

In the Standby Service Water System, power assignments to the cooling return line motor-operated valves 1SWP*MOV506A and B are provided from essential power sources B and A respectively. This power assignment was required by Stone & Webster to permit isolation of the return lines for a postulated break coincident with loss of one AC power source.

DESCRIPTION

The Stone & Webster logic diagram incorrectly depicted that valve 1SWP*MOV506A was connected to Division I inoperability lights rather than those associated with Division II. The schematic diagram, however, correctly depicts the power assignment.

BASIS

A power supply assignment inconsistency exists in one logic diagram relative to the flow diagram, other sheets of the logic diagram, and the schematic diagrams for this system.

IMPACT ON DESIGN

This error has no hardware or analysis impact.

EXTENT

This error appeared to be random.

POST CUTOFF WORK

A change (Reference 2) was issued on 5/4/84.

REFERENCES

1. Stone & Webster Logic Diagram, LSK-9-10.3S, Revision 9, 3/4/83.
2. SWEC E&DCR P21912, 5/4/84.

D6.5-1 (Deficiency) RTD Classification Drafting Error

BACKGROUND

Resistance temperature detectors at the outlets of all three diesel generators cooled by the Standby Service Water System were downgraded by Stone & Webster in the 1979-1980 period from safety-related to non-safety-related. The team had no disagreement with this classification change.

DESCRIPTION

Temperature measurements 1SWP-RTD78A, 1SWP-RTD78B, and 1SWP-RTD132 were shown as safety-related on three Analog Wiring Diagrams even though these sheets were labelled as Quality Assurance Category II (non-safety-related). The sensors were correctly depicted on the Standby Service Water System P&ID, its elementary diagram, and the sensor purchase specification.

BASIS

The Analog Wiring Diagrams were inconsistent with other design documents for these sensors.

IMPACT ON DESIGN

This item had no impact on design or analysis. It was considered to be a minor documentation error, corrected by revision to the drawings during the inspection.

EXTENT

This error did not appear to be systematic based on review of other diagrams.

POST CUTOFF WORK

Correction of the three affected sheets was accomplished on April 27, 1984.

REFERENCES

1. FSAR Figure, 9-2.1a,b,c,d; SWP System P&ID.
2. SWEC Technical Data Sheet 247.481, RTD Sensor, page 3-11-8, Revision 0, 12/3/80.
3. SWEC Analog Wiring Diagrams, AWD-9-10.23, Non-1E Service Water, Revision 6, 1/28/83; AWD-9-10.24, Non-1E Service Water, revision 06, 1/28/83, and AWD-9-10.34, Non-1E Service Water, Revision 04, 1/28/83.
4. SWEC Elementary Diagram ESK-11SWP10, SSW Aux. Control, Revision 3, 5/5/82.

D6.5-2 (Deficiency) Standby Diesel Generator Initiation on LOCA

BACKGROUND

Upon receipt of a Loss-Of-Coolant Accident signal from the NSSS protection system (i.e. Residual Heat Removal System relay E12-K110A and B), the standby diesel generators and other portions of the engineered safety feature equipment are required to be automatically initiated. This requirement is stated in 10CFR50 Appendix A General Design Criterion 20 and in Chapter 8.3 of the FSAR.

DESCRIPTION

The nuclear steam supply system generated LOCA signal involving relays K110A and B is not dependent upon plant AC power sources. However, the Stone & Webster Engineered Safety Feature actuation logic for the standby diesel generators and certain Engineered Safety Feature loads includes electrical relays that must energize upon receipt of this LOCA signal and that were dependent upon either off-site or on-site AC power. Consequently, if a LOCA situation is postulated shortly after the loss of off-site power, initiation of the two main diesel generators would not be accomplished by this LOCA signal, but rather only by the loss of off-site power signal.

BASIS

FSAR Section 8.3 provides a commitment that the standby diesel generators are started by an undervoltage signal, a LOCA signal, or from a manual action signal. Stone and Webster sketches identified 1SCM*PNL01A and B as the 120 volt AC regulated power sources for the LOCA signal relays which are supplied from 4KV essential busses 1ENS*SWG1A and B respectively. Thus, whenever the off-site AC power is lost, the LOCA output relays are rendered inoperable until AC power is restored to the 4KV busses. This is a violation of the FSAR commitment, IEEE Std. 279-1971 sections 4.1 and 4.8, and 10CFR50 Appendix A General Design Criteria 20.

IMPACT ON DESIGN

The design was not acceptable and required revision. A power source design change was accomplished during the inspection. Nevertheless, the team was concerned that this type of design error was not detected by one or more of the design review and design verification processes used by these organizations. For example, it was determined that the interruptible power source selection was made by Stone & Webster during the original Balance-of-Plant design effort in early 1974, and had not been detected or corrected as a result of:

(a) subsequent revisions to the affected design documents;

(b) independent design reviews conducted by Stone & Webster and a Low Pressure Core Spray System technical design audit conducted in 1983 by Gulf States Utilities;

(c) individual system design freeze reviews involving General Electric, Gulf States Utilities, and Stone & Webster in the 1978-1980 period;

(d) failure mode and effects analyses performed by Stone & Webster during the past several years, and

D6.5-2 (Deficiency) continued.

(e) subcontract work performed by General Electric involving design and fabrication of the safety-related Balance-of-Plant Power Generation Control Complex panels.

The team believes that the Stone & Webster design verification process, particularly on a system basis, should be reassessed to assure detection of this type of error.

The team was unable to identify or locate any additional information that would provide a basis for Stone & Webster acceptance of this particular power source design selection. Currently responsible Stone & Webster personnel have provided information on the internal reviews performed when power supply choices are made; however, based on the technical significance of this one particular instance, the team is concerned that reassessment of past design decisions may not be a viable part of Stone & Webster's internal design review process. Furthermore, the team believes that the absence of explicit design bases and criteria in the Stone & Webster design documentation was a contributing factor to this design error.

EXTENT

The extent of this type of problem is not known, but is believed to be rather limited.

POST CUTOFF WORK

A design change package has been processed by Stone & Webster to correct this power source problem.

REFERENCES

1. FSAR, Section 8.3, page 8.3-13, 1/84.
2. S&W Elementary Diagram, ESK-7ISC01, Div I BOP LOCA Initiation, Revision 6, 9/22/83.
3. S&W Elementary Diagram, ESK 7ISC03, Div II BOP LOCA Initiation, Revision 6, 1/13/83.

D6.6-1 (Deficiency) Instrument Setpoint Documentation Inconsistencies

BACKGROUND

Safety-related instrument setpoint values, such as those used to automatically initiate the Standby Service Water System, are depicted on loop diagrams, logic diagrams, sensor purchase specification technical data sheets, and in setpoint calculations.

DESCRIPTION

A Stone & Webster internal memorandum applied to River Bend via Technical Guideline 81/1-0 explicitly states that the only source document for setpoint values is the instrument setpoint calculation. A very recent note reiterated this point. During the inspection, it became evident that the applicability of this Technical Guideline, particularly where stated setpoint values are inconsistent, was neither well known nor widely understood. Document users are not cautioned that listed setpoint values are unofficial and subject to change based on final setpoint calculations.

In at least two instances, the setpoint value for a particular sensor was stated inconsistently from one document to another. For example, the setpoint calculation for low pressure measurements in the Reactor Plant Component Cooling Water Header lists the setpoint as 56 $\pm$ 1 psig decreasing; however, the logic diagram shows the setpoint as less than or equal to 90 psig. In another example, the calculation of Service Water Normal Supply Header A and B low pressure lists the setpoint as 76 psig plus or minus 3 psig decreasing, but the loop diagram shows the setpoint as 75 psig decreasing whereas the logic diagram presents the setpoint as less than or equal to 75 psig.

BASIS

Various design documents are inconsistent in depicting setpoint values.

IMPACT ON DESIGN

For these situations, hardware changes or reanalysis are not expected. Documentation changes to assure consistency in setpoint values are required. Improved performance in the Engineering Assurance implementation of internal design review and design verification activities involving engineering documents is also needed.

EXTENT

This problem appears to be systematic throughout the Stone & Webster documentation for River Bend.

D6.6-1 (Deficiency) continued:

REFERENCES

1. Stone & Webster Eng. Div. Memorandum, CSDM(CHOC)-81/3-0, Calculation of Setpoints for Nuclear Power Plant Category I Instruments, Revision 0, 2/10/81.
2. Stone & Webster Internal Note, "Setpoints," J.C. Bisti to distribution, 3/3/[illegible].
3. Stone & Webster Setpoint Calculation, 12210-IA-SWP*1, Low Pressure Trip Units 1SWP*ES21A,C,E,G and 1SWP*ES21B,D,F,H for Service Water Normal Supply Header A & B, Revision 0, 4/12/83 and Revision 1, 9/27/83.
4. Stone & Webster Setpoint Calculation, 12210-IA-CCP*1, Low Pressure Trip Units 1CCP*ES1A,C,E,G and 1CCP*ES1B,D,F,H for Component Cooling Water Header, Revision 0, 8/2/83.
5. Stone & Webster Technical Data Sheet, 247.461 pages 3-4.20, 3-4.21, 3-4.1, and 3.4-2, Revision 1, 8/24/83 and 8/25/83.
6. Stone & Webster Loop Diagram, 1SWP*21, Revision 5, 11/4/81.
7. Stone & Webster Logic Diagram, LSK-22-8.1A, Revision 1, 11/4/82.
8. Stone & Webster Logic Diagram, LSK-9-1C, Revision 7, 5/7/81.

D6.6-2 (Deficiency) Instrument Setpoint Calculation Assumptions

BACKGROUND

Assumptions used in safety-related instrument setpoint calculations, such as those used to automatically initiate the Standby Service Water System, are required to be documented. Furthermore, revisions to established setpoint calculations should employ currently valid input data obtained from other Stone & Webster groups.

DESCRIPTION

The Standby Service Water System low pressure transmitter calculation did not contain stated assumptions prior to May 4, 1984. In addition, revision 1 of this calculation did not modify the radiation effect on accuracy even though a new basis had been provided approximately two months earlier.

BASIS

Setpoint calculation 12210-IA-SWP*1 did not provide assumptions as required by Stone & Webster procedures, and also did not reflect a new radiation basis pertinent to this calculation. Calculation also 12210-IA-CCP*1 did not reflect the new radiation basis.

IMPACT ON DESIGN

No hardware impact is anticipated. Reanalysis of one setpoint calculation was accomplished on May 4, 1984. Revision of the Component Cooling Water setpoint calculation is expected.

EXTENT

The lack of stated assumptions appears to be a random problem based on two calculations examined. However, the lack of use of new input data in these calculations appears to be systematic since both calculations exhibited this characteristic.

POST CUTOFF WORK

One setpoint calculation has been revised.

REFERENCES

1. Stone & Webster Eng. Div. Memorandum, CSDM(CHOC)-81/3-0, Calculation of Setpoints for Nuclear Power Plant Category I Instruments, Revision 0, 2/10/81.
2. Stone & Webster Setpoint Calculation, 12210-IA-SWP*1, Low Pressure Trip Units 1SWP*ES21A,C,E,G and 1SWP*ES21B,D,F,H for Service Water Normal Supply Header A & B. Revision 0, 4/12/83, Revision 1, 9/27/83, and Revision 2, 5/4/84.
3. Stone & Webster IOM, T. Tonden to G. Bell, "12210 Category I Setpoint Calc's Pressure Transmitter Accuracy in Radiation Fields," 7/7/83.
4. Stone & Webster Setpoint Calculation, 12210-IA-CCP*1, Low Pressure Trip Units 1CCP*ES1A,C,E,G and 1CCP*ES1B,D,F,H for Component Cooling Water Header, Revision 0, 8/2/83.

D6.6-3 (Deficiency) Instrument Procurement Specification Inconsistencies

BACKGROUND

Procurement specifications for safety-related instruments are generally issued prior to the determination of required instrument setpoint values; consequently, routine confirmation of the continuing adequacy of procured instruments must be made as various portions of the plant design are completed and verified.

DESCRIPTION

An instrument technical data sheet for the Cooling Tower Pump House Pump Room temperature measurement, 1HVY*RTD24A and B, lists a maximum temperature design requirement of 109^{o} F. This value does not conform with the alarm setpoint value of 115^{o} F stated on the Stone & Webster loop diagram or the logic diagram. Setpoint calculations, which establish the official setpoint values, had not been accomplished at the time of the inspection.

BASIS

Stone & Webster Technical Guideline 81/1-0 applied to River Bend states that the source document for instrument setpoint values is the setpoint calculation. Since this latter document does not yet exist, inconsistencies in stated setpoint values should be detected and corrected by internal or external document review and design verification processes. In this particular instance, the range inconsistency was not detected and corrected.

IMPACT ON DESIGN

This item had no impact on design or analysis since RTD sensors are generally capable of service up to 600^{o} F as demonstrated in the Pyco equipment qualification report for another project. A minor documentation change to the technical data sheet will be needed.

EXTENT

This inconsistency appears to be random.

REFERENCES

1. Stone & Webster Internal Note, "Setpoints," J.C. Bisti to distribution, 3/8/84.
2. Stone & Webster Setpoint Calculation, 12210-IA-(later), 1HVY*RTD24A and B Resistance Temperature Detector, (to be issued in 1984).
3. Stone & Webster Technical Data Sheet, 247.461 page 3-11-12, Revision 2, 7/12/83.
4. Stone & Webster Loop Diagram, 1HVY*24, Revision 3, 11/24/82.
5. Stone & Webster Logic Diagram, LSK-22-8.1A, Revision 1, 11/4/82.

D6.9-1 (Deficiency) Balance of Plant Accident Monitoring Variables

BACKGROUND

Regulatory Guide 1.97 requires that certain balance of plant Type D Engineered Safety Feature Supporting System status variables be monitored prior to and during an accident using instrumentation meeting RG 1.97 Qualification Category 2 requirements. These requirements are less stringent than those imposed on Class 1E safety-related equipment (i.e. River Bend Quality Assurance Category I), but are more stringent than those imposed on commercial grade non-safety-related equipment (River Bend QA Category II). Under design basis accident conditions, the Stone and Webster design philosophy for Engineered Safety Feature system cooling uses the Normal Service Water System (Quality Assurance Category II) until such time as its failure initiates automatic operation of the Standby Service Water System (Quality Assurance Category I).

DESCRIPTION

Stone & Webster has prepared a list of selected variables and instruments to implement Quality Assurance Category II monitoring in the Normal Service Water System and Quality Assurance Category I monitoring in the Standby Service Water System; however, these choices are not depicted in design documents. The list did not include measurement of battery current as required by RG 1.97. In addition, the Normal Service Water System accident monitoring instrumentation for Engineered Safety Feature cooling flow and temperature is implemented with Qualification Category 3 equipment (Quality Assurance Category II) rather than Qualification Category 2 equipment. The loop diagram indicated that the flow measurement is used for pump capacity checks only, and does not indicate an accident monitoring function.

BASIS

Even though the selected balance of plant instruments were reviewed by Gulf States Utilities in mid-1982, formal engineering design documentation does not yet exist to specifically identify these measurements. FSAR Table 7.5-2 has also not been updated to delineate specific equipment mark numbers as requested by Gulf States Utilities. Furthermore, the choice of commercial grade instrumentation for Normal Service Water flow and temperature measurements does not meet the graded Quality Assurance requirements imposed by RG 1.97.

The pneumatic supply pressure instrumentation is designated as Quality Assurance Category I as are the voltage and current monitors for emergency AC and DC power sources.

For this accident monitoring instrumentation issue, the Stone & Webster design process has not been controlled. The systematic omission of accident monitoring instrumentation designations on Stone & Webster design documents may impact some power source selections, cable routings, and environmental qualification activities.

IMPACT ON DESIGN

Upgrading of Normal Service Water instrumentation should be accomplished to conform with RG 1.97 requirements for these status indications. Explicit documentation of the selected instrumentation on design documents and in the FSAR should also be accomplished.

D6.9-1 (Deficiency) continued.

EXTENT

These identified problems appear systematic in Stone & Webster balance of plant Quality Assurance Category II non-safety-related systems.

REFERENCES

1. ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants, paragraph 5.1.4, Documentation.
2. Gulf States Utilities Company letter, RBG-13126, August 2, 1982.
3. Stone & Webster Loop Diagram, 1SWP-117, Normal Service Water Pumps Combined Discharge Flow, Revision 2, 9/26/83.

D6.11-1 (Deficiency) EQ Analysis Error in Normal Environment Enveloping

BACKGROUND

Safety-related pressure transmitters used to initiate operation of the Standby Service Water System are located in areas subject to harsh environmental conditions such as pipe tunnel 3 and at elevation 70 of the Auxiliary Building. These transmitters, subject to environmental qualification (EQ) requirements, have been accepted by Stone & Webster based on a comparison review of vendor test reports relative to actual River Bend requirements.

DESCRIPTION

Stone & Webster used an enveloping method to include the 1SWP*PT21A through H transmitters (PT-3 location) and the 1CCP*PT1A through H transmitters (AB-070-8 location) with other equipment at a chosen CT-2 location because the CT-2 area had a worst case environment profile under accident conditions. This comparison review then used the normal environment for the CT-2 location rather than a worst case normal environment. The selected CT-2 area does not envelope the normal temperature, humidity or pressure values seen by these particular safety related transmitters. Rather than use the normal average 106° F temperature, a normal average temperature value of 70° F was used in the evaluation of these transmitters. Similarly, rather than a 90% maximum relative humidity, a maximum value of 50% was used.

IMPACT ON DESIGN

The normal environment qualification evaluation for these transmitters used an incorrect basis. Re-analysis of the transmitter qualified life is required. The qualified life of these transmitters will probably be reduced from these effects.

EXTENT

It is unknown whether other similar errors have occurred in the enveloping process for environmental qualification.

REFERENCES

1. Stone & Webster Environmental Design Criteria, 215.150, Revision 1, 4/19/83 and Revision 2 (unissued), 2/1/84.
2. Stone & Webster Qualification Review Summary for Specification 247.481, 3/10/84.
3. Gulf States Utilities Company Letter, RBG-17024, Mr. J.E. Booker to Mr. J.A. Kirkebo, Use of 215.150 Environmental Design Criteria Revision 2, 2/14/84.

D6.13-1 (Deficiency) Instrument Change Revision Notice Calculation Reference

BACKGROUND

Due to the particular instrument rack location selected by Stone & Webster, three Residual Heat Removal System transmitters and indicators were moved from a General Electric supplied rack to a local mount designed by Stone & Webster. The local mounting was seismically analyzed by Stone & Webster's Site Engineering Group.

DESCRIPTION

Stone & Webster Instrument Change Revision Notice 316-GE-01 referenced an incorrect calculation in the technical justification for acceptance of the seismic analysis. The calculation referenced was for a cantilevered structure attached to the containment liner rather than a local rack structure welded to embedments at elevation 70 of the Auxiliary Building.

BASIS

An incorrect reference to a calculation was listed in the approval section of an instrument change revision notice; consequently, the technical basis for acceptance of the design change was in error.

IMPACT ON DESIGN

No hardware or analysis impacts are anticipated. The team examined the seismic calculation, and determined that it was satisfactory in all other respects.

EXTENT

This appears to be a random documentation error.

REFERENCES

1. Stone & Webster Instrument Change Revision Notice, 316-GE-01, 2/16/84.
2. Stone & Webster Calculation, 12210-NP(S)-Z-GE-1566, Revision 1.
3. Stone & Webster Calculation, 12210-NP(S)-Z-GE-1570, Revision 1, 12/8/83.
4. Stone & Webster Calculation, 12210-NP(S)-Z-GE-1570, Revision 2, 3/9/84.
5. General Electric Instruments 1E12*PTN055C, PTN056C, and PIR008C.

D.A.1-1 (Deficiency) Compliance of Calculation with the Requirement for an Auditable Review/Check of Results

BACKGROUND

The Power Division uses a simple procedure of signing or initialing a cover sheet without identifying the method used to review/check calculations.

DESCRIPTION

Stone & Webster procedure EAP 5.3 provides for the check/review of calculations. A variety of acceptable approaches are mentioned. Four calculations (References 2, 3, 4 and 5) were reviewed for compliance with ANSI N45.2.11 and it was found that none of them provided an auditable record of how the calculation was reviewed/checked.

BASIS

ANSI N45.2.11-1974 which is endorsed by Regulatory Guide 1.64 states the following: "4.1... The design activities shall be documented in sufficient detail to permit verification and auditing..." Calculations and their review/check are included as a "design activity" by the definition in paragraph 4.1 which is: "The design activities may be prescribed in job specifications, work instructions, planning sheets, procedure manuals, test procedures, or any other type of written form, which provides adequate control and permits reviewing, checking or verifying the results of the activity..."

IMPACT ON DESIGN:

The design process and its success in producing an error free design requires an effective review/check of the calculational effort. In order to assure that the process is working properly an auditable trail must be established. In this case the review/check activity is not auditable.

EXTENT

Since the method of review/check was not stated for each of the calculations selected for review by the team, we concluded that the problem is systematic. It was determined that the Engineering Mechanics Division does comply with this requirement.

REFERENCES

1. S&W Procedure No. EAP-5.3 with Attachment 3.0.
2. S&W Calculation PN-048, Subsystem Fill Pump Sizing, Rev. 1.
3. S&W Calculation PN-307, Piping Minimum Wall Thickness, Rev. 0.
4. S&W Calculation PN-268, RHR Pump NPSH And Total Head, Rev. 2.
5. S&W Calculation PN-263, Sizing Orifice On RHR, LPCS, HPCS And RCIC Subsystem Fill Pumps Recirculation Lines, Rev. 0.

D.A.1-2 (Deficiency) Arithmetic and Reference Errors in Calculation

BACKGROUND

Four calculations (References 1, 2, 3 and 4) were selected for a review of arithmetic accuracy and correct transfer of information from page to page. The calculations selected were typical for mechanical systems with results that in the team's opinion should be reviewed by a number-by-number check due to their system specificity.

DESCRIPTION

A number-by-number check was performed on a selected number of arithmetic activities in each of four calculations. Two of the four calculations (References 1 and 4) had errors in page references and/or transfer of information from page to page. Three of the four calculations (References 1, 2 and 4) had arithmetic errors. These errors were simple addition, the most significant of which is in PN-048 where a set of four numbers was added to give a total of 34 feet 4 inches. The correct sum is 24 feet 4 inches.

BASIS

Arithmetic errors and page reference mistakes were made in the calculations.

IMPACT ON DESIGN

Design decisions are based on calculational results and if these results are in error, the equipment selected might not fulfill requirements. The specific error found in PN-048 would result in a pump that would not have sufficient head for the design flow. The other errors were minor and would not have an impact on design.

EXTENT

The fact that three out of four calculations selected for analysis had simple arithmetic errors and that two out of the four had errors in page references and/or page to page transfer of information indicates that the review/check process for calculations is not being properly implemented. This appears to be systematic for mechanical calculations.

REFERENCES

1. S&W Calculation PN-048, Subsystem Fill Pump Sizing, Rev. 1.
2. S&W Calculation PN-263, Sizing Orifice On RHR, LPCS, HPCS And RCIC Subsystem Fill Pumps Recirculation Lines, Rev. 0.
3. S&W Calculation PN-307, Piping Minimum Wall Thickness, Rev. 0.
4. S&W Calculation PN-268, RHR Pump NPSH and Total Head, Rev. 2.
5. S&W Procedure EAP-5.3 with Attachment 3.0.

U3.4-1 (Unresolved Item) Added Mass for Trapeze Hangers (Nuclear Steam Supply System)

BACKGROUND

Trapeze hangers provide vertical support for a horizontal run of pipe, but no lateral support. In fact, for a seismic event occuring in the horizontal plane, the pipe drives the support mass. Depending upon the pipe size and the support configuration, added mass may have to be lumped at the pipe node located at the support point.

DESCRIPTION

The main steam piping system, which is part of the General Electric Safety Relief Valves scope of work, contains several trapeze hangers that were designed by the Bergen-Peterson Pipe Support Corporation for General Electric. See for example the Reference 1 drawing. Added mass was not lumped at the pipe nodes located at these support points.

POTENTIAL BASIS

The team was informed that no General Electric procedure exists to evaluate the need for added mass at trapeze hanger support points. General Electric stated that although this point had been considered several years ago, the effects would be negligible.

IMPACT ON DESIGN

The impact on design is probably minor.

EXTENT

This is an unresolved item for any stress analyzed piping subsystem in the General Electric Nuclear Steam Supply System scope of work which contains trapeze hangers.

REFERENCES

1. S&W EMDM 81-04, Effect of the Spring Hanger Assembly Weight on Piping Systems.

U4.16-1 (Unresolved Item) Decoupling of Control Rod Drive Piping and Supports

BACKGROUND

Each control rod drive piping assembly consists of approximately seventy two pipes ranging in diameter from 1 to 3 inches. The piping aggregate is supported by a complex structural framework consisting of rectangular structural tubing. The pipes are connected to their supports by a system of clamps which prevent movement in both vertical and horizontal directions.

DESCRIPTION

The mathematical model for piping supports was decoupled from the piping and the masses of piping were not included in the model used to determine the dynamic response of the supports.

BASIS

The position stated in Section 3.7.2.1.1.2A of the FSAR (Decoupling Criteria) have not been complied with.

IMPACT ON DESIGN

Inclusion of the masses of a large number of small diameter pipes could affect the fundamental frequency of the supports.

EXTENT

This deficiency appears to be systematic.

REFERENCES

1. NUREG Standard Review Plan Section 3.7.2, Seismic System Analysis, Rev. 1, 7/19/81.
2. RCI Task File, RCI Document SA-5259, Rev. 0, 4/20/81.
3. River Bend Station Final Safety Analysis Report Amendment 11, January 1984.

U5.5-1 (Unresolved Item) As-built Conduit Installation Drawings

BACKGROUND

Installed conduit 1CK500BB, located inside containment, was routed to tray ITK502B whereas the Conduit Installation and Tray Identification drawings (References 2 and 7) and the EC-38 report required the conduit to be routed to tray ITK500B. The team determined that the conduit routing had been changed in the field to a preferred field-run route. This change was documented on "marked-up" raceways and cable pull tickets. We learned, however, that the Conduit Installation Drawing EE-460AF-4 will not be revised to show the new conduit route and tie point for 1CK500BB.

DESCRIPTION

We observed that the actual path for the installed conduit 1CK500BB3 deviated to a large extent from the path representation (undimensioned) shown on Conduit Installation Drawings EE-460X-5 and EE-406AF-4. The team became concerned that the routing on the Conduit Installation Drawings for safety-related conduit did not closely represent the actual installed conduit location, therefore a reasonable representation of the actual conduit path was not presented on Stone & Webster design drawings. The lack of design drawings which adequately show the approximate location of installed safety-related conduit would appear to inhibit evaluation of accident effects associated with programs such as High Energy Line Break Analyses.

During subsequent discussions, the team learned that Stone & Webster is not providing as-built (as-installed) conduit installation drawings. The Stone & Webster position on this issue is that the design drawings show undimensioned, suggested routing for conduit, and all conduit will be essentially field run. In support of this position Stone & Webster provided the following information. The Electrical Installation Specification 248.000, Section 3.1.1 states that conduit shown on drawings are only approximate to scale and that conduit must be supported and installed by the field in accordance with the drawings. In addition, note 15 of drawing EE-450AJ-2 states the location of conduit shown on plan and detail drawings is recommended, the field may adjust position of conduit to avoid interferences and facilitate installation. Note 11 of drawing EE-460A-3 also states that conduit is shown diagrammatic and should be field run and supported in accordance with the seismic support details. Stone & Webster provides to the field a variety of seismic category I support details (Reference 5), however, the location of the supports and the routing of the conduit between tie points is determined in the field.

BASIS

In summary, conduit is field routed; the design drawings are diagrammatic only. The field and site Quality Control are responsible for ensuring that conduit installation meets separation criteria and seismic support criteria. The team also learned that the Stone & Webster High Energy Line Break Procedure (PMM-152) states that additional targets may require evaluation as a result of a field "walkdown" which will be conducted at the time of building turnover. Stone & Webster intends to identify conduit, instrumentation, and instrument tubing located within High Energy Line Break areas by this walkdown.

We concluded that the Stone & Webster field walkdown in accordance with PMM-152 to determine conduit vulnerable to High Energy Line Break events does not totally

U5.5-1 (Unresolved Item) continued.

satisfy our concern regarding field run conduit. (Section 2 of this inspection report discusses the High Energy Line Break evaluation in detail).

Our experience has shown that on other nuclear plants a significant number of raceway installations did not conform to Final Safety Analysis Report commitments for independence of class 1E equipment and circuits. Deficiencies identified by the NRC Construction Assessment Team on these plants indicate that quality control inspection of construction activities were not totally effective in identifying raceway separation deficiences.

Additional information is required to determine whether the approach used at River Bend is acceptable. The team was unsure whether or not conduit installation drawings were scheduled for voiding following construction of River Bend to prevent inadvertent use of inaccurate information. The team was also unsure what measures were to be developed to ensure field walk downs are performed any time information is required concerning conduit location details. The team felt uncomfortable with the Stone & Webster design. A preferred approach would be to establish tolerance criteria for field run deviations, revising the installation drawings when these criteria are exceeded.

IMPACT ON DESIGN

The team remains concerned that a reasonable representation of the actual path for safety-related conduit is not presented on Stone & Webster's design drawings. There can be large deviations between the installed conduit path and the design path between tie points. We assume that the Conduit Installation Drawings will not be voided after the plant construction phase, therefore these drawings will remain as design documents for the life of the plant. Since the design drawings do not reasonably represent the installed conduit routing, a field walkdown will be required each time the location and routing of safety-related conduit is under study.

EXTENT

River Bend has not been constructed with drawings of safety-related conduit routing which show a reasonable representation of the actual conduit path. This approach was also apparently used to route safety-related instrument air tubing and other pneumatic supplies.

REFERENCES

1. SWEC Seismic Conduit Installation, 114' Reactor Building, 12210-EE-460X-5, Rev. 5, 1-25-84.
2. SWEC Seismic Conduit Installation, 141' Reactor Building, 12210-EE-460AF-4, Rev. 4, 2-13-84.
3. SWEC Procedure PMM-152, Interoffice Memorandum, High Energy Line Break Evaluation Procedure, Rev. 2, 4-6-84.
4. SWEC Specification for Electrical Installation, 248.000, Rev. 7, 12-30-83.
5. SWEC Seismic Conduit Support Standard Details, Reactor Building, 12210-EE-450AJ-2 thru 450AN, Rev. 2, 2-20-83.
6. SWEC Seismic Conduit Installation, Reactor Building, 12210-EE-460A-3, Rev. 3, 11-14-83.

U5.5-1 (Unresolved Item) continued.

7. SWEC Cable Tray Identification, Reactor Building, 12210-EE-34EB-4, Rev. 4, 3-11-83.
8. SWEC EC-38 Report, Cable and Raceway Installation Status, Pages 4573 and 4574, Issue 76, 4-19-84.

U5.12-1 (Unresolved Item) Standby Diesel Generator Lubricating Oil Circulating Pump Motor Qualification

BACKGROUND

Standby Diesel Generators supplied by Delaval have nonqualified motors for driving ASME III Code 3, seismic Category 1 lubricating oil circulating pumps (References 1, 2 and 8). These motors are supplied from a non-class 1E AC source (Reference 9). The High Pressure Core Spray diesel generator supplied by General Electric has a qualified motor for its lubricating oil circulating pump and is supplied from a Class 1E AC source (Reference 10).

DESCRIPTION

Delaval, in their letters (References 3 and 4), maintains that motor qualification is not a contractual obligation for this project and that the lubricating oil recirculating pumps are not critical to the diesel generator or mandatory for its safety function. No failure modes and effects analysis was identified to substantiate acceptance of Delaval's approach. The letters also state that if Gulf States so desires, Delaval will complete the qualification program for the lubricating oil recirculating pump motors for additional cost. The team reviewed meeting notes between Stone & Webster and Delaval (Reference 5) and noted that Stone & Webster did not agree with Delaval's position. Stone & Webster documented this as an unresolved item. Delaval's letters (References 3 and 4) do not provide sufficient basis for Delaval's position that failure of these pumps does not affect the safety function of the diesel generator. Delaval did not provide test results or analysis to substantiate their position. Delaval also failed to substantiate that starting the diesel without pre-lubrication of the engine and turbocharger would not affect the diesel generator's qualified life. NRC NUREG/CR-0660 addresses enhancement of emergency diesel generator reliability. This report recommends that the engine prelube pump be started by the same signal which initiates the cranking of the engine, i.e., the emergency start signal. Also the Standard Review Plan recommends maintaining proper lube oil temperature to improve first-start reliability. The team noted that neither the lube oil heater supply nor the lubricating oil recirculating pump motor supply is a class 1E supply. The team thus concluded that qualification of the prelube pump motor and the lube oil heater power supply should be further evaluated to improve the reliability of the diesel generators.

POTENTIAL BASIS

Since the motor and its power supply are not qualified they may fail to provide motive power to the pump, thereby making the pump inoperative. Prelubrication of the engine and turbocharger may be necessary for the diesel generator to successfully start and come to the speed at which the engine-driven lubricating oil pump starts functioning.

POTENTIAL IMPACT ON DESIGN

If analysis of motor function results in the requirement to make the motor Class 1E, the motor will have to be qualified and its power supply will also have to be Class 1E.

U5.12-1 (Unresolved Item) continued.

REFERENCES

1. FSAR, Classification of Components, Section 3.2 Table 3.2-1, January 1984.
2. Stone & Webster Procedure PTP-34.1.1-0, February 12, 1982.
3. Delaval Letter, (J. A. Siegel) to Stone & Webster (W. S. Raughly), June 13, 1983.
4. Delaval Letter, (J. A. Siegel) to SWEC (J. C. Weller), April 26, 1984.
5. Stone & Webster/Delaval Meeting Notes, of 4/7 and 4/8, 1981, Letter #RBV-0-15,322.
6. NRC NUREG, Enhancement of Onsite Emergency Diesel Generator reliability, NUREG/CR-0660, February 1979.
7. Standard Review Plan Section 9.5.7.
8. Delaval drawing, Lube Oil Piping Schematic TDI DWG No. 09-820-74039, January 9, 1980.
9. Stone & Webster Drawing, AC Single Line Diagram, 12210-EE-1XA-2, Rev. 2, December 20, 1983.
10. Stone & Webster Drawing, AC Single Line Diagram, IE22-EE-ISA-1, Rev. 1.

U6.7-1 (Unresolved Item) Periodic Test of the Standby Service Water System

BACKGROUND

Safety Guide 22 requires that the actuated equipment in safety-related systems be tested during periodic tests of the protection system while the reactor is in operation. Other permissible testing options are also outlined whenever on-line testing of these actuated devices may cause damage to plant equipment or disrupt reactor operation. The River Bend project position of compliance with Safety Guide 22 is stated in the FSAR.

The River Bend design for Engineered Safety Feature cooling water is based on use of the non-safety-related Normal Service Water and Reactor Plant Component Cooling Water Systems in a postulated accident situation for as long as off-site power remains available. This situation is however, complicated by several pertinent factors; namely:

(a) Concurrent operation of both trains of Standby Service Water cannot be demonstrated during normal plant operation because of capacity limitations relative to non-safety-related cooling loads needed for full power operation;

(b) Four possible operating modes of these cooling systems appear to be feasible as listed below:

	Plant Status	Division 1 Cooling	Division 2 Cooling
Mode 1	Normal or Emergency	Normal SW and RPCCW	Normal SW and RPCCW
Mode 2	Normal or Emergency	Normal SW and RPCCW	Standby SW "B"
Mode 3	Normal or Emergency	Standby SW "A"	Normal SW and RPCCW
Mode 4	Emergency	Standby SW "A"	Standby SW "B"

(c) The Gulf States Utilities pre-operational test procedure of the Standby Service Water System, currently under development, appears to address only Mode 4 of these possible combinations;

(d) During the inspection, Gulf States personnel indicated that no pre-operational or startup test procedure had been scoped to include tests for Modes 2 and 3 listed in the preceding table, and

(e) In subsequent telephone conversations with Stone & Webster personnel, it was stated that surveillance tests performed at refueling outages would include Modes 2 and 3. Since these procedures are still in development, this intent needs to be confirmed at a later date.

DESCRIPTION

Engineered Safety Feature instrumentation and control compliance with Safety Guide 22 is provided with no exception taken for other safety-related supporting systems, such as Standby Service Water. However, a system level exception is taken in that the Standby Service Water System is only capable of being tested on a system basis during refueling outages. Consequently, the extent of compliance with Safety Guide 22 was investigated for periodic test conditions involving Standby Service Water pump operation with certain valves rendered inoperable and then with Standby Service Water valve operation when the pumps were rendered inoperable.

Stone & Webster personnel indicated that the Standby Service Water System pumps can be started during full power operation and provide fluid flow in parallel with the Normal Service Water System pumps. Pump discharge pressure and partial fluid flow in the Standby Service Water System can be monitored using Class 1E instrumentation. Only two valves to the Reactor Plant Component Cooling Water loop need to be rendered inoperative during the Standby Service Water pump test so that maintenance of normal water quality can be assured for the Spent Fuel Cooling heat exchanger loop. Hence, the only aspects of Standby Service Water System operation that need to be confirmed during refueling outages are the achievement of full system flow under accident conditions and operability of the fluid connection into the Spent Fuel Pool cooling loop. Based on these discussions, the team formed three conclusions; namely, that:

(1) potential conflicts with the aforementioned Safety Guide 22 commitment for the Standby Service Water System appear to be technically insignificant;

(2) improved FSAR descriptions and analyses to demonstrate River Bend compliance with Safety Guide 22 should be provided to include an identification of those actuated components in Engineered Safety Feature supporting systems that cannot be tested during full reactor power operation, and

(3) a commitment should be made to periodically operate one Standby Service Water loop concurrently with the Normal Service Water System during plant operation to better demonstrate the on-line availability of the Standby Service Water System. A preliminary review indicates that this capability exists in the present design of the Standby Service Water System.

BASIS

During the inspection, two licensing commitments concerning Safety Guide 22 appeared to be in conflict with one another. After detailed review with Stone & Webster personnel of the periodic test provisions provided for the Standby Service Water System, it was concluded that the FSAR does not adequately describe concurrent operation of Standby Service Water and Normal Service Water pumps, nor does it identify specific Standby Service Water System components that cannot be tested during power operation. The FSAR should be revised to include descriptions and analyses of Safety Guide 22 conformances and non-conformances for each balance-of-plant safety-related system. Where systems can be readily tested during power operation, appropriate commitments relative to Safety Guide 22 should be made.

IMPACT ON DESIGN

No hardware or analysis impact is expected. FSAR changes are anticipated; however, other documentation changes should be negligible as plant test procedures related to the Standby Service Water System are currently in a draft form.

U6.7-1 (Unresolved Item) continued.

EXTENT

The extent is unknown since only a limited sample of safety-related Balance-of-Plant systems and pre-operational test procedures have been examined during this inspection.

REFERENCES

1. Safety Guide 22, Periodic Testing of Protection System Actuation Functions, 2/17/72.
2. River Bend FSAR, Section 1.8, page 26 of 193.
3. River Bend FSAR, Section 7.3, pages 7.3-44 and 7.3-45.
4. River Bend FSAR, Section 9.2.7, page 9.2-44.

O2.3-1 (Observation) RHR Discharge Line Abnormal Condition Alarm

BACKGROUND

Each residual heat removal Loop (A, B, and C) discharge line must be maintained filled with water in order to prevent waterhammer upon pump start. Also, each discharge line is connected to a high pressure source (reactor pressure vessel) through valves. The line design pressure is 500 psia which is substantially lower than the pressure in the reactor pressure vessel. Therefore, it is important to protect these lines against; 1) not being filled and 2) overpressurization.

DESCRIPTION

The not-filled condition is identified by a low pressure switch (N654) and the excessive condition is identified by a high pressure switch (N653). Both switches operate a single relay (K41). This relay in turn lights a single alarm window and is annunicated. The window for each loop is designated "Residual Heat Removal Discharge Line Pressure Abnormal." The color of the window is white. There is no other identification as to which condition exists. As indicated above both conditions are serious and require immediate attention. The control room two colors (red and white) are used to provide visual indication of conditions requiring operator attention. Red requires immediate attention and white is of secondary importance.

BASIS

The use of a white window and combining the two abnormal conditions without a way to distinguish which condition exists is not commensurate with the seriousness of the consequences on the emergency core cooling system.

RECOMMENDATION

Gulf States should investigate having each pressure switch (High and Low) operate a separate relay which would initiate either a separate or combined red windowed alarm. Alternatively, control room indication of residual heat removal pump discharge pressure could be provided to allow differentiation of the alarm inputs.

REFERENCES

1. GE Elementary Drawings.

O2.3-2 (Observation) ECCS Subsystem Fill Pump Runout Protection

BACKGROUND

The emergency core cooling subsystems have fill pumps that are intended to keep the main pump discharge lines filled. There is a pressure tap (N053) in each main pump discharge. This pressure tap has a pressure switch connected to it to measure a low pressure situation. The low pressure situation is intended to indicate that the system is not full. The setpoint is to be set at elevation change plus 10 psi (Reference 1).

DESCRIPTION

For the residual heat removal pumps the setpoint would be 81.4 feet of head. With the setpoint at this value the fill pump would be in a high flow condition before the pressure switch would initiate the alarm. The reason is that the pressure being measured at the pressure tap is the sum of the suction pressure and the total discharge head of the fill pump. Since the suction pressure is basically the elevation of the suppression pool minus the elevation of the pressure tap which is relatively constant, the total discharge head of the pump must drop to very low value before the alarm setpoint is reached. In the case of the residual heat removal loops the total discharge head must drop to approximately 42.7 feet, which for the installed fill pump would indicate a flowrate that is beyond the certified pump performance curve (Reference 2).

BASIS

This is not an unresolved item or a deficiency because the alarm will annunciate a not full condition. The licensee will then have to take action required for the inoperability of the associated Residual Heat Removal system.

RECOMMENDATIONS

The team recommends that the available NPSH be evaluated under runout conditions, or alternatively the ability of the pumps to operate in runout should be verified.

REFERENCES

1. GE Alarm Function Diagrams.
2. Goulds Pumps Certified Pump Performance Curves A-28550-N767282, Rev. 0 and A-28379-N767283 Rev. 0.

O2.3-3 (Observation) Control of Design Basis Information

BACKGROUND

During the review of the NPSH available and total discharge head for the residual heat removal pumps operating in the low pressure coolant injection mode, it was discovered that inputs to the calculations were taken from a Stone & Webster project Technical Manual for vertical residual heat removal pumps. A review of this document revealed that it contained the vendor certified pump performance curves for Unit I and Unit 2.

DESCRIPTION

The Unit 1 and 2 curves were found to differ in the high and low flow regions. There is a group of certified head curves for each pump resulting from different required test conditions. Each curve has a separate number that is not relatable to the unit to which it applies. Therefore, these numbers cannot be used to distinguish between the Unit I and Unit 2 pumps. Thus, there is the definite potential for intermixing the two different pieces of the design data. Discussions with the Stone & Webster Power Division personnel revealed past problems with having design data for both units in the same document.

BASIS

This is not a deficiency because the team could not identify that the wrong unit's curve had actually been used.

RECOMMENDATIONS

The problem could be alleviated through Document Control by keeping the information separate or having a more explicit method of identifying to which Unit the information applies.

REFERENCES

1. S&W Technical Manual for Vertical RHR Pumps File No.22.431-000-031A.

O2.5-1 (Observation) Field Walkdown to Identify/Confirm HELB Targets

BACKGROUND

The identification of pipe whip and jet impingement targets within Engineering Mechanics Division are limited to structural targets, piping and pipe supports (generally large bore pipe), electrical cable trays, control instrumentation and equipment. Other potential targets such as electrical conduit, small bore piping and instrumentation tubing are to be identified by field walkdown prior to building turnover.

The inspection team conducted a field walkdown of the targets identified by Engineering Mechanics Division to confirm the accuracy of Engineering Mechanics Division's calculational technique and to gain confidence in Stone & Webster's ability to satisfactorily conduct a walkdown.

DESCRIPTION

During our inspection at Stone & Webster's Cherry Hill office, the inspection team was told that a field walkdown prior to building turnover was going to be used to confirm the target information generated by Engineering Mechanics Division's calculations and to identify other targets not amenable to calculation because of field construction routing. Stone & Webster personnel indicated that an as constructed pipe walkdown was going to be used and its results factored into restraint and jet impingement design. The team was informed that the main source of manpower for the walkdown would be Site Engineering Group personnel and that these walkdowns would be performed with the aid of a walkdown procedure and computer aids from Construction Systems Associates, Inc.'s three-dimensional model. When asked if a walkdown procedure existed the response was positive; however, when the team requested a copy the request was denied. It was explained that the procedure was not in a form such that it could be given to us and that it was only a working draft. The team asked if there were any other methods being used to identify potential targets which did not lend themselves to calculation. Stone & Webster personnel indicated that the plant model was a source of information in which small bore piping, conduit, and electrical junction boxes could be identified. They further indicated that in January 1984 two Engineering Mechanics Division engineers spent a week and a half looking at break locations on the model to identify small bore piping, conduit, and electrical junction boxes. It was also indicated that the model was under design control such that a relatively high confidence could be assigned to the targets identified.

Since the inspection team planned to conduct a field walkdown of the break location 30 of main steam line loop B, the team requested a listing of the targets identified by Engineering Mechanics Division's engineers on their inspection of the model and a listing of targets identified by the Construction Systems Associates, Inc. three-deminsional space model, including computer generated graphics of the jet cones and targets. Engineering Mechanics Division was able to identify 6 small bore pipes, one conduit, and one junction box within the influence of the jets. However the targets identified by the computer model were described as not being in a form usable for a field walkdown. Specifically the computer identified every subcomponent of the targets in the jet cone (e.g. nuts and bolts), and it did not stop at solid structures (e.g., structural walls).

During the team's field walkdown, they found that Engineering Mechanics Division's calculated structural targets did not reflect the as constructed condition (see Deficency D2.5-3). Although the drywell model stopped being maintained in a design control

O2.5-1 (Observation) continued.

fashion approximately 8 months previous to the walkdown, the six small bore pipes identified by Engineering Mechanics Division engineers were observed in the jet stream. However, the junction box and electrical conduit were not in the primary jet stream. Instead a different safety-related conduit (1CC5400A1) was in the jet stream.

During the field walkdown one of four senior mechanical engineers on-site was asked if he was aware of any field input into a high energy line break walkdown procedure. This individual indicated that he was aware that a walkdown was planned prior to building turnover; however, he was not aware of any field input into a high energy line break walkdown procedure.

The inspection team made the following observations:

Stone & Webster has not performed a field walkdown to confirm that their analytical approach to identify target/break interactions worked.

The drywell model appears to be usable as an aid in identifying small bore piping targets prior to actual field walkdown.

The drywell model does not appear to be usable as an aid in identifying electrical conduit or junction boxes.

The Construction Systems Associates, Inc. three-deminsional computer space model is currently not available as a method of identifying potential target/break interactions.

Stone & Webster appears to have a very preliminary draft of a high energy line break field walkdown procedure which has not undergone review by the Site Engineering Group.

BASIS

The above observations are not deficiencies or unresolved items because there are no requirements specifying how to ensure that plants have adequate high energy line break protection. Because a field walkdown has not been performed to date, the team was unable to confirm that any of the above observations will result in errors.

RECOMMENDATION

The inspection team is concerned that insufficient preliminary work has been performed to ensure that a field walkdown prior to building turnover will be effective in verifying that the plant is protected against high energy line breaks. The team is concerned that the purposes of a building turnover walkdown may be too diverse for an adequate verification for high energy line breaks affects. It is the team's experience that walkdowns which are not focused on a particular task are not effective. The team has the following recommendations:

Plan to have a walkdown which has a sole purpose to verify the adequacy of the plant's high energy line break protection.

Prepare a high energy line break walkdown procedure with the aid of Site Engineering Group personnel.

O2.5-1 (Observation) continued.

Ensure that personnel assigned to the walkdown have a thorough understanding of the underlying safety issues involved. This could be accomplished by training of walkdown personnel or by assigning personnel from the Cherry Hill office who have been involved in the development of high energy line break targets and the evaluation of break effects along with field personnel.

Plan to have a shakedown of the high energy line break walkdown procedure and the high energy line break walkdown team by walking down a few high energy line break locations. It is expected that this approach will minimize interference with construction trades and will provide training to team personnel and insight into the viability of the walkdown procedure.

O4.4-1 (Observation) Timeliness of Verification of Design Calculations

BACKGROUND

During review of the Primary Shield Wall it was found that in several instances the calculations had not been checked for a long period of time.

DESCRIPTION

Pages No. 127-143, entitled "Local Analysis" (dated November 8, 1979 through November 13, 1979); pages 402-426 "Supplemental Calculations" (dated October 15, 1979–October 16, 1979); and pages 111A-111S, "RPV Insulation and Bio-Shield Wall Thermal Analysis" (dated March 27, 1975) had no evidence of being checked.

BASIS

Section 17.1.II.3E2.b of the Standard Review Plan "Quality Assurance During the Design and Construction Phases," states that "construction site activities associated with a design or design changes should not proceed without verification past the point where the installation would become irreversible." However, the Standard Review Plan is not a substitute for regulatory guides and regulations and compliance with it is not required unless licensees have made commitments to follow portions of the Standard Review Plan.

In this case, pages 111A-111S have been marked "Unchecked Calculations" and should have been listed as such on the title page.

RECOMMENDATION

We recommend that the design verification be evidenced by signing for the particular verification activity in a traceable timely manner.

REFERENCES

1. S&W Calculation No. 12210-201.120-068, The Primary Shield Wall, Rev. 0, January 10, 1984 and April 9, 1984.
2. Standard Review Plan (NUREG-0800), Section 17.1, Quality Assurance During the Design and Construction Phases, Rev. 1, July 1981.

O6.4-1 (Observation) HPCS Division 3 Dependency upon ESF Divisions 1 and 2

BACKGROUND

General Electric design criteria for the General Electric-supplied High Pressure Core Spray System and its integral diesel generator require that the system be capable of operating independently of normal auxiliary AC power, plant service air, or the emergency cooling water system. The objective of this criterion was to assure that the Division 3 High Pressure Core Spray System would be independent of other Engineered Safety Feature supporting systems in accomplishing its accident mitigation purpose.

DESCRIPTION

A design change implemented by Stone & Webster provides that redundant Division 1 and 2 loops of the Standby Service Water System supply cooling water to the Division 3 High Presssure Core Spray System diesel generator jacket cooler. This change eliminated the self-driven cooling water pump normally provided by General Electric, and makes the High Pressure Core Spray System dependent upon both the auxiliary AC power and emergency cooling water Engineered Safety Feature systems.

BASIS

Fulfillment of the General Electric design criterion for High Pressure Core Spray System independence from these Stone & Webster supporting systems has not been achieved. With this design change, overall reliability of the three Emergency Core Cooling Systems is essentially reduced to that provided by two independent Emergency Core Cooling Systems.

Despite this reduction in predicted reliability, the present River Bend design satisfies existing industry standards and NRC regulatory requirements with regard to the single failure criterion and a minimum redundancy of two independent core cooling systems.

RECOMMENDATION

Performance of probabilistic risk assessment for the River Bend Emergency Core Cooling Systems and their Engineered Safety Feature supporting systems would contribute to additional assurance that sufficient reliability is achieved for all postulated accident scenarios. The team recommends that the licensee consider the benefits of such an effort.

REFERENCES

1. General Electric Electric Equipment Separation Specification, 22A3728, Revision 4, 3/26/81.
2. FSAR Figure 9.2-1, Service Water P&ID.
3. General Electric Purchase Specification, 21A9236, HPCS Diesel Generator, Revision 4, sheet 9, paragraph 4.3.6.1, 7/9/75.
4. General Electric Design Specification, A62-4380, Emergency Core Cooling System Network Specification.
5. FSAR, page 8.3-59, HPCS Compliance with IEEE-387.
6. FSAR, page 8.3-60, HPCS Diesel Generator Physical and Electrical Independence from Standby Diesel Generators.
7. General Electric Drawing, 22A3743, Rev. 2, 12/29/82.

O6.5-1 (Observation) Instrumentation Separation within BOP ESF Systems

BACKGROUND

General Electric Boiling Water Reactor plants make extensive use of four individual sensors connected in a one-out-of-two-twice coincidence logic for each separation division. To realize the full reliability advantage of having four sensors within a single separation division monitoring one variable, redundant power sources for electrical independence and redundant instrument racks for mechanical independence are usually employed. In the Boiling Water Reactor example, the four sensors use two power sources and two racks to match the characteristics of the one-out-of-two coincidence. In the two-out-of-four coincidence widely used with Pressurized Water Reactor plants, four power sources and instrument racks are used. Moreover, in those cases where the reliability and independence requirements are less important, a simpler configuration of either one-out-of-two or two-out-of-two is used in conjunction with one power source and one instrument rack.

DESCRIPTION

The Stone & Webster initiation circuits for the Standby Service Water System have adopted the Boiling Water Reactor concept of one-out-of-two-twice sensors for low pressure measurements made at the Normal Service Water and Reactor Plant Component Cooling Water headers. Even though an individual process tap and an instrument tubing line has been provided for each sensor, no attempt has been made to achieve mechanical or electrical subchannel separation among the four redundant sensors within a given separation division. Consequently, the Stone & Webster design may not realize the full reliability advantage afforded by having four redundant sensors within each division.

BASIS

Use of four individual sensors in a one-out-of-two-twice coincidence logic is done to achieve high reliability and to avoid spurious actuation of Engineered Safety Feature Systems, and requires coordination of mechanical (i.e. physical separation) and electrical (i.e. isolation) aspects on a subchannel basis.

No specific industry standard, code, or NRC regulatory requirement is violated by the existing Stone & Webster design. The design concept used by Stone & Webster does not recognize the need for internal subchannels within a separation division, and therefore, does not impose additional separation requirements. The team observed that the Standby Service Water System initiation logic may not achieve the level of reliability that four redundant sensors per measurement would ordinarily provide.

RECOMMENDATION

No hardware or documentation change is deemed necessary at this time. If a probabilistic risk assessment (see Observation O6.4-1) is performed for River Bend Station, this area should be examined to determine whether the Standby Service Water System meets the reliability goals needed to support the Emergency Core Cooling Systems under postulated accident conditions.

REFERENCES

1. General Electric Electric Equipment Separation Specification, 22A3728, Revision 4, 8/26/81, paragraph 4.6, "Compatibility with Mechanical Systems."

O6.10-1 (Observation) Reversed Instrument Positions on Mercury (Norwood) Rack

BACKGROUND

Safety-related Balance of Plant instruments for the Standby Service Water and Reactor Plant Component Cooling Water Systems are mounted on four racks provided by Mercury of Norwood, Massachusetts in accordance with Stone & Webster purchase specification 247.411. General arrangement drawings were provided to the vendor by Stone and Webster to show approximate mounting locations for each instrument. Rack drawings prepared by Mercury accurately reflected these desired instrument locations.

DESCRIPTION

Mercury installed two pressure transmitters in the wrong location on instrument rack 1JPB*RAK1. Pressure transmitter 1SWP*PT21B was installed in the location designated for 1SWP*PT21D and vice versa. This reversal was not detected by Stone & Webster Product Quality Assurance during their acceptance inspection of the rack. However, the error was detected during installation of instrument tubing to the rack.

BASIS

Two pressure transmitters on instrument rack 1JPB*RAK1 were not installed by the rack manufacturer in accordance with the design drawings. Hardware and documentation changes have been implemented by Stone & Webster to leave the instruments in their "as shipped" locations, and to reroute instrument piping and electrical wiring to accommodate this reversal. The team noted that Gulf States Utilities will need to be alert to this reversal when performing instrument calibration and repair on these two instruments since the customary "B then D" arrangement pattern is reversed on this one rack.

RECOMMENDATION

This error appeared to be random for the River Bend project. Gulf States Utilities should assure that subsequent work performed by Mercury meets the design and fabrication requirements provided in procurement specifications and drawings.

REFERENCES

1. Stone & Webster Instrument Rack Drawing, EK-S5G-1, Revision 1, 4/17/81.
2. Mercury Rack Construction Detail Drawing, PC-N19810-601, Revision 1, 6/24/82.
3. Mercury Rack Wiring Detail Drawing, PW-N19810-702, Revision 1, 2/8/82.
4. Stone & Webster Quality Assurance Inspection Report, R111296002, 3/7/83.
5. Stone & Webster Nonconformance and Disposition Report, 4146, 9/14/83.

O6.12-1 (Observation) Instrument Calibration Hysteresis

BACKGROUND

Two process instrument manufacturers reviewed during a previous Integrated Design Inspection performed transmitter output tests versus applied inputs using both increasing and decreasing directions for the particular variable. For example, pressure transmitters were tested using five input values as input pressure was increased, and these same five input values were used as pressure was decreased. A small, but nevertheless distinct, hysteresis effect between the increasing and decreasing pressure values was observed in the test results reviewed at each manufacturer's plant.

DESCRIPTION

Three pressure transmitter calibration records performed at the River Bend site by Gulf States Utilities personnel for the Standby Service Water System were reviewed during the inspection. Two of these records demonstrated the hysteresis effect between increasing and decreasing pressure input values. However, one record indicated that the transmitter had identical output values regardless of whether the pressure input was increasing or decreasing in magnitude (i.e., no hysteresis characteristic).

BASIS

An inspected calibration record for one transmitter indicated that it provided an idealized output versus input response with regard to hysteresis. The team believes that this might be an unreasonable level of performance to expect for this transmitter, and that sufficient care may not have been used in performing the calibration tests for this particular transmitter.

RECOMMENDATION

A more extensive review of similar transmitter calibration records performed at the River Bend site should be accomplished by Gulf States Utilities to determine whether this was an isolated occurrence and whether calibration procedure or instrument technician training changes should be made. In addition, comparison of the Gulf States Utilities test results for this particular transmitter with those obtained by the transmitter manufacturer should be accomplished.

REFERENCES

1. Gulf States Utilities Calibration Test Report, 1.ILCCP.011, transmitter 1CCP*PT1A, Rev. 0, 10/11/83.
2. Gulf States Utilities Calibration Test Reports, 1.ILCCP.012 and 1.ILCCP.013, transmitters 1CCP&PT1B and C respectively, Rev. 0, 10/11/83.
3. Gulf States Utilities Device Calibration Procedure, MPC-4000, unissued draft.
4. Gulf States Utilities Loop Calibration Procedure, MPC-4001, unissued draft.

O.A.1-3 (Observation) Compliance of Calculations with Established Stone & Webster Procedure Preparation

BACKGROUND

Stone & Webster procedure EAP 5.3 used for the preparation of calculations requires that individual assumptions and inputs be identified if there is a need for later confirmation.

DESCRIPTION

Two of four calculations reviewed (Reference 2) and (Reference 3) were found not to provide any indication on an individual basis of the need or lack of need for later confirmation. Another calculation (Reference 4) did indicate a confirmation status for each assumption, input or reference but in every instance the status was "no confirmation required." In two of these calculations (Reference 1) and (Reference 4) there are items that in the team's opinion should be later confirmed as the design progresses. One calculation (Reference 5) did indicate a single item as needing "later confirmation," but another item should also have been so marked. The type of items that were omitted were certified pump head curves and system isometrics.

BASIS

Engineering Assurance procedure 5.3 further requires that this aspect (confirmation items) of a calculation be reviewed/checked. All three calculations had their cover sheet signed as being reviewed but the conditions mentioned above were not picked up. This would lead to the conclusion that the review process had failed. A potential impact would be in the event of a modification that needs information from an up-to-date calculation. It is suspected that this is systematic with mechanical calculations.

RECOMMENDATIONS

The team recommends a sampling review of the mechanical calculations to evaluate this situation. Where problems are indicated reanalysis or update should take place.

REFERENCES

1. S&W Procedure EAP-5.3 with Attachment 3.0.
2. S&W Calculation PN-048, Subsystem Fill Pump Sizing, Rev. 1.
3. S&W Calculation PN-307, Piping Mimimum Wall Thickness, Rev. 2.
4. S&W Calculation PN-268, RHR Pump NPSH and Total Head, Rev. 2.
5. S&W Calculation PN-263, Sizing Orifice On RHR, LPCS, HPCS And RCIC Subsystem Fill Pumps Recirculation Lines, Rev. 0.