U.S. NUCLEAR REGULATORY COMMISSION
Region I

Report No. 50-354/85-58

Docket No. 50-354

License No. CPPR-120 Category A

Licensee: Public Service Electric and Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Facility Name: Hope Creek Generating Station

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: December 2-13, 1985

Inspectors:	Karl A. Manoly K. Manoly, Lead Reactor Engineer	1/23/86 Date
	Karl A. Manoly for S. Reynolds, Lead Reactor Engineer	1/23/86 Date
	Audrey L. Lodewyk A. Lodewyk, Reactor Engineer	1/23/86 Date
	Fred Paulitz F. Paulitz, Reactor Engineer	1/22/86 Date
	C Woodard C. Woodard, Reactor Engineer	1/22/86 Date
	RW Winters R. Winters, Reactor Engineer	1/23/86 Date
	for PW Eselgroth R. Gramm, Senior Resident Inspector, NMP #2	1/30/86 Date
	for PW Eselgroth R. Fuhrmeister, Reactor Engineer	1/30/86 Date
	L. Wink L. Wink, Reactor Engineer	1/27/86 Date
Approved By:	PW Eselgroth P. Eselgroth, Chief, Test Programs Section, DRS	1/30/86 Date

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Inspection Summary: Announced As-Built Team Inspection on December 2-13, 1985 (Report Number 50-354/85-58)

Areas Inspected: As-built inspection in the areas of Mechanical, Electrical, Instrumentation and Control, and Structural Systems. The inspection also included a review of as-built equipment for selected emergency procedures, and the FSAR accident analysis assumptions.

Additionally, the licensee actions on previous NRC inspection items and IE Bulletins were also addressed. The inspection involved 829 hours.

Results: No violations were identified. The inspectors determined that the systems selected were constructed in conformance to their FSAR descriptions. Four unresolved items were identified in the areas of piping component and equipment supports and instrumentation and controls.

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DETAILS

1.0 Scope and Purpose of the Inspection

This as-built team inspection was conducted by region-based reactor engineers to verify that selected systems were constructed substantially in conformance to the description contained in the Final Safety Analysis Report (FSAR) and in NRC's Safety Evaluation Report (SER). The inspection included examination of fluid systems, Heating, Ventilation and Air Conditioning (HVAC) systems, ac and dc power systems and instrumentation and controls systems. Extensive system walkdowns were performed, during which independent dimensional measurements were made. Also, various project specifications drawings and design calculations were reviewed.

In general, the systems selected for inspection were those associated with meeting reactor safe shutdown and core cooling requirements. This as-built inspection focused particular attention in the following specific areas:

- The shutdown cooling functional systems between the Delaware River and the reactor core.

- Systems and equipment necessary to fulfill the functional requirements of steps, as currently written, in selected Emergency Operating Procedures.

- A compatibility check of the plant as-built condition with selected aspects of plant design specified in the transient analysis portion of the FSAR.

2.0 Persons Contacted

Public Service Electric and Gas Company (PSE&G)

* C. McNeill, Vice President - Nuclear
* R. Salvesen, General Manager - Hope Creek Operations
* F. Cielo, Principal Engineer
* J. Nichols, Technical Manager
* M. Massaro, Lead Engineer
* J. Duffy, Site Engineer
* M. Metcalf, Quality Assurance (QA) Startup Engineer
S. Hilditch, Jr., QA Engineer
R. Donges, Lead QA Engineer
* A. Meyer, Senior Staff Engineer
* C. Allen, Technical Engineer
* R. Griffith, Principal QA Engineer
R. Audette, Facilities Manager

* M. Idell, Lead Engineer
* M. Kobran, Lead Engineer
* T. Ram, Supervising Engineer
* R. Campanella, Licensing Engineer
* W. Merritt, Lead Engineer
* T. McLaughlin, QA Engineer
  W. Mussel, Engineer
  W. Mitchell, Supervisor, Document Control
  M. Finney, Supervisor, Document Control
  R. Ritzman, Supervisor, Technical Document Room
  A. Koa, Principal Engineer
  A. Sternberg, Principal Engineer
  J. Defabo, QA Engineer
  J. Montgomery, Staff Maintenance Engineer
* A. Giardino, Manager, Station QA
* M. Mortarulo, Senior Staff Engineer
* W. Denardi, Engineer
  S. Maginnis, Senior Staff Engineer
* R. Binz, IV, Senior Staff Engineer
* J. Ranalli, Senior Staff Engineer
  H. Chu, Electrical Engineer
  D. Schumaker, Civil Engineer
  M. Massard, Site Engineer
* A. Taylor, Safety Review Engineer
* G. Connor, Operations Manager
* B. Preston, Manager - Licensing and Regulation
* C. Churchman, Site Engineering Manager
  N. Dyck, Chairman, Response Coordination Team

Bechtel

* B. Markowitz, Project Manager
* J. Isaacs, Deputy Group Supervisor
* A. Hargrove, Resident Quality Engineer
  R. Cole, Lead QA Engineer
* R. Goebel, QA Engineer
* G. Moulten, Principal QA Engineer
* C. Headrick, Principal Quality Control (QC) Engineer
  T. Giordano, I&C Engineer
  M. Metcalfe, I&C QC Engineer
  J. Danhert, QC Engineer
  E. Hanselman, Lead Field Welding Engineer
  M. May, Assistant Lead Field Welding Engineer
  J. O'Conner, Field Welding Engineer
* C. Haynes, Resident Engineer, Plant Design

General Electric

* T. Bloom, Resident Site Manager
* J. Cockroft, Engineer

U.S. Nuclear Regulatory Commission (USNRC)

* L. Bettenhausen, Chief, Operations Branch
* J. Strosnider, Chief, Reactor Projects Section 1B
* R. Borchardt, Senior Resident Inspector
* J. Lyash, Resident Inspector

* Denotes individuals present at exit meeting.

Throughout the course of the inspection, other licensee, Bechtel and General Electric engineers and technical personnel were also contacted.

3.0 Mechanical Systems

3.1 General

The scope of inspection in the area of mechanical systems covered piping components, equipment and HVAC systems and their respective supports. The specific systems which were inspected in the piping area included:

- Service Water
- Safety Auxiliary Cooling System
- Residual Heat Removal System
- Control Rod Drive System

The inspection of piping components included the residual heat removal system motor operator valves and the main steam safety relief valves.

The inspection in the HVAC area focused on the recirculation system for the diesel generator room and the unit cooler for the RHR "A" pump.

The objective of this inspection was to verify, by sampling review, that the above systems were designed and fabricated such that they were capable of performing their intended functions as specified in the Final Safety Analysis Report (FSAR) and whether the as-built configurations were in conformance with the FSAR, the SER and system specifications and drawings.

3.2 Piping Systems

The inspection in this area included piping components, equipment and supports. A review of the licensing documents was performed to insure that, for those selected systems, FSAR commitments were correctly translated into specification procedures and drawings. A cross review was also performed of the Piping and Instrumentation Diagrams (P&ID's) and support detail drawings to verify their consistency and agreement with the as-built installations.

3.2.1 Piping As-Built Reconciliation Program

The objective of the NRC review of the piping As-Built Reconciliation (ABR) Program was to assess the various functions and activities contributing to this program and to determine whether acceptable engineering practices, regulatory requirements and licensee commitments had been met. The regulatory positions for the evaluation of as-built safety related piping and support systems are addressed in IE Bulletin 79-14.

To achieve the above objective, the NRC staff performed a review of the governing specifications and procedures which govern this activity in addition to conducting walkdown inspections of selected piping systems and support installations utilizing applicable as-built isometrics and support drawings.

3.2.1.1 Piping As-Built Reconciliation Overview

The as-built reconciliation is performed for nuclear class 1, 2 and 3 large and small bore piping systems and ANSI B31.1 large and small bore piping systems within "Qs" and "Qsh" boundaries. The designation "Qs" identifies the portion of piping beyond the ASME boundary (code break) up to a first anchor or terminal end of the piping run. The designation "Qsh" identifies the non-category I piping in the vicinity of "Q" piping and which is designed to mitigate the consequences of seismic interaction with "Q" piping.

The requirements for the as-built reconciliation of designated piping systems and the description of the activities and work flow performed by the stress group are provided in the Technical Specification P-450(Q) for As-Built Reconciliation. The detail of actions taken by field engineering and quality control to support project engineering for As-Built Reconciliation is provided in a desk top procedure (ABR/DTP0001).

Documentation of as-built piping and support installations is contained in as-built reconciliation packages (ABP) which consist of final pipe support status and index sheets, system isometrics, hanger detail drawings and the stress calculation cover sheets. Stress reports for nuclear class 1 piping are included in the ABR packages as well.

The information identified by field engineering and shown on class 1, 2 and 3 large bore (2½ inch and larger) and class 1 small bore (2 inch and smaller) piping isometrics include: as-built configuration; actual location of pipe supports within 2 inches from design location; outstanding Field Change Requests (FCR's) and Field Change Notices (FCN's) against the piping system; location of lugs for nuclear class 1 piping; grouted-in penetrations; location of components, valves, fittings, flow elements, expansion joints and other in-line components; location of pipe whip restraints and bumpers; type of branch connections other than a tee, such as half coupling, weldolet, sweepolet, threaded connection; valve stem and operator orientations within ±5 degrees; vent, drain and root valve configurations; and as-built configuration of field routed small bore piping connected to large bore within Qs boundaries.

Details identified on class 1, 2 and 3 large bore and class 1 small bore pipe hangers include: type of support, orientation of gaps for whip restraints; length of welds on tube steel if less than the full width; cancelled supports in the AB portion of the isometric; and supporting calculations for small bore support designs by field engineering.

As-built information identified on class 2 and 3 small bore piping isometrics include: as-built configuration; actual location of all pipe supports; orientation and type of all supports; location of supports/guides added by field engineering; U-bolt locations, substitutions of straps by U-bolts; length of welds on tube steel if less than the full width; penetration numbers; substitution of bends for ells or ells for bends; branch connections other than a tee; location of components, valves, fittings and other in-line components; changes to CRD insert and withdrawal piping; valve and stem operator orientation exceeding ±5° from design; root valve AB configuration; and individual hanger details.

Completed ABRs are sent to Bechtel's corporate office in San Francisco for verifying pipe stress orientation and symbols and updating of isometrics. Depending on the as-built conditions, isometrics are either issued or reconciled by the ABR team. The reconciliation calculations are incorporated in either the original or the up-date of pipe stress calculation. Identified modifications are performed before the issuance of the final pipe stress and pipe support calculations. This step is followed by the preparation of the ABR packages which contain the N-5 letters, the pipe stress calculation cover sheets and the pipe support calculation numbers and revisions. Assembled ABR packages are used for the preparation of the N-5 packages.

The following information was gathered by the NRC staff during the review of the ABR program:

- approximately 30 large bore and 20 small bore related modifications have resulted from the ABR activities.

- 343 large bore stress calculations were completed out of 345 total calculations.

- 186 small bore stress calculations were completed out of a 190 total calculations performed for piping inside the drywell.

- 2293 small bore isometrics were completed out of a 2570 total isometrics of piping outside the drywell.

3.2.1.2 Walkdown Verification of As-Built Piping and Support Installations

The verification of as-built installations was performed either by visual inspection or by independent measurements of accessible components and supports.

The criteria used for the assessment of piping components and supports were those described in the installation specifications for these components. The inspection attributes included verification of the following:

- linear and angular measurements related to piping runs and support locations;
- branch connection types and locations;
- piping bend and elbow radii;
- support mark numbers, functions and locations;
- proper flow direction marks on valves;
- correct sequential location of valves on piping runs; and,
- proper identification and orientation of valves and Limitorque operators.

The inspection attributes for equipment (pumps, heat exchangers, etc.) included verification of the following:

- manufacturer specification and purchase orders;
- name plate data consistency with FSAR requirements and manufacturer's data (capacity, type, rate head, horse power); and,
- heat exchanger component class (tube side and shell side).

The inspection attributes for pipe supports included verification of the following:

- as-built configuration against support detail drawing (BZ series) including dimensions of members;
- connection to the proper structure;
- sizes and quality of welds on hangers, included welded attachments to piping;

- baseplate dimensions and location of structural attachment to baseplates;
- baseplate bolt (concrete expansion or Richmond insert) tightness edge distance and the bolt mark identification for Hilti bolts;
- restraint bleed holes open and free of foreign material;
- load setting of spring hangers;
- grouting of floor mounted baseplates and gap sizes for wall mounted plates; and,
- pipe routing and support locations such that movements of piping due to vibration, thermal expansion, etc., would not likely cause contact with other pipes, supports, equipment or components.

3.2.2 Station Service Water System (SSWS)

3.2.2.1 Piping System Walkdown

The SSWS provides river water to cool the Safety Auxiliary Cooling System (SACS) heat exchangers during a loss-of-coolant accident and other design basis accidents. The SSWS removes heat from the SACS heat exchanger and transfers the heated water to the cooling tower discharge canal.

The SSWS consists of two redundant loops. Each loop contains two service water pumps, traveling water screens, service water strainers, spray water pumps and associated valves, piping and instrumentation. Each loop cools a separate SACS loop.

During this inspection, portions of loop "A" of the SSWS were selected for the purpose of as-built verification. The walkdown of piping components and supports was conducted from service water pump AP-502 to SACS heat exchanger AE-201. A detailed inspection was conducted on the accessible portions of the system located within the intake structure. The inspection included those attributes listed in Section 3.2 of this report.

3.2.2.2 Service Water System Materials Evaluation

The inspectors conducted an evaluation of the material selection and installation of safety related service water piping and service water cooled heat exchangers. The material review was based on previously identified pre-service and operational corrosion problems at other sites including Salem with concentration cell corrosion of stainless steel weld metal and copper alloy heat exchanger tubing (the Salem problem was with stainless weld metal). The review showed that the safety related service water piping consisted of epoxy phenolic coated pipe, coal tar epoxy coated pipe for a wall penetration and reinforced concrete pipe which will negate the oxygen concentration cell problem. The SACS HX tubing is titanium with the water box inside surface lined with alloy 625. The inspector noted that the licensee is committed to visual inspection of the HX water boxes each refueling outage to insure there are no holidays in the alloy 625 lining and tube sheet overlay that could cause rapid galvanic small anode/large cathode corrosion problems.

3.2.2.3 Service Water Pump

The inspectors closely examined the Ingersoll Rand (IR) service water pump installation and other adjacent equipment in the pump house. The inspectors noted rusted carbon steel studs and bolts on stainless steel valves, but were provided licensee documents previously identifying this as a potential maintenance problem and committing to corrective action. The inspectors reviewed the IR documentation package and confirmed that the FSAR requirements were reflected in the code data sheet and code data plate.

3.2.3 Safety Auxiliary Cooling System (SACS)

3.2.1 Piping System Walkdown

The SACS is designed to provide cooling water to various engineered safety features equipment, including the residual heat removal (RHR) heat exchanger. Water from the SACS is pumped through the RHR heat exchanger tube side to remove heat from the process for containment cooling during various modes of operation and during loss of off-site power and loss of coolant accidents.

Loop "A" of the SACS was selected for as-built verification. The walkdown of piping components and supports was conducted from SACS pump AP-210 to RHR heat exchanger AE-205 to SACS heat exchanger AE-201 and returning to SACS pump AP-210. A detailed inspection was conducted in accordance with Section 3.2 of this report on the accessible portions of the system loop.

At the time of the NRC inspection, the SACS loop "A" piping had been inspected in accordance with the licensee as-built verification program. The inspection included review of ABR package C-1749 Phase II copy for Large Bore Pipe SACS Loop "A" and the associated ABR isometric drawings 1-P-EG-06, Rev. 12(Q) and 1-P-EG-13, Rev. 12(Q).

3.2.3.2 SACS Heat Exchanger (Hx)

The inspectors reviewed in detail the Graham Manufacturing Co. Inc. SACS Hx documentation package and visually inspected the exterior welds of the Hx, Hx saddle support and Hx structural box supports for the vertically stacked Hx's. The Hx is designed to ASME Section III-ND and the supports to Section III-NF. Conspicuous in the documentation package is NCR 1237 which was initiated by Bechtel after visual inspection revealed potential welding deficiencies in some of the Hx nozzle welds and the heat exchanger supports. This NCR resulted in a complete re-examination by both Graham and Bechtel QC inspectors to explicit criteria indicated in Graham Inspection Procedure 39047-T (which was signed by the Graham cognizant Design Engineer). The re-examination accepted most of the welds and required repair (by Bechtel) of other welds. All repair welds received Magnetic Particle surface examination. The inspector reviewed the welding QC documentation for the repair welds.

An independent inspection was made by the NRC inspectors of 141 welds in the saddle supports and structural box (framework) supports. The inspection showed that all but three welds met the basic fillet weld size requirements and that the three welds met the Inspection Procedure 39047-T II, 2, A, 1 alternative requirements. The voluminous NCR 1237 was determined to be acceptable.

The inspectors noted the absence of A193B7 stamping on the heat exchanger water box plate cover fasteners, but confirmed the material properties by review of the fastener material certifications.

Visual inspection of the Code Data Plate indicated conformance to the Code Data Sheet and minimum FSAR requirements.

The inspector reviewed the maintenance records for the inerting gas protection for the shell side of the heat exchanger.

3.2.3.3 Pumps

The inspectors reviewed the SACS pump documentation package, installation, and Code Data Plate. The inspectors reviewed the Code Data Plate, Code Data Sheet, pump curves and minimum FSAR requirements and verified conformance.

3.2.4 Residual Heat Removal (RHR) System

The RHR system design functions are to remove decay heat from the reactor system during shutdown cooling, to provide Low Pressure Coolant Injection (LPCI) during a design accident, to provide torus spray cooling to limit temperature rise in the torus, and to provide drywell spray cooling to reduce the internal drywell pressure that would accompany a line break accident.

The RHR system consists of four independent loops with motor driven pumps. Two loops are provided with heat exchangers cooled by the Safety Auxiliaries Cooling System (SACS). All loops are capable of the LPCI function while only the heat exchanger loops are used for normal or emergency vessel cooling and primary containment spray cooling.

RHR system loop "A" was selected for the purpose of the as-built verification. The walkdown of piping, supports and components was conducted from the RHRS torus suction nozzle (P-211C), to the RHR "A" pump (AP-202), and the pump discharge lines to the RHR heat exchanger (AE-205), to the torus spray nozzle (P-214B), to the LPCI injection nozzle (N-17C), and up to and including the lower drywell spray header. A detailed inspection was conducted on the accessible portions of the system.

3.2.5 Safety Related Motor Operated Valves

Stroke times for selected valves in the Residual Heat Removal System were reviewed against FSAR requirements, Technical Specifications limits, and General Electric (GE) Design Specification Data Sheet (DSDS). The DSDS values (as modified by FDDR's KT1-571 and KT1-1457), and the FSAR values (as modified by FSAR Change Notice 1040 to be issued as amendment 14) agree with the Technical Specification limiting stroke times. The actual measured stroke times (from PTP-BC-1) are within the allowable limits. The particular valves involved were 1-BC-HV-F015A&B, 1-BC-HV-F0008, 1-BC-HV-F009, 1-BC-HV-F022, 1-BC-HV-F023, 1-BC-HV-F024, 1-BC-HV-F010, 1-BC-HV-F027 and 1-BC-HV-F017A, B, C, & D.

The Torus suction valves were reviewed for design data as opposed to actual conditions which could be expected when lined up for shutdown cooling operation. The purchase specification data, manufacturer's data sheet, Line Index and valve nameplate data were compared to calculated conditions in the line. No discrepancies were noted.

The motor operated valves (and air operated valves) associated with the steam condensing mode of the residual heat removal system were inspected to verify deactivation. This was in accordance with the deletion of the steam condensing mode per FDDR KT1-1323. 1-BC-PV-F051A&B and 1-BC-LV-F052A&B were verified to have the airlines disconnected from the actuators and capped. 1-BC-PV-F051A&B were further verified to be in the closed position with the handwheels chained and locked. The following motor operated valves were verified to have their handwheels locked with the valves in the closed position: 1-BC-HV-F026A&B, 1-BC-HV-F011A&B, 1-BC-HV-F052A&B, 1-BC-HV-4420A&B, 1-BC-HV-4421, 1-BC-HV-4428. In addition, their supply circuit breakers were verified to be danger tagged in the open position.

3.2.6 Control Rod Drive System (CRDS) Scram Discharge Volume (SDV)

The function of the CRDS is to control changes in core reactivity by positioning neutron absorbing control rods within the reactor core. The CRDS hydraulic system supplies and controls the pressurized fluid for control rod drive movement. During a scram, or rapid insertion of the control rods, water is discharged from the control rod drives to the SDV.

The SDV consists of two sets of 12 inch diameter header piping, one header for each bank of Hydaulic Control Units (HCUs). The header slopes downward to a 12 inch vertical Scram Discharge Instrument Volume (SDIV). The SDIV is provided with a 2 inch drain line and a redundant set of level switches and transmitters.

The north SDV system was selected for an as-built verification. The walkdown was conducted from selected HCU exhaust lines, the SDV (1-BF-040-S06/S07/S08), the SDIV (1-BF-040-S05), the vent line from the SDV to the outboard vent valve (V083), the SDIV drain line to the outboard drain valve (V076), and the piping associated with the SDIV level switches and level transmitters LSN 13A/B/G/H and LTN 12 D/C).

3.2.7 Main Steam Safety Relief Valves (MSRVs)

The main steam safety relief valves are part of the nuclear pressure relief system. These MSRVs protect against overpressurization of the reactor coolant pressure boundary. Selected MSRVs are also part of the automatic depressurization system which functions as part of the emergency core cooling system for events involving small breaks in the reactor coolant pressure boundary.

There are fourteen (14) MSRVs mounted on the main steam lines between the reactor pressure vessel and the inboard main steam isolation valves in the drywell at approximately the 124' elevation.

A visual inspection was made of the installed MSRVs. MSRV name plate data was verified to be in accordance with FSAR requirements. A review was also performed of the valve manufacturer's certification of design and performance requirements, including the results of testing required by the FSAR.

3.2.8 Review of Post Weld Heat Treatment (PWHT) For Feedwater Piping

In the process of reviewing Bechtel specification P202 for fabricating piping systems it was noted that paragraphs 3.2 and 3.2.3 indicate a different Code Edition (1977 W78) for PWHT than that utilized for fabrication (1974 W74). As this is an unusual practice, the inspector reviewed the justification for the difference. There were six instances involving Dravo supplied pipe where the wall thickness of the pipe exceeded 1.5 inches due to excessive ID counter bore which had been compensated by excessive OD weld buildup. Bechtel made the decision to PWHT these weld joints. The weld joint QCIR records including time temperature charts were reviewed by the inspector. The welds in question were AE-003 FW6, AE-017 FW18, AE-017 FW2, AE-017 FW17, AE-017 FW1 and AE-017 FW 22. In the process of conducting PWHT operations, Bechtel experienced difficulty obtaining the minimum temperature requirement (Table NB-4622.1-1) of 1100F. This problem led to the utilization of the alternate holding time - temperature rules of Table NB-4622.4(c)-1. Review of the 1974 W74 NB-4622.4(c)(1) indicated the requirement for requalification of the SC IX PQR (for the WPS) regardless of "P" grouping. Further review of this requirement by Section III in the 1977 W78 of the ASME code clarified that the retesting of the PQR was limited to P-3 materials, therefore making it not a requirement for P-1. Bechtel FCR P-9188 requested the utilization of the 1977 W78 rules for PWHT. Concurrence to this request was given by the licensee. At a later date, the Bechtel P202 specification was changed.

Visual inspection of the Code Data Plate indicated it conformed to Code Data Sheet and FSAR requirements. The inspector reviewed this item in detail and has no further questions.

3.3 HVAC Systems

3.3.1 Scope

The inspection in this area included HVAC components, ductwork, instrumentation, and supports. A review was performed for the selected systems to insure that FSAR commitments were correctly translated into procedures, specifications and drawings. The HVAC recirculation system for Diesel Generator room 5307 and the unit cooler for the RHR "A" pump room 4113 were inspected. The Diesel Generator recirculation system consists of two 100% capacity fans, two sets of cooling coils, and associated ductwork.

The inspector additionally inspected the low flow instrumentation and tubing for fan AV-412 and the control room alarms and displays related to the DG area ventilation.

3.3.2 Inspection Criteria

The specific inspection attributes for the walkdown included verification of the following:

Duct Inspection

- proper size and location of duct work
- lack of excessive sheet metal deformation
- proper location and installation of flow sensing devices
- completeness of bolted flange connections

Fan (AV-412/EV-412) and Cooling Coil (EV-412/EVE-412) Inspection

- proper connection bolting
- marking and tagging
- proper location and
- nameplate data

Fan (AV-412) and Unit Cooler Supports (AVH-210)

- location and completeness
- dimensions, weld sizes and weld profiles
- proper attachment to embedment plates
- proper anchor bolt installations

3.3.3 Findings Relative to Ducts, Supports and Components in the HVAC System

The walkdown inspection confirmed that the installed items were in accordance with the design requirements and FSAR commitments.

The inspector had no further questions.

3.4 Findings and Conclusion

The NRC inspectors found that the large bore piping as-built program had generally documented the proper as-built dimensions. However, in several cases, discrepancies were identified between the as-built drawings and the installed piping by quality assurance reviews. The discrepancies necessitated licensee generation of both Nonconformance Reports and Engineering design change documents to correct the as-built drawings. The inspectors were informed that none of the discrepancies would negatively impact the validity of the stress reconciliation efforts.

1. The licensee's program for as-built reconciliation of safety-related large and small bore piping systems, and further verification of as-built installation by the inspection team provides adequate basis for the closeout of IE Bulletin 79-14 at Hope Creek Generating Station.

2. The licensee response letters to the NRC regional office on May 23, 1979 and August 15, 1979 and Bechtel letter BLP 16565 to PSE&G on October 1984 regarding IE Bulletin 79-07 (Seismic Analysis of Safety Related Piping Systems) were reviewed during this inspection. The above letters indicated that the various computer codes which were utilized for performing response spectrum seismic analysis of piping systems at Hope Creek Station did not utilize either the algebraic summation of codirectional spatial components or the algebraic summation of codirectional inter-modal response techniques. The licensee's response had also provided a description of the programs used and their verification by other benchmark programs. The licensee submittal was considered adequate for the closeout of IE Bulletin 79-07 at Hope Creek Station.

3. As a result of the review of the specification for installation of pipe supports P-410(Q), paragraph 4.1.1.D, it was identified that fillet welds may be made on either side of the supplementary steel flange or web when the design drawing specifies a weld on only one side of the flange or web. This deviation is applicable to all supplementary steel beam sections, except channels. The inspector indicated that substitution of fillet welds from the outside to the inside flange of wide flanges and angle shapes is not conservative since this results in a reduced weld section modulus, and subsequently increases weld stresses.

The licensee indicated that the pipe support weld design calculations before June 1982 utilized an allowable weld stress of 15 ksi as opposed to the code allowable of 18 ksi. This resulted in a 16.6% conservatism in weld design calculations. Other margins of conservatism in weld design included: a) increase of support design loads by 15% (verified in IDVP Report, Vol. 3, August 30, 1985); b) design practice prior to September, 1982 which utilized conservative code levels A&B allowables in weld design; c) use of enveloped loads at various support joints for sizing of welds; d) rounding up of the decimal calculated welds sizes to the nearest larger fractional size; and e) design of welds to meet the minimum weld size requirements. When the above margins are evaluated collectively, the staff determined that a sufficient margin in weld design was still present in the support installation even when considering a maximum reduction of weld section modules by 18% as a result of substitution of flange welds on W4x13 shapes.

For supports designed after June 1982, the licensee indicated that all welds were either designed or verified, using Bechtel's Standard Weld Design Computer Program ME-120, and the weld qualification portion of ME-150 of the Structural Analysis Program. Weld section modulii in both programs are automatically computed assuming that welds are in the inside of the wide flange beams and the inside of angle legs in structural shapes.

The licensee further indicated that the computer codes ME-120 and ME-150 were also utilized in the design or verification of support welds involved in FCR and FCN modifications.

4. Two cases of closely spaced rigid supports were identified during the walkdown of the Safety Auxiliary Cooling System (SACS) piping from the discharge side of the heat exchanger IA2E-201:

   a) Vertical Snubber No. H64 on line No. 1-P-EG-107, and vertical rigid restraint No. H21 on line No. 1-P-EG-104 were spaced approximately 5'-0" apart.

   b) North-South Snubber No. H32 on line No. 1-P-EG-107 and North-South Snubber No. H22 on line No. 1-P-EG-104 were spaced approximately 32 inches apart.

   The installation of snubbers in proximity to other snubbers, rigid restraints or anchors could result in the inoperability of these snubbers if the dead band in a snubber is larger than the pipe translation between the two successive close supports. A similar problem could also exist if rigid supports were installed in proximity to other rigid supports or anchors.

Typically, this would be caused by the same circumstances which resulted in the closely spaced snubbers identified above and would result in an overloading of the supports and/or the piping if the gaps between piping and supports exceeded certain limits. The inspectors presented these concerns to the licensee and pointed out the need for the identification of all cases in which rigid supports (including snubbers) were placed in proximity to other rigid supports (including snubbers) or anchors.

This item is unresolved pending the licensee response and NRC review (354/85-58-01).

5. During the review of the design specification for the Safety Auxiliary Cooling System Heat Exchanger (M-069), it was identified in paragraph 10.2 that the specification took an exception to the allowable primary design stress limits specified in subsection NF of the ASME Section 3 for the emergency and faulted conditions. This identification was further compounded by an apparent error in the specification regarding the allowables specified in the emergency condition as 1.25 and 1.85 for membrane stress and membrane plus bending stress respectively. This discrepancy was presented to the licensee during the exit meeting of December 13, 1985 as an unresolved item. On December 18, 1985, the licensee provided a response to the unresolved item which indicated the following:

   - The primary stress limits for the emergency condition stated in paragraph 10.2 of specification M-069 had typographical errors. Field Change Notice (FCN-M-2736) has been issued to clarify the stress limits for emergency condition as 1.25 and 1.85 for membrane and membrane plus bending, respectively. The nomenclature "S" is used in the specification to represent "SM" as specified in the ASME code.

   - The primary stress limits for the faulted condition are 1.5S and 2.25S for membrane and membrane plus bending, respectively. The stress limits provided for the emergency condition was found to be consistent with subsection NF of the 1974 edition of ASME Code. However, the membrane plus bending stress limit for the faulted condition was found to exceed that specified in the 1983 edition of the Code. The later edition of the code was used since the 1974 and 1977 editions did not have a specified limit for the above stresses. This item is unresolved pending licensee response and NRC review (354/85-58-02).

6. The walkdown inspection of the CWS, SSWS and SACS piping components and hangers disclosed the following discrepancies between licensee design documents and existing field installations. The nonconformance reports (NCR's) referenced below were issued during the inspection to resolve each associated issue.

   (i) The existing clearance gap between large bore service water piping and rigid restraint 1-P-EA-026-H02 was found to be .049 inches on one side. Pipe support drawing 1-P-EA-026-H02(Q) Rev. 4, FO specifies 1/16 inch gap on both sides. Specification P-410 specified tolerances with a combined total clearance of 3/32 inches minimum. NCR No. 8875 was written to document the actual gap.

   (ii) SACS isometric drawing 1-P-EG-06 shows valve 1-EA-V804 stem in the horizontal position. The existing valve stem is approximately 45° from the horizontal position. This orientation was found to facilitate installation to avoid handwheel operational interferences and to have minimal impact on system stresses and restraint loading. NCR No. 8891 and FCR-P-16150 were issued and approved to document the as-found orientation.

   (iii) A poorly designed, non-functional (unstable) spring can hanger was found installed on a 1 inch diameter fuel pool make-up line. FCR No. PF-12046 was issued to redesign the hanger which was reinstalled and found to be acceptable.

   (iv) SACS pipe support drawing 1-P-EG-107-H06(Q) specified a 5/16" fillet weld where the existing fillet weld measured ¼". NCR No. 8886 was issued to document the nonconforming condition.

   (v) Hanger 1-P-EG-159-H01 clamp was observed by the inspector to be in contact with nearby support steel during SACS operational testing. The hanger was reinspected under non-operating conditions and a .035" clearance was measured. This gap and the hanger drawing design movement is determined as acceptable for QC inspections. Yet, to enhance the functional operation of the spring support, PSE&G will relocate the clamp within the tolerance specified in specification P4.10.

   (vi) Anchor bolt elevation and top of floor elevation for chilled water system tank 1AT401 on drawing C-0399-0 Sheet 294 appeared to be in conflict. This discrepancy was attributed to a drafting error and is to be corrected via a Field Change Notice.

The discrepancies found in the system as-built configuration are being corrected under various licensee programs. The number of items found is relatively small considering the depth and breath of the as-built walkdown verification. The nonconforming conditions found would not significantly impact upon the safe operation of the systems.

7. The inspector identified that SDV vent valves V776 and V777 had been mistagged as V774 and V775 respectively. The inspector was provided Startup Deficiency Report (SDR) BF-270 that documented the North and South vent valve tags had been mistakenly exchanged. The inspector was informed that the tags were reaffixed to the proper valves. The licensee stated that the system P&ID had been used to tag the valves in accordance with procedure SEI 7.4. The scope of procedure SEI 7.4 is limited to instrument root valves and skid mounted valves. Pending licensee review of the tagging program procedural controls, this item is unresolved. (354/85-58-03)

8. The inspector examined small bore support 1-P-BF-435-H3 and found two cases of underlength fillet welds wherein the design specified end returns had not been provided. The inspector reviewed the associated support calculation and ascertained that the end return weldments would not be required to ensure the support load carrying capacity.

   Field Change Request P-16162 was issued to provide engineering criteria to inspect the end return welds. Quality Action Request F310 was issued to ensure training of appropriate personnel regarding end return weldments. The inspector reviewed structural, electrical, and other pipe support design drawings and found in all cases that weld length was specified. The inspector had no further questions.

4.0 Electrical Systems

4.1 General

The objective of this phase of the inspection was to examine the installation of selected portions of the Class 1E ac and dc power systems and to verify that the as-built conditions agree with FSAR and SER descriptions and project specifications and drawing requirements. The portion of the ac system selected for inspection were those associated with the "A Train" station service water system, RHR system, and the SACS system. In the dc power system, the batteries and battery chargers were examined.

4.2 AC Power System

4.2.1 The inspector conducted a field walkdown of the power feeds from the 4160 volt emergency switchgear 10A401 to the motors of the "A Train" Station Service Water Pump 1AP502, RHR Pump 1AP202, SACS Pump 1A210, to unit substation 1AX401. From the unit substation at 1AX401, the 480 volt power feed to the station service water building intake structure Motor Control Center (MCC) 10B553 was walked down. The 4160 volt power feed circuit breakers cable and conduit for the unit substation, the RHR pump and the SACS pump were all contained within the Reactor Building. The 4160 volt and 480 power feeds to the station service water pump and MCC leave the Reactor Building and are pulled through an underground duct bank to the service water building.

The inspector observed workmanship and the as-built conditions of the switchgear, cable, conduit and cable trays noting in particular, the following attributes:

- Switchgear is of the proper size and rating.
- Cable, raceway and cable trays are properly identified including color coding.
- Electrical separation between redundant trains and Class 1E and non-Class 1E cables is maintained.
- Cable, raceway, and cable tray hardware is properly installed.
- Cable support is proper.
- General equipment conditions are good and cleanliness is maintained.
- Cable terminations are proper.

The governing electrical specifications, standards and procedures for installation and acceptance in these areas are the following:

- Specific Work Plan/Procedures SWP/P-E-17 Cable Installation.
- Master Q-C Instructions 10855/E-5.0 Installation Inspection of Class 1E Terminations.

- SWP/P-E-33 Specific Work Plan/Procedure Installation of Electric Control Boards, Control Complex Equipment, Switchgear, Motor Control Centers, Load Centers and Distribution Panels.
- PQ CI.E-4.0, Quality Control Instruction Installation of Class 1E Cable.

4.2.2 Findings

The inspector determined that the identification of cabling, raceways, trays, and conduit was as required by the specification including the color coding. The inspector also noted appropriate cable tray grounding throughout the runs and verified that the cable support routing and termination agreed with the cable pull and termination tickets. Several instances were noted of debris in open ventilated Class 1E cable trays. In each instance, there was either construction activity still in progress (including cable tray cleaning and placing separation covers on the trays) or the areas were being cleaned of debris in preparation for turnover from Bechtel to PSE&G. However, the amount of debris and the frequency of finding it in several different locations including locations already turned over from Bechtel to PSE&G such as the emergency diesel rooms and the station service water building led to a meeting between the NRC inspector, Bechtel and PSE&G management. The inspector concluded that the current Bechtel and PSE&G cleanup programs are adequate to ensure a satisfactory level of plant cleanliness if properly performed. Commitments by both Bechtel and PSE&G management to place additional emphasis on performance of the programs are expected to resolve the problems observed.

The inspector did not observe any electrical separation problems in the equipment and power runs walked down. However, on-going construction work was in progress in various locations throughout the plant to achieve the FSAR cable tray separation requirements by the installation of metal cable tray covers. This program and its status were reviewed. The work was estimated by the licensee to be 85 percent complete with completion projected by December 20, 1985.

No deficiencies were observed in the class 1E ac electrical power systems inspected.

4.2.3 RHR Valve MCC Wiring

A number of motor operated valves (MOVs) in the Residual Heat Removal System were inspected to verify interlocks, logic, control circuits and field wiring.

The control circuits for the Shutdown Cooling Suction MOVs (1-BC-HV-F006A&B) were checked to verify interlocks for preventing vessel blowdown while in Shutdown Cooling Mode of operation. The control circuit is such that the valve cannot be opened unless its associated suppression pool suction (1-BC-HV-F004A or B), test return (1-BC-HV-F024A or B), and suppression pool spray (1-BV-HV-F027A or B) valve are all shut. This is in accordance with the logic diagram and the GE Elementary Diagram. There is however, no interlock which would prevent the opening of one of these other valves while operating in Shutdown Cooling mode. This is also in accordance with the GE design as shown on the elementary diagram. This arrangement will require particular care on the part of the plant operations staff in order to prevent a blowdown of the reactor vessel due to misoperation of these valves (which has occurred at many other sites).

Control circuits for those valves which receive signals on LOCA logic actuation were also reviewed to verify that the overload function was only bypassed upon a LOCA which is in conformance with regulatory position C.1(b) of Regulatory Guide 1.106. The MCC terminations for these valves were inspected to verify conformance with the design as shown on the EE-580 printout (HCG171-3B). This overall area was found to be satisfactory.

4.2.4 Control Panel Inspection

Five control panels were chosen for a detailed inspection of panel construction, seismic qualification, device mounting, and wiring. These panels all contained Class 1E wiring, are safety-related, and located in the Station Service Water System (SSWS) intake structure. The designations of these panels are:

1AC515
1CC515
1AC516
1CC516

The inspector verified that panels 1AC515 and 1CC515 manufactured by Royce Equipment Company, drawing ND-359-00, and panels 1AC516 and 1CC516 manufactured by Comsip, Inc. drawing 7374-4 conformed to the drawings. Panel mounting to the floor as well as device mounting within the panel were also verified. Accessible portions of the wiring in all panels were inspected for correct wire identification and adequate terminations. No defects were found. The inspector used the vendor supplied drawings for inspecting terminations made by the vendor and the site wiring Termination Document EC580 for field terminations. At the time of the inspection Low Level Transmitter 1EP-LDT-2225C had been removed for repairs per Startup Deviation Request EA-0506.

Seismic qualification of panels 1AC515 and 1AC516 was accomplished by testing. Testing of panel 1AC515 was performed by Wyle Laboratories and is reported in their report 58878. Testing of panel 1AC516 was done by Computech Engineering Services, Inc. and reported in their report 56301. Testing was performed in accordance with the applicable Bechtel specification as follows:

10855-G-011(Q) General Project Requirements for Seismic Qualification of Class 1E Control Devices and Instrumentation

10855-G-012(Q) General Project Requirements for Seismic Qualification of Class 1E Control Panel Assemblies

No violations were observed.

4.3 DC Power System

The inspector conducted a walkdown inspection of the 125 volt dc class 1E batteries and battery charger to verify conformance with FSAR and SER commitments. Verification also included confirmation that installation, construction and operational problems identified in previous inspections had been resolved.

4.3.1 Batteries and Battery Chargers

The inspector examined the four class 1E 125 volt dc batteries and battery chargers and verified that:

- The rooms were properly illuminated with lighting systems equipped with explosion proof fixtures.

- Battery and charger room doors were locked and the keys are controlled in accordance with approved administrative procedures.

- Rooms were equipped with temporary ventilation and cooling until the HVAC systems can be completed and turned over from construction to startup. The rooms ventilation system was not operating properly - construction activity was in progress.

- The battery rooms were monitored by an operable hydrogen detection system.

- Identification of batteries, chargers, cable, conduit, rooms and equipment are in accordance with approved drawings.

- Equipment and batteries are procured, received, inspected and installed in accordance with approved procedures.

- Equipment, batteries and rooms are clean.

- Items identified as unresolved on previous inspections have been resolved. There were no outstanding open items or construction deficiency reports.

4.3.2 Documents reviewed for this inspection include the following:

-- Technical Specification for Batteries, Spec. No. 10855-E-151 (Q), Rev. 5, March 15, 1984

-- Technical Specification for Battery Chargers, Spec. No. 10855-E-151 (Q), Rev. 6, September 12, 1984

-- C&D Stationary Battery Installation and Operating Instructions, 12-800, 1981

-- Drawing No. M-8004, "Battery Arrangement" C&D Batteries, Rev. 1, January 18, 1984

-- IEEE 450, "Maintenance, Testing and Replacement of Large Lead Storage Batteries", 1980

-- IEEE 380, "Standard Criteria for Class IE Power Systems", 1980

-- Inspection Record for "Installation of Batteries and Racks" QCIR No. IDD410-E-6.7

-- Inspection Record for "Installation of Electrical Equipment, Class 1E Channel D 1250 Battery Chargers" QCIR No. 1DD444-E-6.0

-- Quality Control Inspection Record, Job No. 10855 R-1.00, Rev. 13, "Battery Racks"

4.4 Findings

No deficiencies were observed in the 125 volt d-c battery systems inspected.

5.0 Instrumentation and Control Systems

5.1 General

The scope of inspection in the area of instrumentation and control (I&C) systems covered the following:

- Impulse lines
- Instruments
- Instrument cable, cable routing and terminations
- Control panels
- Switchgear and motor starter controls
- Control cable, cable routing and terminations
- Control functions
- Review previous identified I&C problems
- Investigate current identified I&C problems

The specific systems which were inspected in the I&C area included:

- Reactor Protection (RPS)
- Engineered Safety Features, Emergency Core Cooling System (ECCS)
- Residual Heat Removal (RHR)
- Safety Auxiliary Cooling System (SACS)
- Station Service Water System (SSWS)

The objective of this inspection was to verify, by sampling review, that the above systems were designed and installed to meet their intended safety function as specified in the Final Safety Analysis Report (FSAR) and the Safety Evaluation Report (SER). Further, the as-built systems were installed in conformance with controlled specifications, controlled drawings and implementation of the Quality Assurance program.

The criteria used during the as-built inspection are as follows:

Instrument Impulse Lines

The visual inspection during the walkdown of the instrument impulse lines included checks for the following technical requirements:

- protection of redundant channels was maintained by physical separation or barriers designed to withstand the specific hazard. In non-missile jet stream areas, the minimum separation between redundant instrument sensing lines was three feet in air in both the horizontal and vertical directions;
- the minimum slope requirement for Bechtel instruments was ¼ inch per foot and GE instruments ½ inch per foot;
- there was a minimum of two valves between the process tap and the instrument;
- isolation valves were located just beyond a penetration on the Zone 11 side of a shield wall;
- surface defects did not exceed 0.016 inch;
- there were no carbon steel deposits on stainless steel tubing from welding arcs;
- tubing, tubing restraints (guides) and anchors were located in accordance with the drawings and no tubing was located in walkways; and
- tubing minimum bend radius not less than 3 tube diameters.

Cable, Cable Terminations and Raceway

The visual inspection during the walkdown of the cables, cable terminations and the raceway included checks for the following technical requirements:

- safety related instrument and control cables were identified at each terminating end and at each 15 feet;
- there was no visual damage to the cables;
- the conductors were connected to the terminal point and terminal block as shown on the wire termination slip;
- the wire termination terminals were tight;
- the conductor terminations were in accordance with the licensee visual acceptance criteria;

- redundant cables and raceways were separated in accordance with the electrical installation specification;
- raceways were identified as required; and
- cables were installed in their respective raceways in accordance with the cable schedule.

Controls

The logic diagrams, schematic diagrams and field installations were reviewed to check for the following technical requirements:

- redundant components were properly identified;
- the functional requirements for the controls were achieved;
- resetting of a protective system actuation, at the system level, would not cause component action; and
- there was a system bypass status alarm.

The documents reviewed during the inspection are listed in Attachment 11. In addition, the applicable outstanding Startup Deviation Reports were reviewed.

5.2 Visual Inspection Details

The inspector performed the walkdown of the following safety systems and components using the visual criteria listed in paragraph 5.1.

5.2.1 Instrument Impulse Lines

Reactor Vessel Level Common ECCS & RPS

The impulse line, BB-1"-CCA-230, was visually inspected from the reactor vessel nozzle, N12B, elevation 165'-2", reference AZ 190 degrees, to the drywell penetration J-1350. This inspection continued from outside of the drywell, at J-1350, where downstream from the excess flow check valve the line changed to the tubing. The walkdown continued to the high side connections of level transmitter BB-LT-N091A which is channel A of the Emergency Core Cooling System (ECCS) logic input. This line is also connected to the high pressure side of level transmitter BB-LT-N080A which is channel W of the RPS system input to A1 trip channel. Both transmitters are located on instrument rack 10C004 in the reactor area 21 (north-west) at elevation 77 feet.

Reactor Vessel Level RPS

The impulse line, BB-1"-CCA-231, was visually inspected from the reactor vessel nozzle, N16B, elevation 145'-9", reference AZ 190 degrees to the drywell penetration, J1351. The inspection continued from outside the drywell at J1351 to the low pressure connection of level transmitter BB-LT-N080A.

Reactor Vessel Level ECCS

The impulse line, BB-1"-CCA-232, was visually inspected from outside the drywell at penetration, J1352 to the low pressure connection of level transmitter BB-LT-N091A.

Drywell Pressure Common RPS and ECCS

The impulse line, HCB-1"-054, was visually inspected from outside the drywell at penetration, J6A, to pressure transmitter, BB-PT-N094A, which is Channel A of the ECCS logic input. This line is also connected to pressure transmitter, BB-PT-N050A, which is Channel W of the RPS input to A1 trip channel.

5.2.2 Instrument Cables

Reactor Vessel Level RPS

The instrument cable for reactor vessel level transmitter, BB-LT-N080A, was visually inspected at the analog/digital panel, 10C609CW (GE H11-P609 Bay C). The cable at the transmitter end was not visually inspected because its termination is within an environmental barrier. It then enters a terminal box where the terminations were checked. The conduit leaving the terminal box was verified and visually inspected to where it passed through a barrier wall.

Reactor Vessel Level ECCS

The instrument cable for reactor vessel level transmitter, BB-LT-N091A, was visually inspected at the analog/digital panel, 10C617BA (GE H11-P617). The cable raceway was visually inspected from the instrument to where it passed through the first barrier wall.

Drywell Pressure RPS

The instrument cable for drywell pressure transmitter, BB-PT-N050A, was visually inspected at the analog/digital panel 10C609CW. The cable raceway was visually inspected from the instrument to where it passed through the first barrier wall.

Drywell Pressure ECCS

The instrument cable for drywell pressure transmitter BB-PT-94A, was visually inspected at the analog/digital panel 10C617BA. The cable raceway was visually inspected from the instrument to where it passed through the first barrier wall.

5.2.3 Control Cables

RHR Pump AP 202

The control cable, AP1Q0893D, was visually inspected from the switchgear breaker, 10A40106, to the first barrier wall. The other end of the cable was visually inspected from the solid state output panel, 1AC657BA (Div 1 RHR & CS RLY Vertical Board), to the first barrier wall. Similarly, control cable, AP1Q0893B, was visually inspected at the breaker, 10A40106, and panel 1AC657CA.

SACS Pump AP 210

The control cables, AP1C0301 B&D, were visually inspected from the switchgear breaker, 10A41004, to the first barrier wall. The other end of the cables was inspected from panel, 1AC657BA, to the first barrier wall.

Service Water Pump AP 502

The control cables, AP1C0205 B&D, were visually inspected from the switchgear breaker, 10A40109, to the first barrier wall. The other end of the cable was inspected from panel, 1AC657BA, to the first barrier wall.

5.2.4 Equipment

The following equipment was visually inspected to confirm their location, identification, and to verify the condition of instruments or control cables entering the electrical raceway system:

- Traveling Screens EP-AS501, CS501, BS501 & DS501.
- Traveling Screen Spray Booster Pumps EP-AP507, CP507, BP507. The DP507 pump was removed.
- Screen Level Instruments LE-2225A1, A2, C1, C2, B1 & B2, D1 & D2.
- River Level Instruments LE2220-1. LE2220-2 was removed.
- Service Water Pump Suction Level Transmitter LE-2241A, C, B&D.
- Service Water Strainers Differential Pressure Transmitters:
  - -- PDT2194A, C, B&D; PDT2195A, C, B&D; PDR2196A, C, B&D and PDT2197A, C, B&D
- Traveling Screen Spray Booster Pump Flow Switch EP-FS2225A, C, B&D
- Service Water Pump Area Heating and Ventilation Control Panels A & C.
- Service Water Pump Area Motor Control Centers 10B553, 63, 73 & 83.
- Traveling Screen and Wash System Control Panels AC, CC, BC & DC 515 and AC, CC, BC & DC 516.

5.3 Findings

The inspector found that the state of workmanship in the area was generally good and the instrumentation and control systems inspected conformed to the criteria of paragraph 5.1. However, as a result of the as-built inspection, the following specific findings were noted for which licensee corrective actions were in progress at the end of the inspection.

1. Following the visual inspection of the nuclear boiler instrument impulse lines, the inspector reviewed selected reactor water level instrument calibration data sheets and the documentation which provides the basis for these initial calibration settings.

During this review, the inspector noted that the initial calibrations were not based on the as-built elevations of the instrument lines. The failure to incorporate the as-built elevation data into the calibration calculations could result in the systematic miscalibration of the reactor vessel water level instruments.

The inspector informed the licensee that the adequacy of the reactor vessel water level instrumentation to perform its design functions would be considered unresolved until the following concerns are addressed: (354/85-58-04)

- Incorporation of the as-built elevation data into the reactor vessel level instrument calibration calculations; and

- Subsequent re-calibration of the affected instruments.

2. The inspector review of the draft Technical Specification, Section 4.8.4.4 Reactor Protection System Electrical Monitoring Surveillance Requirements, finds that the 132 VAC over-voltage setpoint may not protect the scram solenoids from a power supply over-voltage condition. The solenoids' electrical tolerance for operability is 115 volts plus or minus 10 percent. Thus, the over-voltage value is 127 volts not 132 volts. Neither value accounts for the voltage drop between the Electrical Protective Assembly and the furthest solenoid.

   The licensee has agreed to follow the recommendations contained in General Electric Spec Data Sheet MPL Item No. C71-4010, "Reactor Protection System" 22A3083AK Revision 6 for the setting of the Electrical Protective Assembly which accounts for the voltage drop. After these values are obtained, they will be used in the final Technical Specification.

3. The inspector noted the following during the visual inspection:

   - Five Instruments without identifying tags;

   - One loose wire termination within each of four control panels;

   - Three auxiliary relays within each of four control panels did not have a identifying name plate;

   - One equipment name plate on each of four motor control centers had incorrect information;

   - One equipment name plate missing from a motor control center;

   - One motor power conduit identification missing;

- One equipment name on P&ID differed from other identification used for the same equipment;

- One diesel generator room with the incorrect color identification; and

- Need for completion of plant program for labeling of pumps, piping and motor operated valves etc.

The licensee is taking action to assure proper identification of plant systems and components. The pump and valve portion of the last item is being addressed by the licensee in a Site Engineering Instruction (SEI) 3.7 Revision 0, "Plant Labeling Programs" dated April 19, 1985. The licensee has not specified when this program will be completed. The licensee should assure that completion of plant labeling receives continued attention.

4. The inspector noted that a cable tray fire stop had been partially opened. A discussion with the licensee and a review of procedure "Penetration Seal Review, Sign-Off, and Work Tracking" SWP/P-C120, Revision 6 indicated that all acceptance and modification of penetrations, including fire stops, are being controlled.

5. The inspector reviewed Startup Deviation Reports associated with damaged Rosemount transmitters resulting from the use of Neolube 100 thread sealant on each of two covers per instrument. Because of the potential damage to the pressure boundary when the covers were removed SDR ZC-0061 was issued, with procedure PSE-PR-E-006, Revision 0, December 11, 1985, "Pressure Leak Test for Rosemount 1153 Nuclear Transmitters."

   Thirty six transmitters are to be removed and replaced with new transmitters. Seventeen transmitters will have covers replaced. One transmitter will have its Conex EQ cable nipple replaced. One hundred and fifty five transmitters will be restored to the original installed conditions and recalibrated.

   The inspector visually walked down all transmitters at elevation 55 feet and 77 feet in the reactor building. As a result of this inspection, three additional SDRs were issued. These were for the following: one additional transmitter to be removed and replaced; one transmitter cover replaced and one Conex EQ cable replacement. The licensee was also advised that some of the temporary covers had become dislodged. The licensee should assure that before the recalibration takes place that the concerns of finding item 1 above are addressed.

No violations were observed during this inspection.

6.0 Civil/Structural

6.1 General

The scope of inspection in the civil/structural area included a review of the Building Verification Program and an inspection of the Control Rod Drive housing supports and the control area Chilled Water System equipment supports. The review also included an evaluation of the licensee's activities related to the implementation of the Visual Weld Acceptance Criteria (VWAC) for welds designed to the requirements of AWS D1.1 Code.

6.2 Building Steel Verification Program

The scope of inspection in this area focused on the as-built load verification program for Category I structural steel. The licensee's program for this activity was undertaken to verify the adequacy of the as-built structures since the initial design was based on estimated loads. In this verification process, evaluation of building structures is performed utilizing actual as-built loads induced by large bore piping and major equipment supports in addition to the support loads from bulk installations which include small bore piping supports, minor equipment, HVAC, conduit, cable tray, tubing, and other miscellaneous attachments.

The objective of the inspection of this activity was to provide an assessment of the licensee's program and to determine whether acceptable engineering practices, regulatory requirements and licensee commitments had been met.

The inspector performed a review of the design procedures which are used in the load verification program and conducted meetings with cognizant licensee and Bechtel engineers who are involved in carrying out this activity. Further, the inspector performed a review of some sample design evaluation packages performed for the qualification of selected structural members.

The load verification program addresses three major types of support attachments:

- Pipe support reaction loads as determined from the As-Built Reconciliation of piping systems. Evaluation of building steel is performed using the actual magnitudes of large bore reaction loads and actual location of attachments as indicated on the larger drawings.

- Bulk installations identified above are evaluated by performing walkdowns to review and record as-built conditions.

- Major equipment is evaluated based on final loads provided by vendors and installation locations as verified by walkdowns.

The inspector determined that the load verification of bulk attachments require considerable judgement from engineering personnel performing the evaluation since it involves an assessment of building structures on the basis of an evaluation of attachment locations from above and below the floor (via walkdowns) and determining reaction loads using simplified calculations rather than actual as-built final loads (as in the case of large bore piping attachments). Effects of computed bulk loads on structural members are assessed against the original assumed design uniform floor loads (lbs. per sq. ft.) to verify the design adequacy of these members.

Initial design of structural beams typically includes a minimum of 50 lbs. per sq. ft. floor load to account for all bulk attachments. Based on the review and assessment of bulk installations, if a floor contains attachments which exceed the assumed design uniform load, the most heavily loaded areas of the floor are selected and the attachment loads which are tributary to the most heavily loaded beams are determined. Calculations are performed to verify the adequacy of these identified beams. Detailed calculations of structural adequacy are also performed for structural beams when changes in loads occur (as in large piping attachments and major equipment). Thus, the adequacy of some beams is determined by their similarity with other beams for which detailed calculations are performed.

The licensee indicated that of the approximately 800 areas reviewed, only 10 cases were found where the 50 psf floor load was exceeded. In all cases the structural steel beams were determined to be adequate. Further, the licensee indicated that the installation of most commodities other than conduit and small piping, were complete at the time of the structure load verification walkdowns.

The walkdown is typically performed by a team of engineers consisting of an originator and a checker. A check sheet is prepared for each area which documents the walkdown results. Rooms on both sides of a common boundary (wall or slab) are walked down in order to determine the total attachment loads on the wall or slab.

Many of the observations which were noted during this review had been already addressed in the Independent Design Verification (IDVP) report. Further specific evaluations which were performed by Bechtel in response to the IDVP findings were found to be generally acceptable.

6.3 Installation of the Control Rod Drive (CRD) Housing Supports

The function of the control rod drive housing supports is to prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

The CRD housing supports consist of horizontal beams installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings. The beams are welded to brackets that are welded to the steel liner of the reactor support pedestal. Hanger rods are supported from the beams on stacks of disc springs. Support bars are bolted between the bottom ends of the hanger rods. Individual grids rest on the support bars between adjacent beams. Each grid assembly is made from two grid plates, a clamp and a bolt. The top part of the clamp guides the grid to its correct position directly below the CRD housing. With the support bars and grids properly installed, a gap of slightly more than one inch exists between the grid assembly and the bottom surface of the CRD housing flange.

A visual inspection was performed of the support bars and grids to verify proper assembly. Measurements were made, on a sampling basis, to insure that adequate spacing existed between the grid assemblies and CRD housing flanges.

This area was found to be satisfactory.

6.4 Component/Equipment Supports

Control Area Chilled Water System (CWS)

The control area chilled water system components were considered for as-built verification of their supports due to the following:

- The inspection of the CWS provides a continuation to the inspection of the Safety Auxiliary Cooling System (SACS) since the CWS is cooled by the SACS
- The system has a significant safety function since it provides chilled water to maintain satisfactory ambient air temperatures in the following areas: main control room, auxiliary equipment room including computer room and battery rooms, emergency switchgear rooms, SACS pump rooms, and class 1E panel rooms.

The control room area CWS consists of two subsystems: the main control room chillers and the class 1E panel room chillers. Five major components (from both subsystems) were selected for as-built verification of their support and foundation.

The selected components are:

- Control room A/C unit 1AVH-403
- Chilled water circulating pump 1AP-400
- Control equipment room A/C unit 1AVH-407
- Chilled water chemical feed tank 1AT-401
- Chilled water head tank 1A410

The inspection attributes for the above equipment supports included:

- verification of as-built support or foundation configuration dimensions.
- verification of hold-down anchor bolt sizes, location and tightness
- identification of cracks in the concrete foundations
- visual inspection of welded joints

6.5 Visual Weld Acceptance Criteria (VWAC)

The Nuclear Construction Issues Group (NCIG) document NCIG-01 provides alternate visual weld acceptance criteria (VWAC) for structural welding conducted to AWS D1.1 requirements. This document has been endorsed by NRR with the stipulations that the licensee obtain an FSAR change, conduct adequate training in the interpretation of the document, and assure that the applicability of the NCIG-01 is acceptable to the cognizant engineer.

The inspector reviewed the licensee's VWAC inspection activities. FSAR change Notice 985 addresses the request for the use of NCIG-01. The inspector noted that the use of NCIG-01 will be limited to welding conducted under Bechtel specification C-130Q as amended to add the VWAC criteria in Appendix "D". The training program consisted of a 3 hour lecture (with specially prepared samples) which was given to more than 250 people representing FQC's and FWE's. The FWE's were required to take a written test on the VWAC criteria. In a previous inspection a regionally based inspector attended a typical training program conducted on September 23, 1985. The engineering control and the scope of usage of the VWAC criteria is evidenced by the restriction of its use to the C130Q specification. Full implementation of the VWAC document commenced September 26, 1985 in accordance with the N.D. Griffin (Bechtel) memo FE-1955 dated September 20, 1985.

6.6 Finding and Conclusion

The inspectors concluded that the building structure as-built verification program had met the intent for which it was established. Though some questions were raised regarding the degree to which engineering judgement was used in carrying out the walkdown verification and evaluation of bulk attachments, nevertheless, the cognizant engineering person interviewed by the inspector was found to be knowledgeable in performing the required activity. Further assurance regarding the completeness of this activity was derived from the review of result of the IDVP comprehensive evaluation in this area.

Components and equipment supports and foundations verified during the inspection were found to be in conformance with the installation drawings. No items of noncompliance were identified.

7.0 As-Built Verification of Equipment for Selected Emergency Operating Procedures

7.1 General

The scope of this phase of the inspection was to examine the installation of selected portions of safety related systems that would be used during implementation of the plant specific Emergency Operating Procedures. Portions of the following systems were included in this area of the inspection:

- Condensate Storage and Transfer System
- Service Water System
- High Pressure Coolant Injection System
- Reactor Core Insolation Cooling System
- Nuclear Steam Supply Shutoff System
- Reactor Protection System

The objective of this inspection was to verify that the as-built configurations were in conformance with the FSAR, the SER and system specifications and drawings and that they were capable of performing their intended functions as specified in the FSAR and in the Emergency Operating Procedures.

7.2 Alternate Reactor Cooling Water Sources

The Emergency Operating Procedures identify three alternate water sources that may be used in the extremely unlikely event that both normal and emergency core cooling systems are unavailable. These three sources are: (1) Condensate Storage and Transfer System; (2) Service Water System and (3) Fire Water System.

7.2.1 Condensate Storage and Transfer System

Accessible portions of the system were visually inspected from the Condensate Storage Tank (CST) in the yard to the residual heat removal and core spray flushing connections in the reactor building.

7.2.2 Service Water System

The Service Water System was visually inspected from the supply header (reactor building elevation 77') to the intertie connection with Loop B of the residual heat removal system. In addition a visual inspection was performed of the system from the fire hose fill connection (Auxiliary Building Elevation 77') on service water Loop B to the residual heat removal system Loop B intertie.

7.2.3 Fire Water System

The ability to connect the fire water system (via fire hose) to the Loop B service water fill connection was verified.

7.3 Suppression Chamber Level Control

The Emergency Operating Procedures identify four systems that may be employed for emergency makeup to the suppression chamber and three systems that can be used for emergency drawdown. Two of these systems, High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC), were selected for inspection since they could be used in both modes.

7.3.1 Emergency Makeup

When employed for suppression chamber makeup both the HPCI and RCIC systems would be aligned to take suction from the CST and would discharge to the suppression chamber via their separate minimum flow lines. A visual inspection was made of both systems from their separate connections to the condensate storage and transfer system near the pump suctions (Reactor Building Elevation 54') to their minimum flow return lines to the suppression chamber.

7.3.2 Emergency Drawdown

When used for suppression chamber drawdown both the HPCI and RCIC systems would be aligned to take suction from the suppression chamber and discharge to the CST via a common return line at valve BJ-HV-F011. This alignment requires that valve interlocks on BJ-HV-F011, which wou'd normally prevent opening if either system's suppression chamber suction valve were open, be defeated. A visual inspection was made of both systems from their separate connections to the suppression chamber near the pump suctions (Reactor Building Elevation 54') to the common return line to the CST at valve BJ-HV-F011 (Reactor Building Elevation 77'). In addition, portions of panels H11-620 and H11-P621 (Auxiliary Building Elevation 102') were inspected to verify that the as-built wiring would support the intended bypassing of the valve interlocks on BJ-HV-F011.

7.4 Bypassing Main Steam Isolation Valve (MSIV) Interlocks

During certain degraded modes of operation the Emergency Operation Procedures direct the re-opening of the MSIVs to aid in reactor pressure control and to reduce the heat load on the containment. To accomplish this task it would be necessary to equalize pressure around the inboard MSIVs and, in certain instances, defeat the MSIV isolation interlock on low water level in the reactor vessel (L1:-129 inches). A visual inspection was made of the accessible portions of the main steam equalizing lines from inboard of the MSIVs in the drywell via the BB-HV-F016, BB-HV-F019 and BB-HV-F020 valves to the main steam lines in the steam tunnel. In addition, bays A and C in panel H11-P609 and bays B and D in panel H11-P611 (main control room) were inspected to verify that the as-built wiring would support the intended bypassing of the L1 MSIV isolation interlock.

7.5 Scram Solenoid De-energization

In the extremely unlikely event that some (or all) scram pilot valve solenoids should fail to de-energize when required by the reactor protection system the Emergency Operating Procedures direct actions to manually remove power from these solenoids. The method chosen requires that eight fuses be removed to achieve a complete de-energization of all solenoids. An inspection was made of the as-built wiring in bays A and F of both panels H11-P609 and H11-P611 (main control room) to verify that, with the as-built wiring, the removal of the indicated fuses would, in fact, produce the desired de-energization.

7.6 Findings and Conclusions

The inspection in this area demonstrated that the systems examined were constructed in accordance with the descriptions in the FSAR and system specifications and drawings. The portions of systems inspected were found to be capable of performing their intended functions as described in the FSAR and as required by the Emergency Operation Procedures.

No discrepancies were observed.

8.0 Comparison of FSAR Accident Analysis Descriptions to As-Built Plant

8.1 General

The objective of this phase of the inspection was to insure that design changes made to the facility during construction were being properly incorporated into the accident analysis of the FSAR. A review was made of sections 1 through 4 of Chapter 15 (Accident Analyses) of the FSAR to identify any assumptions or inputs into the accident analysis which were in conflict with the as-built plant.

The inspector identified three instances in which the Chapter 15 discussions failed to reflect the as-built plant. These items were discussed with the licensee and resolved as indicated below.

8.2 Reactor Recirculation Automatic Flow Control

The automatic flow control mode of the reactor recirculation system is a non-safety related control system which would provide the plant with limited load following capabilities. While the licensee has elected to defeat this control feature, several sections of the Chapter 15 analysis still contain discussions of the plant response to transients while operating in this mode. The licensee indicated that the deletion of this control mode was a recent change and provided the inspector with the change notice that was in process to update the FSAR. Following review of the change notice the inspector was satisfied that this change was being properly addressed.

8.3 Residual Heat Removal Steam Condensing Mode

The steam condensing mode of the residual heat removal system is a non-safety related mode which would provide an alternative means of decay heat removal. The licensee has elected to defeat this operating mode. However, in the analysis of the loss of feedwater flow transient, the use of this mode is indicated as part of the operator actions in response to the transient. The inspector discussed this discrepancy with the licensee. The licensee indicated that this item had been recently identified and provided the inspector with a copy of the applicable change notice which was in process to update the FSAR.

8.4 Main Steam Line Isolation on Low Reactor Water Level

The inspector identified an internal inconsistency in Chapter 15 involving the reactor water level setpoint which would cause a full main steam line isolation. Most analyses indicated that a full isolation would occur at a reactor low water level of -129 inches (L1). However, in three cases (Generator Load Rejection, Reactor Recirculation Pump Trip and Recirculation Flow Control Failure with Increasing Flow), the analyses indicates that a full isolation would occur at a reactor low water level of -38 inches (L2). Discussion with the licensee indicated that the correct setpoint for a full main steam line isolation is -129 inches (L1). The licensee agreed that the three cases identified by the inspector were in error and committed to revising those sections to reflect the correct setpoint. The inspector noted that the use of the L2 setpoint for full main steam line isolation was conservative in all three cases and provided results which bound the actual plant response.

8.5 Findings and Conclusions

The inspection in this area demonstrated that accident analysis of Chapter 15 of the FSAR, including pending Change Notices, is in substantial agreement with the as-built facility. Also, the licensee's review program provides reasonable assurance that design changes will be evaluated for potential impact on the FSAR accident analyses.

In response to the inspector's concern that long time delays may exist between the approval of a design change and the updating of the FSAR, the licensee briefed the inspector on a recently instituted program to accelerate updating of the FSAR. The program provides for a significant reduction in turn around time for incorporation of field changes into the FSAR. In addition, the program will review all NSSS design changes made to date against the FSAR to insure it accurately reflects the as-built facility.

No violations were identified.

9.0 Independent Verifications

9.1 Motor Operated Valve Operability

The RHR low pressure system injection (LPSI) motor operated valve (MOV), 1-BC-HV-F017A, was selected by the inspector to verify control operability during a degraded grid voltage condition coincident with a loss of coolant accident (LOCA) condition.

The 4.16 KV Class 1E bus A401 supplies power through a load center transformer where the 480 volt side in turn supplies power the motor control center (MCC) 10B212. The MOV is controlled and supplied power from this MCC.

- One equipment name on P&ID differed from other identification used for the same equipment;

- One diesel generator room with the incorrect color identification; and

- Plant wide lack of labeling of pumps, motor operated valves etc.

With the exception of the last item, the licensee took prompt corrective action to assure proper identification of components or areas. The last item is being addressed by the licensee in a Site Engineering Instruction (SEI) 3.7 Revision 0, "Plant Labeling Programs" dated April 19, 1985. A purchase specification is being prepared for plant labels. The licensee has not specified when this program would be completed.

4. The inspector noted that a cable tray fire stop had been partially opened. A discussion with the licensee and a review of procedure "Penetration Seal Review, Sign-Off, and Work Tracking" SWP/P-C120, Revision 6 indicated that all acceptance and modification of penetrations, including fire stops, are being controlled.

5. The inspector reviewed Startup Deviation Reports associated with damaged Rosemount transmitters resulting from the use of Neolube 100 thread sealant on each of two covers per instrument. Because of the potential damage to the pressure boundary when the covers were removed SDR ZC-0061 was issued, with procedure PSE-PR-E-006, Revision 0, December 11, 1985, "Pressure Leak Test for Rosemount 1153 Nuclear Transmitters."

   Thirty six transmitters are to be removed and replaced with new transmitters. Seventeen transmitters will have covers replaced. One transmitter will have its Conex EQ cable nipple replaced. One hundred and fifty five transmitters will be restored to the original installed conditions and recalibrated.

   The inspector visually walked down all transmitters at elevation 55 feet and 77 feet in the reactor building. As a result of this inspection, three additional SDRs were issued. These were for the following: one additional transmitter to be removed and replaced; one transmitter cover replaced and one Conex EQ cable replacement. The licensee was also advised that some of the temporary covers had become dislodged. The licensee should assure that before the recalibration takes place that the concerns of finding item 1 above are addressed.

   No violations were observed during this inspection.

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- One diesel generator room with the incorrect color identification; and

- Plant wide lack of labeling of pumps, motor operated valves etc.

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   No violations were observed during this inspection.

During this review, the inspector noted that the initial calibrations were not based on the as-built elevations of the instrument lines. The failure to incorporate the as-built elevation data into the calibration calculations could result in the systematic miscalibration of the reactor vessel water level instruments.

The inspector informed the licensee that the adequacy of the reactor vessel water level instrumentation to perform its design functions would be considered unresolved until the following concerns are addressed: (354/85-58-04)

- Incorporation of the as-built elevation data into the reactor vessel level instrument calibration calculations; and

- Subsequent re-calibration of the affected instruments.

2. The inspector review of the draft Technical Specification, Section 4.8.4.4 Reactor Protection System Electrical Monitoring Surveillance Requirements, finds that the 132 VAC over-voltage setpoint may not protect the scram solenoids from a power supply over-voltage condition. The solenoids' electrical tolerance for operability is 115 volts plus or minus 10 percent. Thus, the over-voltage value is 127 volts not 132 volts. Neither value accounts for the voltage drop between the Electrical Protective Assembly and the furthest solenoid.

   The licensee has agreed to follow the recommendations contained in General Electric Spec Data Sheet MPL Item No. C71-4010, "Reactor Protection System" 22A3083AK Revision 6 for the setting of the Electrical Protective Assembly which accounts for the voltage drop. After these values are obtained, they will be used in the final Technical Specification.

3. The inspector noted the following during the visual inspection:

   - Five Instruments without identifying tags;

   - One loose wire termination within each of four control panels;

   - Three auxiliary relays within each of four control panels did not have a identifying name plate;

   - One equipment name plate on each of four motor control centers had incorrect information;

   - One equipment name plate missing from a motor control center;

   - One motor power conduit identification missing;

- Traveling Screens EP-AS501, CS501, BS501 & DS501.
- Traveling Screen Spray Booster Pumps EP-AP507, CP507, BP507. The DP507 pump was removed.
- Screen Level Instruments LE-2225A1, A2, C1, C2, B1 & B2, D1 & D2.
- River Level Instruments LE2220-1.LE2220-2 was removed.
- Service Water Pump Suction Level Transmitter LE-2241A, C, B&D.
- Service Water Strainers Differential Pressure Transmitters:
  - -- PDT2194A, C, B&D; PDT2195A, C, B&D; PDR2196A, C, B&D and PDT2197A, C, B&D
- Traveling Screen Spray Booster Pump Flow Switch EP-FS2225A, C, B&D
- Service Water Pump Area Heating and Ventilation Control Panels A & C.
- Service Water Pump Area Motor Control Centers 10B553, 63, 73 & 83.
- Traveling Screen and Wash System Control Panels AC, CC, BC & DC 515 and AC, CC, BC & DC 516.

5.3 Findings

The inspector found that the state of workmanship in the area was generally good and the instrumentation and control systems inspected conformed to the criteria of paragraph 5.1. However, as a result of the as-built inspection, the following specific findings were noted for which licensee corrective actions were in progress at the end of the inspection.

1. Following the visual inspection of the nuclear boiler instrument impulse lines, the inspector reviewed selected reactor water level instrument calibration data sheets and the documentation which provides the basis for these initial calibration settings.

The discrepancies found in the system as-built configuration are being corrected under various licensee programs. The number of items found is relatively small considering the depth and breath of the as-built walkdown verification. The nonconforming conditions found would not significantly impact upon the safe operation of the systems.

7. The inspector identified that SDV vent valves V776 and V777 had been mistagged as V774 and V775 respectively. The inspector was provided Startup Deficiency Report (SDR) BF-270 that documented the North and South vent valve tags had been mistakenly exchanged. The inspector was informed that the tags were reaffixed to the proper valves. The licensee stated that the system P&ID had been used to tag the valves in accordance with procedure SEI 7.4. The scope of procedure SEI 7.4 is limited to instrument root valves and skid mounted valves. Pending licensee review of the tagging program procedural controls, this item is unresolved. (354/85-58-03)

8. The inspector examined small bore support 1-P-BF-435-H3 and found two cases of underlength fillet welds wherein the design specified end returns had not been provided. The inspector reviewed the associated support calculation and ascertained that the end return weldments would not be required to ensure the support load carrying capacity.

   Field Change Request P-16162 was issued to provide engineering criteria to inspect the end return welds. Quality Action Request F310 was issued to ensure training of appropriate personnel regarding end return weldments. The inspector reviewed structural, electrical, and other pipe support design drawings and found in all cases that weld length was specified. The inspector had no further questions.

4.0 Electrical Systems

4.1 General

The objective of this phase of the inspection was to examine the installation of selected portions of the Class 1E ac and dc power systems and to verify that the as-built conditions agree with FSAR and SER descriptions and project specifications and drawing requirements. The portion of the ac system selected for inspection were those associated with the "A Train" station service water system, RHR system, and the SACS system. In the dc power system, the batteries and battery chargers were examined.

6. The walkdown inspection of the CWS, SSWS and SACS piping components and hangers disclosed the following discrepancies between licensee design documents and existing field installations. The nonconformance reports (NCR's) referenced below were issued during the inspection to resolve each associated issue.

   (i) The existing clearance gap between large bore service water piping and rigid restraint 1-P-EA-026-H02 was found to be .049 inches on one side. Pipe support drawing 1-P-EA-026-H02(Q) Rev. 4, FO specifies 1/16 inch gap on both sides. Specification P-410 specified tolerances with a combined total clearance of 3/32 inches minimum. NCR No. 8875 was written to document the actual gap.

   (ii) SACS isometric drawing 1-P-EG-06 shows valve 1-EA-V804 stem in the horizontal position. The existing valve stem is approximately 45° from the horizontal position. This orientation was found to facilitate installation to avoid handwheel operational interferences and to have minimal impact on system stresses and restraint loading. NCR No. 8891 and FCR-P-16150 were issued and approved to document the as-found orientation.

   (iii) A poorly designed, non-functional (unstable) spring can hanger was found installed on a 1 inch diameter fuel pool make-up line. FCR No. PF-12046 was issued to redesign the hanger which was reinstalled and found to be acceptable.

   (iv) SACS pipe support drawing 1-P-EG-107-H06(Q) specified a 5/16" fillet weld where the existing fillet weld measured ¼". NCR No. 8886 was issued to document the nonconforming condition.

   (v) Hanger 1-P-EG-159-H01 clamp was observed by the inspector to be in contact with nearby support steel during SACS operational testing. The hanger was reinspected under non-operating conditions and a .035" clearance was measured. This gap and the hanger drawing design movement is determined as acceptable for QC inspections. Yet, to enhance the functional operation of the spring support, PSE&G will relocate the clamp within the tolerance specified in specification P4.10.

   (vi) Anchor bolt elevation and top of floor elevation for chilled water system tank 1AT401 on drawing C-0399-0 Sheet 294 appeared to be in conflict. This discrepancy was attributed to a drafting error and is to be corrected via a Field Change Notice.

During a degraded grid voltage condition, the bus A401 normal supply breaker is tripped at 92% bus voltage. Under those conditions, the bus would be reenergized from the standby emergency diesel generator associated with this bus.

A study "Millstone Voltage-1E Buses" Calc. No. 15.1, Revision 2 established the 92% trip setting for all 4.16 KV Class 1E busses. This study also provided the voltage condition at MCC, 10B222, which is the redundant MCC to 10B212. The MCC 10B222 has a longer cable length from the loadcenter than MCC 10B212, therefore, the low voltage at the MCC for MOV F017A is conservative. This value is 86.21% of 480 volts which is equal to 413 volts. The voltage on the secondary side of the control transformer is 413 divided by the turns ratio of 3.804 which is equal to 106 volts. This value is representative and was used in the study "Control Transformer Selection and Maximum Circuit Wire Lengths for MCC Control Circuits" Calc. No. 17A, Revision 1.

The method of calculation is to add vectors of the control wires and control transformer series impedances to solve for the voltage at the contactor coil. This voltage would then be compared to the minimum pickup voltage specified by the vendor. The inspector's independent calculations are contained in Attachment 1. The inspector concluded that this MOV will function during a degraded low voltage grid condition.

9.2 Field Measurements of Piping and Pipe Support As-Builts

The inspector used a tape measure and fillet gages to independently verify piping and pipe support measurements on the Residual Heat Removal, Scram Discharge Volume, Service Water, and Safety Auxiliary Cooling Systems. The verified measurements included:

- linear pipe run dimensions
- pipe support locations and unsupported pipe span lengths
- Mechanical component locations
- pipe support member size
- concrete expansion bolt size
- pipe support weld size and length

With minor exceptions as discussed in Section 3.4 of this report, the independent measurements were in correlation with licensee design and as-built documentation.

9.3 Independent Evaluation of Available Voltage at Selected Loads

The inspector selected the train A station service water pump and motor control center in the station service water building as representative safety loads to determine that the voltage available would be adequate under worst case degraded grid voltage conditions for starting and running the motors in this location. In conducting this evaluation, the inspector reviewed the following:

- Cable type, sizes, length and impedances
- Circuit breaker type, size and ratings
- Pump motor size, starting and running currents
- Motor control center starting and running currents
- Worst case voltage conditions at the emergency busses
- Hope Creek (Millstone) Voltage Study Calculation 15.1(Q) - 1E Buses, Revision 2, dated 10/3/85
- Safety Evaluation Report NUREG-1048, Section 8.3.1.1 "Voltage Drop Analysis"

The inspector reviewed the Bechtel voltage study including voltage profiles at the various 1E safety buses under various and worst case type conditions including conditions of degraded grid voltage. In addition, the inspector reviewed staff conclusions made in SER NUREG-1048 Section 8.3.1.1 related to the fact that there is reasonable assurance that all class 1E loads will operate at or within design voltage limits under all conditions of plant operation.

The inspector also conducted a walkdown of the 4160 volt and 480 volt power cable runs from the emergency switchboard and unit substation to their respective service water pump and motor control center loads to verify circuit breaker adequacy; cable type, size, support, spacing, routing, marking, and lengths. The inspector compared cable pull cards and termination records to the actual installation. No discrepancies were discovered.

Using Okonite Company Cable Technical Bulletin EHB-78, the inspector performed independent voltage calculations as shown in Attachment 2 to determine the voltage drops in the power feeder to the Station Service Water Pump "A". Cable impedances, motor starting and running currents and the calculated voltage drops were found to be consistent with the data reported in the referenced Bechtel "Millstone Voltage Study." Voltage drops calculated provide assurance of adequate voltage for starting and running these loads within the 80 percent minimum motor voltage requirement of Section 8.3 of the FSAR.

9.4 Independent Evaluation of Cable Pulling Tension for Power Cable

The inspector selected the power cable for the train A station service water pump as a representative cable to perform an independent evaluation of cable pulling tension calculations for comparisons to construction calculations and to actual tension measured during the cable pull. The calculation and comparisons are made to assure that the cable pulling tension calculations and methods used are adequate to ensure protection of the cables during installation. In conducting the evaluation, the inspector reviewed the following:

- Bechtel Power Corporation "Users Manual ECG-102 Cable Pulling Calculations Using a Programmable Calculator, Horizontal and Vertical Pulls, Book Number Two"

- General Electric Technical Handbook "Wire and Cable Selection, Section 8C1 Cable Installation Data"

- Okonite "Bulletin EHB-78 Engineering Data for Copper and Aluminum Conductor Electrical Cables"

- IEEE Standard 422-1977 IEEE Guide for the Design and Installation of Cable Systems in Power Generating Stations

- Bechtel Drawing 10855-E-1449-0, Sheet 31A "Cable Pulling Notes and Diagrams"

- Bechtel Drawing 10855-E-1000-0, Sheet 1 of 9 "Electrical Cable Description"

In making the cable pulling tension calculations, the inspector determined the following:

- The service water pump 1AP502 power cable is Okonite 5KV, #4/0, cable code A04, 3 single conductors. Each conductor is 1.219 inches OD, 1.161 pounds per foot, and has a minimum bend radius of 14.6 inches and a maximum pulling tension of 1,693 pounds. The cable is standard copper with a tinned copper tape shielding and with an overall hypalon insulating jacket.

- Okonite specifies a maximum cable sidewall pressure of 500 pounds per conductor per foot of bend radius for pulling the cable to preclude cable damage. The minimum bend radius for the cables is 3 feet which provides the most restricting sidewall pressure limitation for this cable pull of 3 x 500 = 1500 pounds, x 3 cables = 4500 pounds which is restricted to 2/3 of this value or 3000 pounds due to the fact one of the cables may try to ride the other two during the pull.

- For pulling the cable Okonite recommends the use of a pulling lubricant compound indicating that for the hypalon jacket a lube made by Utility Industries maybe used. The cable was lubricated during the pull.

- The cable pull card shows that 810 feet of the cable was pulled from the service water pump into a buried concrete duct through manhole 15 AMOD01A to manhole 15 AMOD01 just outside the diesel generator building where it was spliced with 200 feet of cable pulled from the emergency switchgear room to make an overall length of 1010 feet for this power cable.

- This cable is identified as cable number AC10205A.

- The coefficient of friction used for this pull calculation through the duct bend for the three single conductors pulled at one time and properly lubricated is 0.5 (Bechtel used a value of 0.4).

- The cable pull routing description horizontally and vertically including straight run lengths and angular turns are described on Drawing E-1449-0 Sheet 31A. This description forms one of the basis used in this calculations, except that this calculation assumes that the cable is all in the horizontal plane (the actual difference in elevation only varies from 89.83 feet to 94.7 feet).

- The allowable maximum pull tension for the three cables of 3X1693 lbs = 5079 lbs is reduced by 1/3 to 3388 pounds since the cables are pulled from separate reels in parallel into the same conduit and one cable may ride the other two during the pull.

The inspector conducted a walkdown of the cable run from the service water pump out of the service water building and followed the duct bank routing to the manholes and into the diesel generator building and then to the emergency 1E switchgear breaker cubicle. The cable lengths from the cable pull cards were compared to the length estimated during the walkdown. The actual length on the pull card appears to be correct.

Using the formulas and tables in General Electric Technical Handbook for Wire and Cable Installation Data Section 8C1, the inspector calculated the expected tension for pulling this cable. These calculations are shown in Attachment 2. The calculated value obtained was compared to the value calculated by Bechtel as shown on the cable pull card. The value calculated by the inspector was 2315 pounds as compared to a value calculated by Bechtel of 2059 pounds. The actual pulling tension measured by dynamometer and shown on the cable pull card was

2078 pounds. The inspector finds that the cable pull tension calculation formula and methods used by Bechtel to calculate expected cable pulling tensions to be satisfactory and finds no reason to question the calculations.

No deficiencies were discovered.

10.0 Quality Assurance Program Inspections

10.1 QC Inspection Records

The inspector examined QC inspection reports associated with:

- pipe supports;
- piping installation;
- mechanical equipment installation;
- structural steel erection; and
- Nondestructive Weld examinations.

The records were found to specify the requisite information regarding the item inspected; reference documents utilized during the inspection; Quality Control Inspection Report (QCIR) number; inspector identification; accept reject notation; and report review.

The inspector had no further questions.

10.2 Quality Control Instructions

Quality Control Instructions (QCI) are written to provide inspection checklists consisting of surveillance and mandatory holdpoints. The QCIs document inspection attributes contained within engineering specifications. The QCIs include generic QCIR forms that capture the requisite inspection attributes. The QCIs are originally written by home office quality staff and can be subsequently revised on-site.

The inspector reviewed QCIs associated with the following activities:

- Structural Steel Erection;
- HVAC Ductwork;
- Piping Fabrication;
- Piping and Pipe Supports Final Inspection.

These QCIs contained appropriate inspection criteria for the associated activities.

The inspector had no further questions.

10.3 QA/QC Interface in Building Load Verification Program

The inspector determined that the only QA interface in this activity was conducted as part of an audit (Report No. NH-85-026) performed by the licensee at the Hope Creek site during the week of August 5, 1985, and at the San Francisco Bechtel Home Office (SFHO) during the week of August 12, 1985. The audit included verification of controls associated with preparation of piping As-Built Reconciliation packages, review of pipe support calculation and distribution of ABR required information to layout pipe support and stress group. The applicability of this audit to the building verification program is limited to the verification of attachment reaction loads from large bore piping.

The licensee also identified that an engineering audit was being conducted by PSE&G staff at SFHO, during the NRC inspection of Hope Creek. The audit was to address the building verification program and to verify that all building steel supporting piping, equipment and other bulk installations have been qualified by documented calculations.

The inspector had no further questions.

10.4 Document Control

During this inspection, some time was allocated to reviewing the adequacy of administrative controls associated with preservation of the as-built plant conditions. As construction nears completion, a two step turnover of plant systems is in process. The first step is that as systems are completed, the Contractor (Bechtel) turns the system over the Licensee Startup Group. When the Startup Group has completed testing, control of the system is transferred to the Operations Group. During this period, work controlling documents may be issued by any one of these groups depending on the system status.

The inspector reviewed the operations of two document control centers used by the Startup Group. These are controlled by the Site Engineering Section. The first area reviewed was Document Control - Test. The functions in this area were:

- Test Engineers request document packages for specific tests to be performed. These requests include specific procedures, forms, drawings, etc. required for the test.

- Document Control - Test assemblies the package and issues it to the Test Engineer.

- Upon completion of testing, the Test Engineer returns the package for distribution and filing.

The inspector reviewed several packages and found them complete. The inspector asked to see the package for the test that identified defective transmitter 1EP-LDT-2225A reported on NCR 5902. The information available on the NRC did not match the filing designators but the package was readily retrieved. In this package, one of the work controlling documents was General Test Procedure (GTP)-2, Revision 2. The inspector determined that the revision of GTP-2 as of 12/10/85 was Revision 5, but at the time of the test, October 1984, the Revision 2 was correct.

The inspector reviewed the Technical Document Room (TDR) located in the administration area of the plant operations section. The TDR is the distribution point for documents to be used for work on systems turned over to operations. Hard copies of vendor manuals, aperture cards of drawings and microfiche of other documents are available in the TDR. Facilities are available for making hard copies from the microfiche and aperture cards but not for duplicating the microfiche or aperture cards. Access to the computer system for verifying revisions is available. When documents are issued they are either stamped "For Information Only" or "Working Copy User Responsible for Confirming Validity for Field Use. Issue date ___________. This document cannot be used in the field after the next revision or 7 days after the issue date." Only approved documents received from the Site Engineering Document Center or Change Authorizing Documents (CAD) received from the Bechtel Document Control Center are available in the TDR. To determine if the system for updating vendor manuals was adequate, the inspector randomly chose several manuals and noted the revisions to various pages of these manuals. He then witnessed the verification against the computer data and subsequently reviewed the same revisions in the master file kept in the Site Engineering Document Center. All of these references agreed with the revisions chosen.

The inspector reviewed the operations of the Site Engineering Document Control Center. This center is the primary distribution center for vendor manuals and documents generated by the Site Engineering organization. Vendor manuals are being received from Bechtel, San Francisco as well as the licensee purchasing organization. As they are received they are stamped, duplicated and the computer database updated. Control of the original manuals is accomplished by maintaining them in a locked cage. The original manuals are used solely for the source of controlled copies kept in the TDR or for Engineering reference.

No violations were observed.

10.5 Review of Nonconformance Reports

In accordance with the Bechtel Quality Program nonconformance reports (NCR) are used only on "Q", "F" and seismic systems. With few exceptions, NCR's are written only when deficiencies are found during final inspection. Deficiencies found during work in process are reported on one of several documents including Field Change Request, Field Change Notice, Supplier Deviation Disposition Request and Project Change Request.

From project start to December 10, 1985, there have been approximately 8860 NCRs written. Of these, approximately 2825 have been written in 1985. This increase was caused by the large number of final inspections being performed as construction nears completion.

The inspector reviewed the NCR log and chose eight NCR's written during 1985 to determine the adequacy of the disposition. For three of these eight, the inspector verified that the work described was done and was acceptable. These were:

NCR Number	Subject	Disposition
5902	Level transmitter 1EO-LDT-2225A was found defective during pre-system turnover testing.	The inspector verified the level transmitter has been repaired (by the Vendor) and replaced.
8086	Excess material removed from pressure tight door sealing surface (arc strike).	The inspector verified the contour of the sealing surface had been restored by welding and subsequent grinding.
8489	Primary Containment Instrument gas inlet filters installed backwards.	The inspector determined the filters had been reinstalled correctly, the welds were visually acceptable and the fittings indicating correct fitup for socket welding were present.

Functioning of the Nonconformance Report system was found adequate.

10.6 Nonconformance Reports Logged in But Not Issued

Bechtel maintains a file of Nonconformance Reports that have been placed in the tracking system but not issued based on management assessment that the items in question did not constitute NCRs. To determine if these unissued NCRs were properly dispositioned, the inspector reviewed those concerning piping installation. The inspector selected two unissued NCRs for a detailed review. The results were as follows:

Inspection Report Number 0-P-EA-01-8-P-1.10

Subject: Wall Thickness Below Minimum on 28" Diameter Schedule 40 Pipe. Spool No. 1-EA-034-503, 503A, 503B

Note: This is part of Service Water Cooling System piping located on discharge side of the strainers in the Intake Structure.

Disposition: Spool pieces were replaced with acceptable products.

Control Number N-15

Subject: 1" Diameter Pipe Lacked Markings for Traceability

Disposition: This 4 ft. section of pipe was originally properly marked when installed. A design change required installation of a Tee in the line. When the pipe was cut in place for the Tee, the identification was not transferred. The inspector verified the original pipe was not removed and the required material identification was shown on drawing 1-P-EE-387.

No violations were observed.

10.7 Trend Analysis

The inspector reviewed the Bechtel system for trend analysis of Nonconformance Reports (NCR). Sorting of the NCRs for trending is computerized using a nine digit code. The first three digits are the problem code, the next three, the commodity and the last three responsibility. Responsibility has only two code numbers indicating Bechtel or Vendor. The inspector chose NCR's relating to pipe support location for verification of trend analysis.

In reviewing the QA Tracking and Trending Profile on a computer printout of this category, the inspector noted approximately two hundred entries from the original date of July 1984. Two Quality Assurance Reports had been written on the results of analysis of these reports.

The first was dated 3/29/84 closed 8/7/84, the second QAR was dated 1/25/85 and closed 4/3/85. The analysis was performed by individuals in the Quality Assurance organization. The inspector considered the resolution of these QARs to be satisfactory and no violations were identified.

The inspector also reviewed the program for trend analysis of Deficiency Reports (DR), Audits, Corrective Action Requests and Management Action Requests by the licensee. This program is the responsibility of the Training and Analysis Group of the Nuclear Department. This group is doing the trending for both Hope Creek and Salem. The computerized system is set up to sort DRs by department (maintenance, chemistry, etc.), System, Component (based on the material equipment list designations), vendor, cause and organization reporting. Data entry has started, however, at the time of this inspection, there was insufficient deficiency data entered in the system to support a meaningful analysis.

10.8 Personnel Questionnaire for Departures

When Quality Assurance/Control personnel terminate employment with Bechtel or Public Service Electric and Gas Company, they are requested to fill out a questionnaire to give their opinions of the quality assurance program on the site. This questionnaire includes questions on the implementation, plant design and other concerns the departing employee might have. The inspector reviewed approximately 160 of these questionnaires. The results of this review were:

a. One complaint about lack of training to give consistency in the implementation of the program.

b. Two individuals refused to fill out the questionnaire.

c. Two individuals were complimentary about the program implementation.

d. There were no other comments.

The inspector concluded that no further followup in this area was warranted.

10.9 QC Inspections of Steam Condensing Mode Isolations

The steam condensing mode of the Residual Heat Removal system has been deleted in accordance with FDDR KT1-1323. The deletion of this mode of operation requires the isolation of various piping and components. This is accomplished primarily by closing and deactivating motor and air-operated valves. In addition, those lines which connect to the primary containment were blocked off by the installation of blind flanges in the lines on the containment side of certain relief valves and vacuum breaker valves. In the course of verifying this isolation from the containment, an inspection of the physical piping and the associated quality control records was performed. The following Quality Control Inspection Reports (QCIRs) document the installation of blind flanges on the outlet of relief valves 1-BC-PSV-F097, 1-BC-PSU-F055a and 1-BC-PSU-F055B respectively: BC-01-040A-P1.10, BC-03-07-D, and BC-01-039-C. The following QCIRs document the installation of blind flanges on the containment side of vacuum breaker MOVs 1-BC-HV-4420A, 1-BC-HV-4420B and 1-BC-HV-4421 respectively: BC-06-02A-P1.10, BC-06-19-A, and BC-06-12-B. These QCIRs were reviewed and no deficiencies were noted. The blind flanges are not shown on the system Piping and Instrumentation Drawing (PRID) but are on the fabrication isometric drawing. Licensee personnel stated that the flanges would be added to the system isometric drawings based upon the information from the as-built verification walkdown.

The inspector had no further questions.

10.10 Engineering Assurance

The Bechtel Nuclear Quality Assurance Manual (NQAM) specifies, in Section II, the requirement to have Engineering Department Procedures. Reference is made to ANSI-N45.2.11 as a source of requirements. The following Engineering Department Procedures (EDP) were reviewed and evaluated for conformance with Section II of the NQAM:

EDP 4.27, Revision 0, Design Verification
EDP 4.49, Revision 2, Project Specifications
EDP 4.37, Revision 7, Design Calculations
EDP 4.34, Off-Project Design Review (Design Control Check List and Design Review Notice)
EDP 4.62, Field Change Request/Field Change Notice (FCR/FCW)

No deficiencies were identified.

11.0 Followup on Outstanding Inspection Findings

11.1 (Closed) Construction Deficiency Report (85-00-08) High Resistance Connection on Bailey Type RZ Push Button Control Modules

Reference: Letter to NRC July 17, 1985

The push button switch units had oxidation of the spring steel jumper clip used to connect the normally open contacts in series. This problem was discovered when contact resistance testing of twelve RZ modules indicated 112K ohms and 125K ohms resistance on two of seventy two push button switch units. These values of resistance were high enough to prevent the 862 digital and the 70000 analog systems from performing as required when the push buttons on the RZ modules was depressed.

The licensee corrective action was to replace the jumper clips with soldered connections using #22 AWG bare solid wire. This replacement required 8400 connections to be made to 700 RZ modules.

The inspector discussed this completed field change with the licensee. The work was performed by Bailey who also provided quality control (QC) for this modification. The inspector reviewed the Problem Report, Engineering Notice, selected QC documents and concluded that this item is closed.

11.2 (Closed) Construction Deficiency Report (85-00-09) Environmental Qualification Failure of Bailey 862 Logic Modules

Reference: Letter to NRC October 10, 1985

During environmental qualification (EQ) of the 862 logic module misoperation was noted when the relative humidity (RH) was 60%. The specification requires that the units be designed to operate at 80% RH continuous and between 80% to 90% RH non-condensing for 24 hours. This misoperation was caused by electrical leakage current between the printed circuit board pads for the front panel set/reset toggle switches. The physical separation of these pads is insufficient to prevent current leakage when the RH is within the specified design limits.

The licensee corrective action was to modify all logic modules including 200 spares. This modification was to increase the gap between the pads by removing a portion of each pad followed by a general cleaning of the immediate area on the printed circuit board. This modification was to be performed by Bailey who would also provide quality control.

The inspector discussed this modification with the licensee. The licensee indicated that this problem occurred some time after the logic modules were installed because the EQ was held up due to the EMI problem. The EQ would include the logic module as modified for CDR 84-00-14. The inspector reviewed the Engineering Notice, selected QC documents and conclude this item is closed.

11.3 (Closed) Unresolved Item 85-34-01; Construction Deficiency CRD 85-00-04

This item relates to an excessive heat buildup problem in the emergency diesel generators local generator potential and excitation control panels. The excessive temperature rise within these units could cause failure of current transformers insulation causing short circuiting of coils and resulting in the loss of excitation current to the diesel generator. Loss of power from the emergency diesel generators would adversely affect safe shutdown of the plant during emergency conditions. This problem was reported by the manufacturer in accordance with 10 CFR 21 and by the licensee in accordance with 10 CFR 50.55(e).

The licensee has modified the panels in accordance with the manufacturers instructions and has conducted tests and evaluations which demonstrate that the heat buildup problem has been solved. These tests and evaluations were reported to Region I by letter dated November 6, 1985. This item is closed.

11.4 (Closed) 85-00-06 Unresolved Item - Bussman Fuses

In accordance with 10 CFR 50.55(e), on April 16, 1985, the licensee reported a potentially significant construction deficiency concerning 155 unqualified Bussman fuses amp and under in panels supplied by Comsip, Inc.

The inspector verified that the unqualified fuses have been replaced with qualified Bussman type BBS fuses for under 1 ampere and Bussman KTK fuses for 1 ampere applications.

This item is closed.

11.5 Follow-up on IE Bulletin (80-17) Resolution

11.5.1 General

IE Bulletin 80-17 was issued to document an event during which almost one-half of the control rods at Brown's Ferry Unit 3 failed to fully insert during a scram.

The Scram Discharge Volume System (SDVS) has been evaluated relative to generic NRC design criteria and found acceptable during the licensing process.

The inspector was informed that additional licensee actions are underway to enhance plant operation or surveillance procedures, therefore the inspection scope was limited to the system hardware aspects.

11.5.2 Scope

The SDVS documents identified in Attachment B.4 were reviewed. The components identified in Attachment C.4 were inspected which included for the North bank SDVS:

- Scram Discharge Volume (SDV);
- Scram Discharge Instrument Volume (SDIV);
- SDV vent lines and supports;
- SDIV drain line and supports;
- SDV and SDIV supports;

- SDIV level instrument switches and transmitters;
- control room alarm for SDV high water level;
- control room indications for vent and drain line valve positions;
- control room keylock switches for high water level trip bypass; and
- control room level switch indications.

11.5.3 Findings

The Scram Discharge Volume design and installation was found in accordance with the FSAR commitments.

The inspector noted the licensee has established a well controlled program, the Response Coordination Team (RCT), to assess generic NRC documents such as I&E Bulletins and to institute appropriate corrective actions.

The inspector had no further questions.

12.0 Unresolved Items

Unresolved items are matters about which more information is necessary to determine whether they are acceptable, violations or deviations. Unresolved items are discussed in Sections 3.4 and 5.3.

13.0 Exit Interview

The inspectors met with the licensee representatives denoted in paragraph 2 at the conclusion of the inspection. The inspector summarized the scope and findings of the inspection and the need for licensee attention to address those issues remaining unresolved. No written material was given to the licensee during the course of this inspection. At the exit, the licensee did not identify any proprietary material contained within the scope of the inspection.

ATTACHMENT 1

RHR/LPSI MOV DEGRADED GRID OPERABILITY

The purpose of this calculation is to ascertain if the RHR/LPSI motor operated valve(MOV) ,1-BC-HV-FO17A,starter contactor will pickup(close) to permit the valve opening during a low voltage condition(92%) at the 4KV bus which is 86.2% at the 480 volt motor control center(MCC). This low voltage condition is for the redundent MCC and MOV which was taken from Ref.1 and is a worst case.

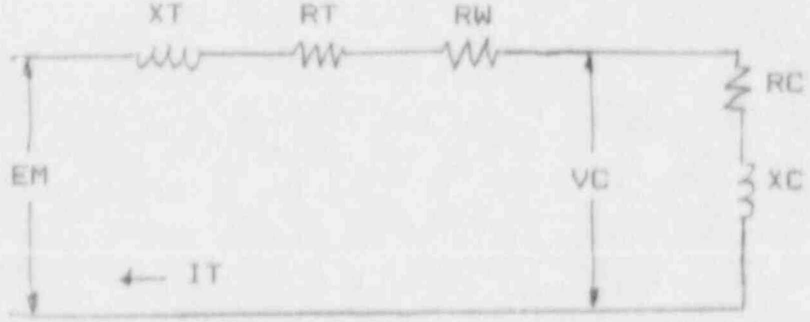
References:

1.Millstone Voltage Study-1E Busses 10855 Calculation No.15.1(Q)
2.Control Transformer Selection and Maximum Circuit Lengths for MCC Control Circuits 10855 Calculation No. 17A(Q)
Eaton /Cutler -Hammer Data 10855-E118(q)-107-1
Okonite Cables -General Conductor Information-Bulletin 721.1
3.Cable Termination Tickets

Assumptions:

1. The operating temeperature of the cable conductor is assumed to be 50 degrees C.
2. The control cable wire reactance as compared to the resistance is neglible and is not considered in this calculation.

Calculation



Starter Control Circuit Impedence Diagram

EM-minimum control transfermer secondary voltage
RT-control transformer secondary resistance
XT-control transformer secondary reactance
RW-control circuit wire resistance
VC-control contactor coil minimum pickup voltage
RC-control contactor coil resistance
XC-control contactor coil reactance

Control Transformer Data (Ref.2)

Rated voltage primary= 480 volts
Rated voltampere= 200 VA
Turns ratio= 3.804
%IZ=4.26,%IR=4.25,%IX-.21
Tolerance =plus or minus 7 1/2%

Contactor Coil Data (Ref.2)

Size 1 Starter
Rated voltage 120 volts,Inrush VA=102.60,Inrush amps=0.855,Inrush watts=56.55,Inrush P.F. Calc=55.12,
Inrush VAR Calc =85.61,Minumim pickup voltage=84% of rated

Control Wire Data(Ref 2.)

Conductor 14AWG annealed coated copper ,stranded class B,
25 degrees C ,dc resistance per 1000 feet=2.73 ohms
Resistance temperature correction factor 50 degree C=1.096

Control Transformer Calculation

RT=%IR x rated voltage squared x tolerance divided by 100 x rated VA
=4.25 x 120 x 120 x 1.075 divided by 100 x 200
=3.29 ohms
XT=%IX x rated voltage squared x tolerance divided by 100 x rated VA
=.16 ohms

Contactor Coil Calculation

RC=Inrush watts divided by Inrush current squared
=56.55 divided by 0.855 x 0.855
=77.36 ohms
ZC=Rated voltage squared divided by Inrush VAR
= 120 x 120 divided by 102.6
=140.35 ohms
XC=Square root of ZC squared minus RC squared
=Square root of 140.35 x 140.35 minus 77.36 x77.36
=Square root of 19698 minus 5985
=Square root of 13713
=117.10 ohms

EM-minimum control transfermer secondary voltage
RT-control transformer secondary resistance
XT-control transformer secondary reactance
RW-control circuit wire resistance
VC-control contactor coil minimum pickup voltage
RC-control contactor coil resistance
XC-control contactor coil reactance

Control Transformer Data (Ref.2)

Rated voltage primary= 480 volts
Rated voltampere= 200 VA
Turns ratio= 3.804
%IZ=4.26,%IR=4.25,%IX-.21
Tolerance =plus or minus 7 1/2%

Contactor Coil Data (Ref.2)

Size 1 Starter
Rated voltage 120 volts,Inrush VA=102.60,Inrush amps=0.855,Inrush watts=56.55,Inrush P.F. Calc=55.12, Inrush VAR Calc =85.61,Minumim pickup voltage=84% of rated

Control Wire Data(Ref 2.)

Conductor 14AWG annealed coated copper ,stranded class B, 25 degrees C ,dc resistance per 1000 feet=2.73 ohms
Resistance temperature correction factor 50 degree C=1.096

Control Transformer Calculation

RT=%IR x rated voltage squared x tolerance divided by 100 x rated VA
=4.25 x 120 x 120 x 1.075 divided by 100 x 200
=3.29 ohms
XT=%IX x rated voltage squared x tolerance divided by 100 x rated VA
=.16 ohms

Contactor Coil Calculation

RC=Inrush watts divided by Inrush current squared
=56.55 divided by 0.855 x 0.855
=77.36 ohms
ZC=Rated voltage squared divided by Inrush VAR
= 120 x 120 divided by 102.6
=140.35 ohms
XC=Square root of ZC squared minus RC squared
=Square root of 140.35 x 140.35 minus 77.36 x77.36
=Square root of 19698 minus 5985
=Square root of 13713
=117.10 ohms

ATTACHMENT 1 PAGE 3

Control Circuit Wire Calculation

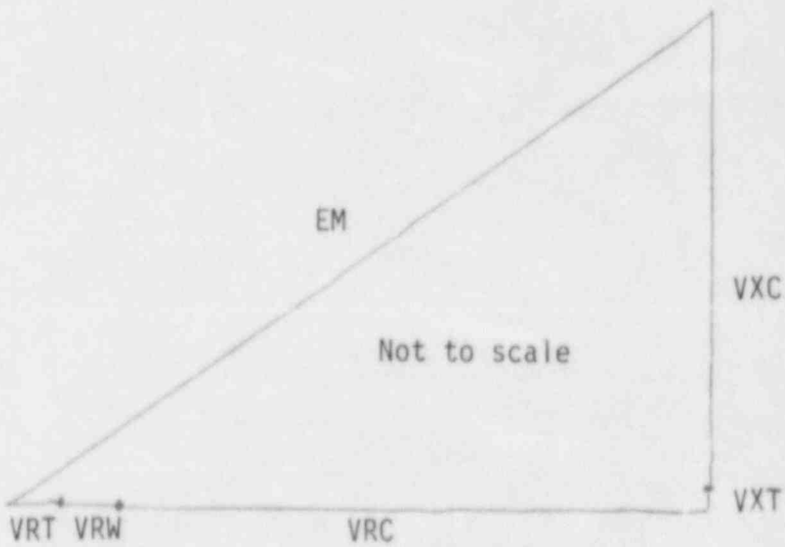
Cable, AP1Q0827B, length from MCC to MOV=200 feet (Ref.3)
Cable, AP1Q0827C, length from MCC to logic panel 10C617BA
=300 feet(Ref.3)
Actual circuit length is from the MOV limit switch to the logic panel x 2
L=200 plus 300 x 2 =1000 feet
The resistance of the 14 AWG wire is 2.73 ohms per 1000 feet x 1.096=2.99 ohms corrected to 50 degree C.Make this 3 ohms.
Control wire circuit resistance RW=3 ohms

Inrush Current At Minimum Pickup Voltage

IT=.84 x 120 divided by XC
=100.8 divided by 140.35
=.7182 amperes

Voltage Drop Calculation

Transformer
VRT= RT x IT
= 3.29 x 0.7182
= 2.36 volts
VXT= XT x IT
= 0.16 x 0.7182
= 0.115 volts
Control Wire
VRW= RW x IT
= 3 x 0.7182
= 2.15 volts
Coil
VRC= RC X IT
= 77.36 x 0.7182
= 55.6 volts
VXC= XC x IT
= 117.10 x 0.7182
= 84.1



Voltage Vector Diagram

The minimum voltage required on the control transformer secondary EM = Square root of (VRT + VRW + VRC)squared + (VXT+ VXC)squared
=Square root of(2.36+2.15+55.6)squared + (0.115 + 84.1) squared
=103.47 volts
The minimun voltage available on the control transformer secondary is =Minimun primary voltage (Ref.1) divided by the transformer turns ratio(Ref.2)=413 divided by 3.804=108.56 volts.

Conclusion

Since the available minumun voltage, 108.56, is greater than that required ,103.47 the valve should open during this degraded voltage condition.

Attachment 2

Service Water Voltage and Cable Pull Tension Calculations

I. Voltage at motor terminals of station service water pump A 1AP502 and at station service water motor control center A MCC 10B553

A. Station Service Water Pump

-- Pump Data: 4KV, 800 hp, 84% efficient, 885 rpm, 16,500 gpm, 150 ft. head, 111 amperes full load running current, 666 amperes locked rotor (maximum starting current)

-- Power Cable: Okonite 4/0, 3 conductor, stranded copper, 1010 ft. long (actual pull length)

Impedance from Okonite Company Technical Bulletin EHB-78, Tables 1.3, 1.4, 1.5 and 3.1

Rdc = 0.0525 ohms per 1000 ft. at 25°C from Table 1.3

Rdc = 0.0525 ohms x 1.25 = 0.0656 ohms at 90°C from Table 1.4

Rac = 0.0656 x 1.05 = 0.0689 ohms from Table 1.5

Paired cable outside dimension = 1.219 inches which provides a cable cradle factor of 1.219 x 1.15 = 1.40 inches.

Using the 1.40 inches cradle factor the cable reactive impedance x_r = 0.046 ohms from Table 3-1.

$z = \mathrm{sqrt}\ (R^2 + x_r{}^2) = \mathrm{sqrt}\ (0.0689^2 + 0.046^2) =$ 0.08352 ohms per 1000 ft. of cable

$z = 0.08352 \times \frac{1010}{1000} = 0.0843752$ ohms for 1010 ft. of cable

-- Voltage Drops:

Running V = 1.73IZ = 1.73x111x0.08437 = 16 volts

Locked Rotor V = 1.73IZ = 1.73x666x0.08437 = 96 volts

-- Voltage Motor Terminal

Nominal 4160 volts - 16 volts drop running = 4144 running

Nominal 4160 volts - 96 volts drop starting = 4064 volts starting

Degraded Grid (92%) 3827 volts - 16 volts drop running = 3811 volts running

3827 volts - 96 volts drop starting = 3731 volts starting

FSAR Section 8.3.1.1.5b. Design Criteria for Electrical Equipment states "The Class 1E motors are specified with accelerating capability at 80% of nominal voltage at their terminals." The voltage drops shown above do not appear excessive for motor starting or running.

B. Station Service Water Motor Control Center MCC 10B553

-- Motor Control Center Data: 480 volts Full Load Current 150 amperes. Locked Rotor current 900 amperes

-- Cable Data: Two Okonite 500KCM Triplex Copper Stranded Cable, Insulation Class B, length 1000 feet. Impendance data from Okonite Technical Bulletin EHB-78, Tables 1-3, 1-4, 1-5 and 3-1.

Rdc = 0.0222 ohms at 25°C per cable

Rdc = 1.25 x 0.0222 = 0.02775 ohms at 90°C

Rac = 0.02775 x 1.13 = 0.03136 ohms at 60 cycles

$$Rac = \frac{0.03136}{2} = 0.01568 \text{ ohms per phase}$$

Using a spacing of 1.24 inches between conductors and considering magnetic binding for multiconductor cable;

X_r = 0.03478 per phase per conductor

X_r = 0.01739 ohms per phase.

$Z = \text{sqrt}(R^2 + X^2) = \text{sqrt}(0.01568^2 + 0.01739^2)$

Z = 0.03552 ohms

-- Voltage Drops: Running $V = 1.73IZ =$ 1.73x150x0.03552 = 9 volts
Locked Rotor 1.73x900x0.03552 = 54 volts

-- Voltage at Motor Control Center

Nominal 480 - 9 volts = 471 volts running

Nominal 480 - 54 volts = 426 volts starting voltage

Degraded Grid 442 - 9 volts = 433 volts running

Degraded Grid 442 - 54 volts = 388 volts starting voltage

FSAR Section 8.3.1.1.5b Design Criteria for Electrical Equipment states "The Class 1E motors are specified with accelerating capability at 80% of nominal voltage at their terminals". The voltage drops shown above do not appear excessive for motor starting or running.

II. Cable Pull Tension Calculations for Service Water Pump Cable AC 10205A from Pump 1AP502 to Manhole 15 AMODOL. (Refer to Figure 1 for the routing referred to in the calculation)

Formula calculations from GE Technical Handbook 8C-1 and from Okonite Technical Bulletin EHB-78

- Straight line pull tension = Length x cable weight x coefficient of friction

- Pull tension through an angle = tension up to the angle x angle factor

- Maximum cable tension to prevent exceeding cable sidewall pressure limitations when pulling cable around a bend is equal to the radius of the bend in feet times the sidewall pressure for this cable which is 500.

Formula input data: Cable weight = 3 conductors x 1.161 lbs./ft. per conductor = 3.483 lbs./ft. (use 3.5lbs/ft.)

Coefficient of friction: 0.5

Angle factors: 1.14 for 15°; 1.30 for 30°; 1.48 for 45°; 1.70 for 60°; 1.94 for 75° and 2.20 for 90°.

Calculations:

Tension	AV to AU = 2 x 3.5 x 0.5 =	3.5*
	AV to AT = 2.5 x 2.2 =	7.8
	AT to AS = 1 x 3.5 x 0.5 =	1.8
	AV to AS = 7.8 + 1.8 =	9.6
	AN to AR = 9.6 x 1.14 =	11
	AR to AQ = 16 x 3.5 x 0.5 =	28
	AV to AQ = 28 + 11 =	39
	AV to AP = 39 x 1.14 =	44.5
	AP to AO = 2 x 3.5 x 0.5 =	3.5
	AV to AO =	48
	AV to AN = 48 x 1.14 =	55
	AN to AM = 3 x 3.5 x 0.5 =	5
	AV to AM = 55 + 5 =	60
	AV to AL = 1.14 x 60 =	68
	AL to AK = 12 x 3.5 x 0.5 =	21
	AV to AK = 68 + 21 =	89
	AV to AJ = 89 x 1.14 =	101
	AJ to AI = 99 x 3.5 x 0.5 =	173
	AV to AI = 173 + 101 =	274
	AV to AH = 274 x 1.14 =	312
	AH to AE = 85 x 3.5 x 0.5 =	149
	AV to AE = 312 + 149 =	471
	AV to AD = 471 x 1.14	537
	AD to AA = 408 x 3.5 x 0.5 =	714
	AV to AA = 537 + 714 =	1251
	AV to Z = 1251 x 1.48 =	1426
	Z to Y = 68 x 3.5 x 0.5 =	119
	AV to Y = 1426 + 119 =	1545
	AV to X = 1545 x 1.48 =	2287
	X to V = 16 x 3.5 x 0.5 =	28
	AV to V = 2287 + 28 =	2315

* All values are in pounds of pulling tension.

The cable tension from point AV to point V represents the total calculated tension to pull the three single conductor 4/0 service water pump power cables from the pump 1AP502 to the manhole 15 AMODOL just outside the emergency diesel generator building.

Cable pulling tension is limited in this pull by cable strength rather than sidewall pressure. The radius of the sharpest cable bend is three feet. Using this as a basis for calculating allowable pulling tension to prevent exceeding sidewall pressure limitations yields the following:

500 pounds x 3 = 1500 pounds per cable x 3 cables = 4500 pounds x 2/3 derating factor for one cable riding another during the pull = 3000 pounds allowed pulling tension. Since the maximum cable tension calculated is 2315 pounds, sidewall pressure is not limiting.