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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

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AUG 11 1987

Docket No. 50-443

Public Service of New Hampshire
ATTN: Mr. Robert J. Harrison
President and Chief Executive Officer
P. O. Box 330
Manchester, New Hampshire 03105

Gentlemen:

Subject: Inspection Report NO. 50-443/87-07

This refers to the special allegation inspection conducted by Mr. Jacque P. Durr of this office on April 6 - 10, 20 - 24 and May 4 - 8, 1987 of activities authorized by NRC License No. CPPR-135 at Seabrook Facility Unit No. 1, Seabrook, New Hampshire and to the discussions of our findings held by Mr. Durr with Mr. T. Feigenbaum of your staff at the conclusion of the inspection.

Areas examined during this inspection included technical issues and new allegations raised by the Employee's Legal Project in letters to the NRC dated February 9 and April 2, 1987, and to information provided during an interview on April 20, 1987. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, measurements and observations by the inspectors.

Within the scope of this inspection, no violations were observed.

No reply to this letter is required. Your cooperation with us in this matter is appreciated.

Sincerely,

William V Johnston

William V. Johnston, Acting Director
Division of Reactor Safety

Enclosure: NRC Region I Inspection Report Number 50-443/87-07

cc w/encl:
Seabrook Hearing Service List
Ted. C. Feigenbaum, Vice President of Engineering and Quality Programs
William B. Derrickson, Senior Vice President
Warren J. Hall, Regulatory Services Manager
Donald E. Moody, Station Manager
Employee's Legal Project
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New Hampshire

C. Scott Kennedy
9 Keldon Way
Norwich, MA 02645

Ms. Sharon Tracy
c/o COL G.W. Tracy
Rt. 2 Box 60B
Leesburg, VA 22075

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/87-07

Docket No. 50-443

License No. CPPR-135 Category B

Licensee: Public Service of New Hampshire

P. O. Box 330

Manchester, New Hampshire 03105

Facility Name: Seabrook Unit No.1

Inspection At: Seabrook, New Hampshire

Inspection Conducted: April 6-10, 20-24, May 4-8, 1987

Inspectors:	E H Gray E. H. Gray, Lead Reactor Engineer	7/13/87 date
	Chaudhary S. K. Chaudhary, Lead Reactor Engineer	7/13/87 date
	Kamal A. Manoly K. A. Manoly, Lead Reactor Engineer	7/13/87 date
Team Leader:	J. P. Durr J. P. Durr, Chief, Engineering Branch	7/17/87 date
Approved by:	William V Johnston W. V. Johnston, Director (Acting) Division of Reactor Safety	8/4/87 date

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EXECUTIVE SUMMARY

Background

The NRC performed an inspection on November 3 - 14 and 18 - 21, 1986, report 50-443/86-52, which dealt with a series of allegations presented by the Employee's Legal Project (ELP) regarding the quality of construction at the Seabrook station. Subsequently, the ELP performed a review of the report and raised technical questions and comments concerning the contents of the report, and, in some cases, provided new information regarding the original allegations. This inspection was initiated to deal with the questions, comments and new information and to provide answers where a response was appropriate.

The NRC staff met with the ELP and some of the allegers on April 20, 1987, to gather additional detailed information concerning the technical issues. During the interview, the ELP presented a document to the inspection team containing new allegations. These new allegations, where sufficient information exists, are addressed in this report. Where there is insufficient information available to perform an inspection, the ELP was requested to supply more information to enable a proper disposition of the allegation.

Findings

The inspection team performed on site inspections during the period of April 6 - 10, 20 - 24, and May 4 - 8, 1987. Based on the new information, the team was able to more accurately identify the specific areas of concern relative to the original allegations. The team determined that the allegations that were previously inspected remained as originally characterized and do not represent a safety concern. In two instances the original allegations identified conditions which, although the NRC and the utility were aware of the issues and the utility is in the process of addressing them, have not yet been resolved. The remaining allegations could not be substantiated. The new allegations also could not be substantiated as valid technical issues that the licensee was not previously aware and addressing.

It has become apparent from the nature of the allegations and the resulting engineering evaluations that the allegers are reporting events to ELP that are only partly based in fact. Apparently, sources of information to ELP have witnessed an event or heard part of a story that would, if not identified and corrected, constitute a safety concern. However, in almost every case, the event has been previously identified and properly dispositioned by the licensee. In the case of the fire protection piping alleged to be clogged, it is clear that two separate events have been mixed to arrive at the inaccurate perception that the fire piping is clogged.

The report contains two unresolved items that will receive further attention. One item deals with the long term effects of ground water on the reinforcing steel and the licensee's program to monitor the condition of the structures affected.

The second issue is an in-depth review of the concrete cracks in structures such as the waste process building and the cooling towers. Neither of these items present an immediate concern to the operation of the facility.

SEABROOK ALLEGATION FOLLOWUP

1.0 Introduction

1.1 Background

The Employee's Legal Project (ELP) transmitted a series of allegations to the NRC which were inspected and resolved in inspection report 50-443/86-52. The allegations were, in some cases, not specifically defined; however, the inspection team addressed the issues based on the information presented in the allegations and supplemented by an interview of ELP conducted on November 3 - 4, 1986. Subsequent to the issue of the NRC report, ELP generated comments on the content of the report, dated February 9 and April 2, 1987, which contained additional information regarding the original allegations, new allegations, and technical differences that warranted further NRC review and inspection.

The NRC mobilized another inspection team to address the new information and conducted an onsite inspection on April 6-10, 20-24, and May 4-8, 1987. The team again interviewed ELP representatives on April 20, 1987, to obtain clarification of the issues presented in the February 9 and April 2, 1987, submittals. During the interview, the ELP presented a package of additional allegations which addressed new subjects and restated some of the original issues. Contained within this new information were affidavits dated October and November, 1986.

1.2 Inspection Scope and Report Organization

The February 9 and April 2, 1987, ELP documents were reviewed by the NRC staff to extract those statements that constituted new allegations or warranted a technical response to resolve any differences between what was previously inspected and what the ELP and their allegers felt were technical errors in the inspection. These statements were sequentially numbered as they appeared in the ELP documents to ensure that all issues were identified and addressed. If, after review and evaluation, the statements were determined not to fit in an allegation or technical response category, they were deleted from the inspection list.

The package that was presented to the NRC inspection team on April 20, 1987, at the ELP interview was also reviewed for new allegations or information that warranted further NRC action. Any specific information identified in the review that was inspectable as presented was addressed in the May 4-8, 1987, inspection. Issues that required more information because of their nonspecificity, were referred back to the ELP with a request for further information in a letter dated May 27, 1987.

This report is organized into two types of issues, allegations and technical responses. The sources of the allegations are the February 9 and April 2, 1987 submittals by the ELP and the April 20, 1987, transcript of the interview with the allegers and ELP representatives and the package of additional allegations received during that interview. The technical responses are requests for added information by the ELP concerning statements made in Inspection Report 50-443/86-52 by the NRC. They also represent NRC responses to questions or technical differences expressed by the ELP in response to NRC findings in Inspection Report 50-443/86-52 that warrant clarification.

1.3 Summary Conclusion

The inspections have not disclosed any violations of regulatory requirements and have not identified any equipment deficiencies that were not already under evaluation or previously dispositioned by the licensee and/or the NRC.

The NRC has expended approximately 1000 man-hours in direct on-site inspection effort using highly qualified engineers to evaluate the allegers concerns. The 1000 man-hours does not represent additional NRC time that was spent in-office reviewing documents, preparing for the inspections and documenting the findings. Physical inspections and tests which necessitated the disassembly of equipment have been a part of this inspection effort. It is apparent from this effort that, although the allegers may have been factually correct in part, they were not aware of all relevant technical information relative to the issues they raised. When all relevant information is considered, the NRC has not identified a technical concern.

Throughout the ELP's response to the NRC inspection report 50-443/86-52, they allege and make reference to program deficiencies that may have resulted in substandard construction. These alleged program deficiencies range from poor training of craftsmen, drawing controls, design deficiencies, construction practices and quality control/quality assurance to widespread use of drugs and alcohol. Although, during construction these programs were, in some cases, found to warrant improvements, the licensee applied the appropriate resources to correct identified deficiencies prior to this inspection. Based upon the allegers and ELP concerns, the licensee and the NRC have performed independent inspections and have confirmed that none of the allegations have resulted in equipment deficiencies. In most cases, the alleged condition was previously identified by the licensee or the NRC and corrective actions documented. This discrepancy between ELP's sources of information and NRC findings appears to result from the fact that allegers to ELP witnessed the initial condition but not the corrective process.

2.0 Inspection

2.1.0 Technical Response

2.1.1 Source of the Issue

ELP draft response, page 4, last paragraph: "...Allegation No. 46 refers to cold pulling in the condenser piping, not the main steam feedwater system mentioned by the NRC."

2.1.2 Scope of the Issue

The ELP draft response referred to the discussion provided by the NRC, in inspection report 86-52, in response to allegations No. 40 and 46. The NRC's understanding of the concerns in the allegations was that they related to the use of comealongs in the fit-up operation for a main steam line final closure weld, and to the practice of cold pulling to align piping during erection. Allegation No. 46 was obtained from the Newburyport Daily News, October 20, 1986, issue and did not contain any detailed information.

The ELP statement in the draft response quoted above was reiterated during the meeting between the NRC and the ELP on April 20, 1987. The alleger stated that he observed a piping crew attempting to cold spring a pipe into place using a 2-3 ton capacity chainfall. He further stated that the piping of concern was in the 13th stage steam dump in the No. 8 condenser at approximately 48' - 0" elevation inside the turbine building. The alleger also stated that the pipe was in the area between the exhaust nozzles of the turbine and the condenser shell.

2.1.3 NRC Inspection

The NRC evaluation of the allegation concerning the condenser piping is provided in this section. The NRC review of the overall concern regarding cold pull of piping during erection is provided in section 2.18 of this report.

To address the condenser piping allegation, the NRC team conducted the following:

- Review of the turbine extraction steam and drain piping drawings
- Inspection of the steam extraction piping inside the condenser shell from each Low Pressure (LP) Turbine to the feedwater heaters and the drain lines to the condensers.
- Interviews with responsible turbine system engineering personnel.

Based on the above, the NRC staff determined that the cold pulling of piping in the 13th stage steam dump in the No. 8 condenser could not be possible since there are only three (3) low pressure turbines designated as A, B & C, respectively. Further, there are no provisions for steam extraction from the 13th stage in each L.P turbine. Steam extraction is provided for, however, in each of the 8th, 9th, 11th and 12th stages through two lines in each L.P hood.

The extraction lines vary in size between 10" and 20" in diameter and are directed to the feedwater heaters (no. 21 through no. 24). The 13th stage is provided with four 12" diameter drain lines (with a 2 3/4" diameter orifices) in each L.P turbine which dumps directly to the condensers. The drain pipes are designed for water removal from the 13th stage buckets and are open ended at their discharge locations (i.e not welded); thus, they are only fixed at one end. Further, none of this piping is safety related.

2.1.4 Conclusions

Based on the above evaluation, the staff concluded that the allegation involving cold pulling of the 13th stage turbine dump piping at the Seabrook Station is unsubstantiated. Cold pulling, as defined in section 2.18.3, of piping fixed at only one end is not possible.

2.2.0 Technical Response

2.2.1 Source of the Issue

The ELP Draft Response, on page 7, asked a question as to when were Control Building Air Handling System (CBA) drawings issued.

2.2.2 Scope of the Issue

This question arises from the original allegation from ELP regarding the above system. The original allegation stated that the CBA system was constructed without design drawings and that the original installation consisted of 2 and 3 inch steel pipe which was replaced with 3/4" diameter copper tubing. This allegation was evaluated by the NRC as item Number 54 in the NRC Report No. 50-443/86-52.

2.2.3 NRC Inspection

The NRC staff examined the design/construction history and schedule of the CBA system. The review consisted of the examination of design documents such as design drawings and specifications, document control logs, Engineering Change Authorizations (ECA), construction schedule and procedures, and the preoperational test sequences. The inspector also interviewed and held discussions with engineering and supervisory personnel responsible for construction, maintenance, and operation of the system.

Based on the above reviews and examinations, the inspector determined that design drawings for the CBA system were issued for construction in the later part of 1980, and the engineering specification (9763-006-248-1) was originally issued in 1975 which was revised and reissued as revision 4 in 1980. The erection of the system did not start until 1982.

The inspector also determined that the piping material specified in the specification 9763-006-248-1 had always been copper for CBA refrigerant lines (ASTM B-88, Type L, Seamless Hand Drawn), not steel as previously alleged. Also, detailed construction drawings issued in late 1981 and early 1982 by Pullman Power Products (P-H) for the erection of the system piping showed copper tubing in the bill of materials for CBA piping. The inspector did not find any evidence that steel piping (2" and 3" diameter) was ever specified and/or used in CBA refrigerant lines; and established that the design of the CBA system was completed and drawings issued for construction well before the start of the erection of the system.

The inspector, however, also determined that a modification in the air duct work in the system was implemented in the CBA system. This modification resulted from a design change due to a revised heat-load calculation in the computer room. The computer room was removed from the main CBA system, and was provided with a self-contained air conditioning unit entirely within the room. Steel air ducts carrying refrigerated air from the CBA system to the computer room were removed. Air-handling ducts generally are galvanized sheet metal, but in some cases steel piping is also used. At Seabrook, the removed air ducts were circular steel piping.

2.2.4 Conclusions

The CBA system design was completed, and design drawings issued before the erection of the system started. At no time during the construction was 2" and 3" diameter steel piping used in the refrigeration piping.

2.3.0 Allegation

2.3.1 Source of Issue

The Draft ELP Response, page 11, states:

"On page 1-5, the Executive Summary, the report says workers were trained and qualified to American National Standards Institute requirements. However, Doug Richardson, an as-built piping inspector for P-H was not certified to these requirements and never heard of "ANSI N45.2.6."

2.3.2 Scope of Issue

The NRC concern in this case is that the licensee was utilizing personnel for safety related quality work that should have been qualified to ANSI N45.2.6 and were not.

2.3.3 NRC Inspection

The NRC met with the alleger on April 20, 1987, in order to determine the exact nature of work he performed at the facility. The interview disclosed that the alleger was employed at Seabrook on two different occasions. He was first employed by the Pullman-Higgins Company during the period of May 1982 through January 1983 as an "As-builder". In this position, he stated that he did not do safety related work. He was then employed by United Engineers and Constructors, Inc. during the period of June 1983 through February 1984. He was unable to recall his exact job title but described his function as taking, "United drawings (which) had already been worked up as design drawings and I was merely modifying them. I was merely marking them up with the as-built dimensional data and other information that was pertinent."

A review of the allegers personnel records indicated that he was employed as a "Construction Drafter" by United Engineers and was assigned to the Site Power Engineering As-Building Group. The activities of this group are described in the United Engineers Field Administrative Construction procedure (FACP) No. 13 and are the responsibility of engineering not quality control. The group consisted of as-builders and draftsmen assigned the responsibility of updating reproducible drawings with as-built data as provided by the as-built crews.

FACP No. 13, paragraph 5.1.7.n), requires that as-builders be certified to Seabrook Procedure QA-9-2, Appendix E, which is the "The Instruction for Selection and Certification of Visual Inspection Personnel". There is no requirement for the qualification of drafting personnel. Further, the procedure requires that after the field marked-up drawing was returned to the United Engineers as-builders office it would be reviewed for deviations and deficiencies.

The Quality Services Manager stated that as-built personnel were responsible for the collection of data and detailing of as-built drawings not for acceptance or rejection of items. He further stated that final acceptance inspection of installed piping systems, components and supports was performed by quality control personnel certified to ANSI N45.2.6.

2.3.4 Conclusion:

The records and the interview with the alleger do not support the statement that the alleger was an inspector in the sense he did quality control or quality assurance work. The allegers primary function and job description is that of a draftsman. The work he performed in the field was reviewed and verified procedurally by his supervision and all acceptance or rejection of items was performed under quality control. There appears to be some confusion between gathering of "as-built" data and acceptance inspection. A system or an item is not "as-built" until and unless it has been inspected and finally accepted by Quality Control. The gathering of "as-built" data is an engineering and drafting function to upgrade the design drawings.

2.4.0 Technical Response

2.4.1 Source of the Issues

The Draft ELP response, received February 9, 1987, stated:

Page 16, 2nd paragraph: "if the cracks leak water, then aren't they more than simply "surface cracks"?

Page 16, 2nd paragraph; ".... is it possible the cracks would leak when ground water was higher?"

Page 16, 3rd paragraph; "There is no discussion of the cracks which did leak water, and the water's effect on the rebar."

Page 17, 1st paragraph; "...if water can leak in, can't radiation leak out in a pressurized situation?"

The ELP supplemental submittal, dated April 2, 1987, also stated:

Page 3, 1st paragraph; "Patching on the inside of the walls does not stop groundwater from affecting the rebar."

2.4.2 Scope of Issues

The above questions and statement all arise from a concern that there are cracks in some of the concrete structures and some of them leak groundwater. The above concerns were forwarded to the NRC in ELP's original list of allegations. These allegations were examined, evaluated, and the conclusions of the NRC were documented in Inspection Report No. 50-443/86-52. The above concerns were addressed in that report as items number: 1, 2, 3, 4, and 17. Except for item 17, all other allegation referred to the containment structure. Item number 17 was the only item which partially dealt with cracks and water seepage in the equipment vault area.

2.4.3 NRC Inspection

In light of the questions raised and statements made in ELP's draft response regarding the adequacy of the NRC's evaluation of these allegations, the NRC staff re-examined the results of the original evaluation documented in NRC report No. 50-443/86-52. Additionally, the NRC performed extensive re-inspections at the plant site of the concrete construction in containment and other safety-related structures. The intent of this re-review and inspection was to collect any additional information which might have become available after the last inspection in this area, or to examine any condition that might have developed or become apparent due to time, change in weather conditions, or service.

2.4.3.1 Containment

Based on the above efforts, the NRC staff has determined that the examination, evaluations, and the conclusions reached by the NRC as documented in IR 50-443/86-52 regarding the integrity and safety of containment are valid, and no safety concern exits.

The staff also believes that the original allegations and the current questions and statements stem from a lack of understanding of the highly complex and specialized field of structural design, intended functions of the containment structure, and the physical and chemical properties of materials used in the construction. In an effort to clarify the design intent and function of the containment for those persons expressing concerns, the following information is presented:

The containment system at Seabrook is a seismic category I reinforced concrete, dry (as opposed to a wet pressure suppression system) structure. It is designed to function at a maximum pressure of 52 psig and is tested for integrity at 60 psig (115% of design pressure), although during normal service, the containment will function at atmospheric pressure. The Code to which the containment structure is designed provides a factor of safety of 1.65; however, this includes conservatisms which in reality provide a safety factor of approximately 2.8.

It consists of an upright cylinder topped with a hemispherical dome, supported on a reinforced concrete mat ten feet in thickness keyed into the bedrock. A welded steel liner plate, anchored to the inside face of the containment wall, serves as a leak tight membrane. This steel liner is generally 3/8 inch thick, and at the junction of the basemat and at penetrations it is 3/4" inch thick. The liner on top of the base mat is protected by an additional four feet thick mat which supports the containment internals and forms the floor of the containment. The concrete wall thickness in the cylinder up to the springline (the point where the dome joins the cylindrical part of the structure) is generally 3'-6", except at personnel and equipment hatches where it is further thickened. The containment has been tested for leak tightness and found to meet the acceptance criteria of 10 CFR 50, Appendix J.

Under the above circumstances it was difficult for the NRC staff to visualize any migration of free water across the containment wall (in or out - leakage,) and the postulated corrosion of reinforcing bars attendant with such water seepage. However, a thorough examination of the interior of the containment did not disclose any evidence of water seepage through the containment wall which confirmed the above conclusion. The NRC staff, therefore, is satisfied that its previous evaluation of containment safety is technically valid. There is no evidence of water seepage, consequently no concern regarding the degradation of reinforcing steel (rebar) due to corrosion.

Regarding the cracks in the containment wall, the NRC's evaluation of their safety implications is based on the following technical information and rationale.

Due to low sustained stresses associated with conventionally reinforced concrete, the structure is normally under a compressive stress. The effects of creep, however, are negligible. The controlling load combination that is used for rebar design is based on accident pressure, which effectively cracks the concrete and places the reinforcement in tension. Since it is accepted practice to ignore the tensile capacity of concrete, a cracked section is the basis of containment design and functional analysis for thermal, pressure, and dead loads. Because the containment wall and dome concrete will crack under maximum design pressure (this includes the structural integrity test and periodic integrated leak rate tests), a crack pattern is predicted for the concrete. The procedure for determining crack patterns involves a check of stresses in each concrete layer of the cross section. The predicted crack pattern and its magnitude is compared to the pattern and magnitude of cracks actually developed during the structural integrity test to verify the validity and accuracy of the containment design and analysis methods. Additionally, these cracks developed during the structural integrity test have the advantage of relieving the stresses due to shrinkage and creep of concrete. However, cracking of concrete does not imply cracking of the steel membrane and possible release of contained fission products.

It, thereby, follows that cracks in the containment wall and dome concrete are not something unusual or unwanted; but are in fact supposed to be there, and are accounted for in the design and analysis. It is not a safety concern.

2.4.3.2 Other Safety-Related Structures

Cracks and groundwater seepage through some of the walls in the equipment vault and waste processing building, has been known and evaluated by the NRC previously, and a summary of the findings is referenced in IR 50-443/86-52. The NRC has a continuing interest in the long term corrective actions that the licensee implements for seepage of groundwater in the equipment vault and waste processing building, and the repair of concrete to alleviate the problem. The effectiveness of such corrective measures are assessed by the NRC inspection staff during their routine walk-through inspections in the plant. The noticeable seepage of groundwater is a process which is affected by several external conditions such as: the hydrostatic pressure of the water table due to weather conditions, permeability of the concrete, permeability of the surrounding soil, and the effectiveness of the waterproofing membrane. Also, the effect of this water seepage on the rebar is related to the groundwater quality and the length of time the seepage is allowed to continue. Therefore, evaluation of the adequacy of corrective measures requires an extended period of observation to encompass as many variable as praticable that affect the seepage process.

The licensee has been engaged in identification and repair of concrete cracks in these areas as they become apparent. The NRC staff's interest in this area is clearly documented in a number of inspection reports starting from the year 1982 (Report Nos. 50-443/82-03, 82-07, 84-12, 86-43, 86-52). As ELP has correctly pointed out, water table and weather conditions affect the leakage; thus, the inspection and evaluation of this problem has to extend over a period of time. One other factor affecting adequate evaluation of this problem has been that the plant site was, until recently, operating a dewatering system to lower the watertable during construction. After the dewatering has been discontinued, sufficient time must elapse for the watertable to stabilize before a realistic evaluation of the magnitude and location of all seepage points can be made, its significance can be established, and any corrective measures implemented.

In the meantime any examination, evaluation, and repair only affects the identified seepage under the prevalent circumstances during and up to that period. In light of the above, the NRC has examined and evaluated the corrective measures to eliminate water seepage and has found them to be generally effective. Thus as the variables affecting watertable change, additional attention is continually needed to control this problem.

In addition to the above, concrete is inherently a permeable material. If exposed to sufficient pressure and time span, water will migrate across concrete members regardless of the quality or construction methods of a structure. Secondly, all conventional concrete cracks due to temperature stress, shrinkage, and volume change which is an inherent characteristic of the material. However, noticeability and visual appearance of seepage depends upon the rate of seepage, temperature and humidity in the environment on the observed side.

The occurrence of concrete cracks and any attendant seepage is not important in itself; rather, the condition they create, and their affect on the serviceability of the structures' intended function is important. It can range from minor asthetic consideration due to dampness and discoloration to free water affecting proper functioning of equipment.

In light of the above, the NRC's evaluation and findings are based on the following engineering consideration. To assess the magnitude of seepage and its effect on the serviceability of the affected structures, the NRC performed an inspection during April 6 - 9, 1987. This particular time was picked by the NRC for several reasons; more important amongst them were: 1) an extended period of rain in the area with consequent rise in hydrostatic pressure due to a rise in the groundwater table; 2) expected accentuaton of seepage locations; 3) to evaluate the licensee's activities with regard to mitigation of the problem; and 4) to assess the effectiveness of previous repairs performed in this area.

The NRC staff visually examined the repaired areas, new seepage locations and their impact on the operation of equipment in the vincinity, and reviewed the licensee's planned corrective actions for technical adequacy. The inspector also requested the licensee to collect samples of seepage water from locations selected by the inspector for chemical analysis of water quality, and to obtain a sample of ground-water from the observation well in the vicinity of the emergency diesel generator building. The groundwater sample was collected to assess its potential for corrosion on rebars, and to compare the chemical changes suffered by seepage water during its transit thorough the concrete. The inspector also reviewed the licensee's plan for long term corrective measures.

Based on the above efforts, the inspector determined that the repairs performed previously to control the seepage were generally effective; however, new seepage locations were identified. The seepage magnitude was from very minor wetness of the surface to moderate drips capable of accumulating puddles of water on the floor. The majority of the new seepage was through hairline cracks and form tie holes in the walls of the waste processing building stairwells on the north and south side. The other seepage locations were randomly distributed in the equipment vault, primary auxiliary building (PAB) and electrical cable tray tunnel, and were generally between elevation - 31' 0" and + 20' 0". The inspector identified three of the heaviest seepage locations in order to collect seepage water samples. One sample was taken in the PAB where there was a leak at the penetration of an electrical through-wall grounding cable. The other two were in the electrical tunnel, one through a hairline crack and another through an unplugged form-tie hole. From the samples collected, the rate of seepage was determined at these locations. The heaviest seepage was in the PAB at the electrical ground cable penetration at an average rate of approximately 3.47 cubic inches per hour during the sampling period, and the other two locations in the electrical cable tunnel the crack and from tie-hole had approximately 0.07 cubic inches and 0.11 cubic inches, respectively.

The seepage water samples and the groundwater sample were analyzed at the onsite chemistry lab for their chemical composition. The NRC inspector monitored the sampling operations and witnessed the entire chemical analysis procedure.

From the above observations, tests, and analyses, the inspector determined that the groundwater at the site is of potable quality and the pH value is on the alkaline side. The most pronounced change in the chemistry of the seepage water was in the pH value. The seepage water was considerably more alkaline, also there was an increase in chlorine and sodium content. The iron content of the groundwater and seepage water were also compared; because, any corrosion in the rebar would increase the iron content of the seepage water due to corrosion products. However, no such increase was evident. These findings were consistent with the metallurgical principal that iron/steel does not corrode in an alkaline environment. Concrete being mainly a combination of Tricalcium and Dicalcium Silicates (3 CaO.SiO3; 2CaO.SiO3) provides an alkaline environment for rebar. Therefore, steel embedded in concrete with adequate concrete cover is relatively protected from corrosion. However, high seepage over a prolonged period of time leaches out the free silicates from the concrete reducing the corrosion inhibiting environment, consequently increasing the potential for corrosion. By the same token, any repair process that stops seepage and eliminates visible leaching assures a prolonged alkaline environment, thereby reducing a potential for severe corrosion of rebars.

Based on the above evaluation and analysis, the NRC has determined that corrosion of rebar due to seepage currently is not a safety concern. However, the effectiveness of the licensee's corrective measures, and the long term effects of seepage, if the corrective measures are not effective, is a matter of concern to the NRC. This matter, therefore, is unresolved pending further review and analysis of the licensee's actions and surveillance measures by the NRC staff. (443/87-07-01)

With regard to ELPs expressed hypothesis of radiation seeping out if the water can seep-in, the NRC evaluated this concern, and determined as-follows.

As discussed previously in paragraph 2.4.3.1, the containment building is designed to withstand an internal pressure of 52 psig and has a continuous steel liner which will preclude radioactive materials from leaking outward. The waste process, primary auxiliary and equipment vault buildings are not pressurized structures and the groundwater in-leakage is driven by the hydrostatic pressure differential across the walls. For any material to leak out through the cracks, a motive force of sufficient magnitude to overcome this hydrostatic pressure would need to be developed. This would require the flooding of these structures to a significant depth. Considering the groundwater levels in relation to the building foundations, it is highly unlikely that a water level high enough to overcome this differential pressure can be established.

2.4.4 Conclusion

The concrete cracks in the containment structure do not seep water and generally are there by design as a result of internal pressurization during testing. Out leakage during a pressurized condition is precluded by the internal steel liner. The integrity of the structure and its leak tightness will be verified throughout the plant's life by periodic testing.

There is no immediate threat to the reinforcing steel of the structures that are currently experiencing water seepage. The NRC and the licensee will monitor the long term affects of groundwater seepage and its affect on the reinforcing steel.

2.5.0 Technical Response

2.5.1 Source of Issue

The ELP Draft Response, dated February 9, 1987, states: "The NRC statement that this process makes the rod stronger than if they had never been improperly severed is debatable".

2.5.2 Scope of the Issue

This issue stems from ELP's original allegations covered in NRC report 50-443/86-52. This issue was listed as item 5 in that report.

It appears that ELP has misread the statement in the report. The NRC never stated that any construction process makes any rod or rebars stronger. To the NRC's knowledge no metallurgical process was used at Seabrook site to alter the strength properties of rebars. What the NRC did say regarding "Cadweld" splices is quoted below:

"'Cadweld' splices are stronger than the rebar itself." (emphasis added)

2.5.3 NRC Inspection

The above statement by the NRC is based on accepted technical literature, industry codes and standards, and the acceptance criteria for such splices established at the project site, which was met or exceeded by the actual test results of production and sister splices tested at the laboratory.

Cadwelds are not true welds in the sense there is fusion between the melted filler metal and the reinforcing steel. The strength developed by the process is essentially mechanical and the splice is designed to exceed the strength of the reinforcing steel bars.

Some of the code and standards requirements regarding mechanical splices of rebar by positive connection, to include Cadwelds, is quoted here:

(1) American Concrete Institute: Building Code Requirements for Reinforced Concrete (ACI 318-71). Section 7.5.5.2.
"Full positive connection shall develop in tension or compression, as required, at least 125 percent of specified yield strength of the bar." (emphasis added)

(2) American Society of Mechanical Engineers" Code for Concrete Reactor Vessels and Containments, ASME, Section III, Division 2, Article 4000, paragraph CC-4331.2:

"The types of splices and joints listed in (a), (b), and (c) below are permitted within the limitations described in the subparagraphs:

(a) lap splices

(b) mechanical splices of the following types:

(1) sleeve with ferrous filler metal splices;"

Paragraph CC-4333.4.4; subparagraph (a):

"The tensile strength of each sample shall equal or exceed 125% of the specified yield strength as shown in Table CC-4333.1."

The NRC has verified over a period of time that the Seabrook project has met these requirements, and specifically verified that "cadweld" splices have met the above acceptance criteria.

2.5.4 Conclusion

ELP's assertion is not valid. The NRC's statement regarding "Cadweld" splices is supported by National Codes and Standards.

2.6.0. Technical Response

2.6.1. Source of the Issue

The ELP Draft Response stated on page 17, last paragraph; "... he did not find it helpful to learn that the inspector simply reviewed documents and ensured that procedures were in place for maintenance of the fire protection equipment.

2.6.2 Scope of the Issue

The alleger, after review of report 443/86-52, is of the opinion that the NRC has only reviewed documents and assured that procedures are in place for maintenance of the Seabrook plant fire protection equipment.

2.6.3. NRC Inspection

The NRC inspection report 86-52 clearly states on page 25 the following: "To establish the present condition of the fire protection system, the inspector observed fire hydrant testing, sprinkler system testing and examined screens as removed from two locations in the fire protection piping." Further, on page 26 of the report it states, "On November 4, 1986, the NRC observed the insurance tests for systems..." The NRC inspector observed the foregoing activities during the on-site inspection conducted during November, 1986.

2.6.4. Conclusion

Report 50-443/86-52 includes accounts of both field inspection and testing by the NRC in addition to interviews with personnel and procedure and documentation reviews.

2.7.0. Technical Response

2.7.1. Source of Issue

The ELP Draft Response on page 18, 2nd paragraph stated "It would be helpful to know what specific procedures exist".

2.7.2. Scope of the Issue

The ELP is requesting information on the types of procedures applicable to the fire protection program and its implementation at the Seabrook plant.

2.7.3. NRC Inspection

The Master Procedure Index, dated 12/19/86, lists 49 procedures applicable to the fire protection program and its implementation. These procedures apply to the following specific activities.

- Fire Pump Tests
- Valve Alignment Checks
- Fire Protection System Flushing
- System Functional Test
- Hydrant Flow Checks
- Spray and Sprinkler Inspection
- Hose Station Inspection
- Hose Replacement, Inspection and Inventory
- Portable Fire Extinguisher Inspection
- Halon 1301 System Inspection
- Fire Door Inspection
- Emergency Lighting Inspection and Testing
- Wet Sprinkler Testing

2.7.4 Conclusion

None

2.8.0. Technical Response

2.8.1 Source of the Issue

The ELP Draft Response on Page 18, 2nd paragraph states "The report fails to indicate from where the screens were removed".

2.8.2 Scope of the Issue

During inspection 50-443/86-52 the screens or systems strainers at two locations in the fire protection system were removed by plant staff and examined by the NRC inspector for evidence of corrosion deposits, debris and overall cleanliness. The ELP is requesting the location of the screens examined during the inspection.

2.8.3. NRC Inspection

Page 26, paragraph 3 of the NRC report 86-52 states: "...the clean condition of system strainers at the input to the containment building FP system and at the most remote portion of the FP system piping, the Service Water Chlorination Building." (emphasis added)

The system strainers identified above are the two fire protection system screens examined by the NRC inspector during 86-52.

2.8.4. Conclusion

Report 86-52 does indicate from where the screens were removed.

2.9.0. Technical Response

2.9.1. Source of Issue

The ELP Draft Response on page 18, 2nd paragraph states; "One assumes the reference is to piping inside the fire pump house and not outside the pump house where the actual piping was observed to contain thick sedimentation."

2.9.2. Scope of the Issue

The above ELP statement is in reference to "Inspection ...of fire pump house piping" discussed in paragraph 3, page 25, of NRC inspection report 50-443/86-52 which is quoted below:

"To establish the present condition of the Fire Protection (FP) System, the inspector observed Fire Hydrant Testing, Sprinkler System Testing and examined screens as removed from two locations in the fire protection piping. Further, documentation of detailed cleaning packages for eleven Fire Protection Systems and the General Test Procedure GT-C-01, Revision 11 for flushing, and the YAEC-SQC Inspection Report Q-02-03-01 for QC inspection of disassembly, cleaning and reassembly of fire pump house piping (Work Requests FP 841 and IIL #FP-1037) was reviewed. These reviews verified that quality control inspections and operational tests were established and performed."

2.9.3 NRC Inspection

The piping inside the fire pump house was the pipe containing the microbiological induced corrosion which was disassembled and the piping taken outside the pump house for cleaning. The underground piping, which is cement lined, was not being removed for cleaning, but as part of a modification ongoing concurrently with the MIC work. See paragraph 2.12.3 for further discussion.

The YAEC-SQC inspection report Q-02-03-01 for QC inspection of disassembly, cleaning and reassembly of fire pump house piping is in reference to piping inside the fire pump house where microbiological corrosion caused a buildup on the inside of unlined pipe as discussed by the alleger in his transcript dated November 4, 1986. Other portions of the above paragraph, address portions of the fire protection system outside the fire pump house.

The original allegation in regard to the fire protection system was the concern that the volume of water would be inadequate as quoted below:

"The volume of the water from the fire sprinkler system is not adequate because the pipes are partially clogged from sitting for several years with water in them. When the fire sprinkler system was tested, it was only checked for pressure not for volume."

The inspection during 50-443/86-52 was directed toward observation of tests in progress, review of documentation of licensee identified problems with fire protection piping and corrective actions taken, with consideration given to the testing of portions of the fire protection system. The inspection work during 86-52 was supplemented by reviews, examinations and observations made during special inspections 443/85-06, 443/85-22 and 443/86-32 which were directed specifically toward the Seabrook plant fire protection system including equipment, procedures and staffing.

The fire protection piping located outside the fire pump house is cement lined piping per drawing F604052 as discussed in inspection report 86-52. Underground fire protection piping was not dug up to observe its internal condition during the period the alleger is referencing. However, flow tests to verify adequate capacity and pressure characteristics were observed and similar tests run previously by the licensee were also reviewed.

With respect to flow volume, page 26 of 50-443/86-52 reports the NRC inspector's observation of a hydrant test on November 4, 1986. This flow test had a water volume of 1753 gallons per minute (gpm) with the system pressure stable at 148 psi. For comparison, the Seabrook SSER 4, pages 9-10 and 9-11 defines the largest water demand for any fixed fire suppression system installed in a safety related area to be 1292 gpm.

2.9.4. Conclusion

The NRC staff has found the fire protection system water piping to be capable of providing both the volume and pressure necessary to meet safety related fire suppression demands. The claim of the alleger that the fire protection piping in the ground outside the fire pump house was not cement lined is incorrect. Underground fire protection piping at Seabrook is cement lined, with the exception of the recently installed plastic piping extending from the fire pump house to the general office building.

2.10.0. Technical Response

2.10.1. Source of Issue

The ELP Draft Response on page 19, 1st paragraph states; "Again, paragraph 1 does not provide us with any dates."

2.10.2 Scope of the Issue

The ELP is requesting the date on which the NRC inspector interviewed the Insurance Representative of the American Nuclear Insurers as discussed on page 26 of report 86-52.

2.10.3. NRC Inspection

The interview was conducted on November 4, 1986.

2.10.4. Conclusion

None

2.11.0. Technical Response

2.11.1. Source of the Issue

The ELP Draft Response on page 19, last paragraph states; "The NRC does not specify what activity it was involved in or what it observed that caused the NRC to conclude that the Fire Pump House was always kept locked when not attended".

2.11.2 Scope of the Issue

What is the NRC basis for concluding that the fire pump house is kept locked when not attended?

2.11.3. NRC Inspection

The fire pump house contains the pumps that supply water that may be necessary in the event of a fire involving certain safety related equipment. The fire protection water system may also be required to mitgate fire in non safety related plant systems or areas. As such, the licensee has required the fire pump house to be locked by plant security procedure SE 004. Additional details of the apsect of the security program will not be discussed, however during site inspections during the period of November 3 - 14, 1986, the inspector verified the locked condition of the fire pump house. Furthermore, during subsequent inspections at the site on April 6 - 10 and April 20 - 24, 1987, the inspector verified the fire pump house to be locked when unattended per this procedural requirement.

2.11.4. Conclusion

The NRC verified the fire pump house to be locked during inspection activities and determined that the plant security procedure requires the fire pump house to be locked when not attended. Due to the automatic start characteristics of the fire pumps, locking of the pump house doors is not an impediment to required fire fighting capability.

2.12.0. Technical Response

2.12.1. Source of the Issue

The ELP Draft Response on Page 20, 2nd paragraph states; "Allegations #6, #49, and #50 do not speak about corrosion, rather they speak about sedimentation, a completely different process. Furthermore, the NRC talks about preventing corrosion in the "unlined pipes," while the pipes in question were not unlined pipes."

2.12.2. Scope of the Issue

Why is microbiological induced corrosion (MIC) discussed in report 86-52?

2.12.3. NRC Evaluation

The original allegation in this area, dated October 3, 1986, as an affidavit stated a concern with sediment and pipe clogging but was not specific as to the type (lined or unlined) pipe that was affected.

During the time the alleger was present at the site (February 1986 to July 1986) two activities were in progress in the vicinity of the fire pump house. First, unlined pipe and elbows were being disassembled in the fire pump house to remove MIC deposits from the inside of the components. In some cases, the MIC deposits significantly reduced the inside diameter of the piping. Pipe and elbow sections of unlined pipe were taken outside the fire pump house for removal of the MIC deposits.

During the spring of 1986, the licensee added an extension of the fire protection system to the new site general office building (GOB). Extension of the fire protection system to the GOB required cutting a section of the 12 3/4 inch, outside diameter (OD) inground, cement lined pipe near the outside corner of the fire pump house and installing a Tee connection. On April 23, 1987, the

NRC inspector observed the section of removed 12 3/4 inch OD pipe noting the presence of a cement lining on the inside from 5/16 to 7/16" thick and an absence of either sedimentation or MIC deposits.

During the 11/4/86 interview with the alleger, the alleger indicated his concern with something growing inside the fire protection piping. The inspector investigated this claim and established that the licensee had found and corrected the problem of microbiological induced corrosion of unlined pipe as discussed in report 86-52. In summary, the NRC addressed the issue of corrosion because this is the problem of concern to the alleger. Sediment would be located on the bottom of pipes, not all around the pipe as observed by the alleger and discussed in the interview with the alleger of 11/4/86. With respect to sediment, report 86-52 states that water flow visible during sprinkler loop testing was slightly discolored but not indicative of significant sediment.

2.12.4. Conclusion

Corrosion and sedimentation of fire protection piping as applicable to lined and unlined piping have been examined. The Seabrook fire protection piping system has demonstrated both adequate flow and pressure capability. Periodic planned maintenance and testing of the system is fully expected to provide necessary fire response capability. It appears there has been confusion between microbiologically induced corrosion of unlined fire pump house piping and buried, cement lined piping.

2.13.0. Technical Response

2.13.1. Source of Issue

The ELP Draft Response on page 20 states that; "the NRC states that the pipes were removed or disassembled inside the fire pump house to remove MIC deposits and not disassembled outside the pump house where workers actually observed the sediment or clogging."

2.13.2 Scope of the Issue

At question is the relationship between pipe clogging and the original lined or unlined condition of piping.

2.13.3. NRC Inspection

Pipe inside the fire pump house is unlined steel pipe (See Photo No. 1). The main fire loops from the pump house, with the exception of the new loop to the GOB, are cement lined pipe (See Photo No. 2). Reference the following drawings.

9763-F-604058 Fire Pump House Piping Plan
9763-F-604068 Yard Fire Protection Diagram
9763-F-604052 Yard FP&PW Underground Piping

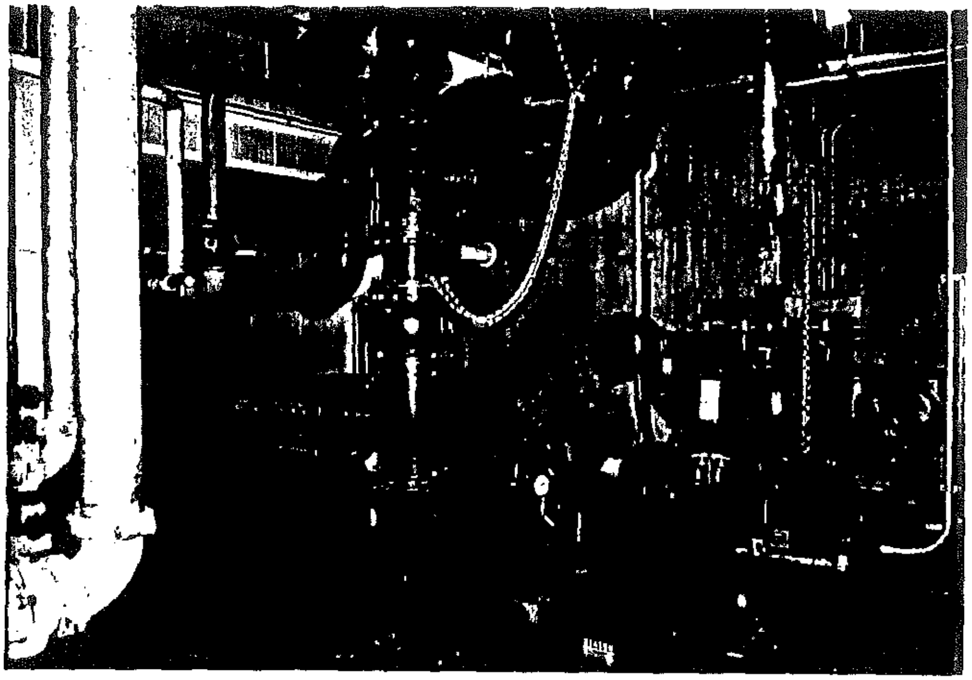


PHOTO NO. 1
FIRE PUMP PIPING

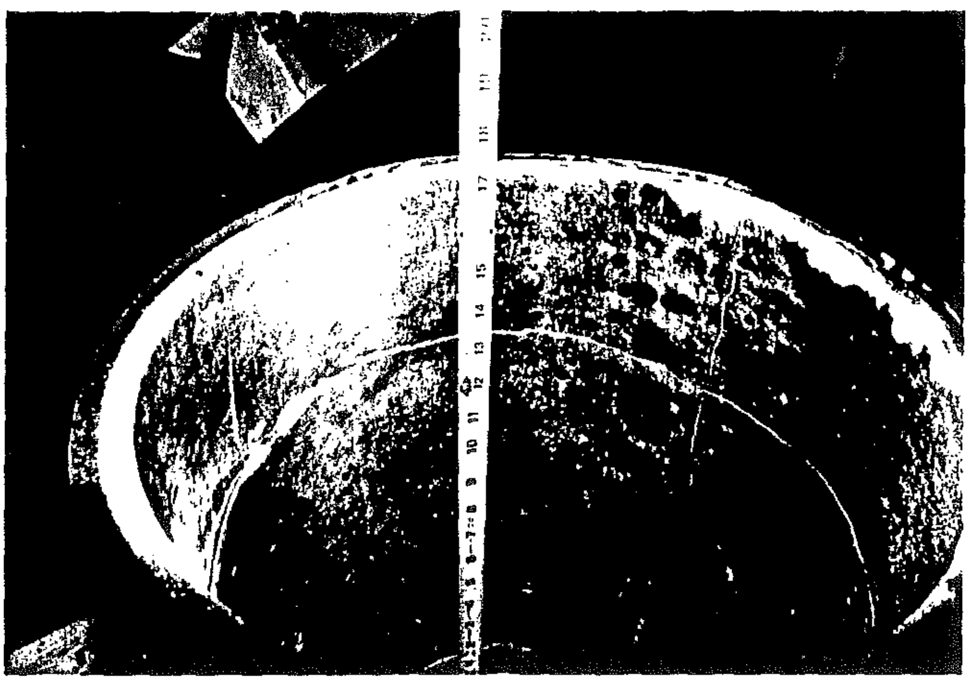


PHOTO NO. 2
CEMENT LINED FIRE PROTECTION SYSTEM PIPE
PHOTO NO. G

The unlined pipe in the fire pump house was found by the licensee to be subject to Microbiological induced corrosion (MIC) deposition and partial clogging as pointed out by the alleger. The unlined piping inside the pump house was disassembled, removed to the area outside the pump house and cleaned. This cleaning occurred during the period when the cement lined buried fire system piping was being excavated and the GOB modification installed. The alleger mistook the cement lined piping as being clogged when, in fact, the alleger observed the cement lining of the pipe.

The 86-52 inspection as supported by previous NRC fire protection system inspections has established that corrective actions including unlined pipe cleaning, chlorination and flow testing have provided for fire protection system design water flow rate capability. The licensee has in place a program for fire protection system surveillance and testing.

2.13.4 Conclusions

Contrary to the claim of the alleger, the NRC staff has found that cement lined pipe typical of that in use at Seabrook is not subject to MIC deposition or clogging. The level of sedimentation as evidenced during flow testing is insignificant. Flow tests witnessed by the NRC inspector which were typical of those performed by the licensee on 100% of the fire protection system have confirmed the fire protection water system to be able to provide water flow and pressure in excess of design requirements.

2.14.0 Technical Response

2.14.1 Source of the Issue

The Draft ELP Response, page 22, states:

"The fire in the electrical conduit caused by a cigarette was dismissed by the NRC with rather speculative language, "if a fire had occurred..." The alleger reiterates: a fire very definitely occurred. Is it possible debris or garbage had also fallen into the conduit and burned hotly enough to burn the cables? That possibility was not considered by the NRC in addressing this allegation."

2.14.2 Scope of the Issue

The original allegation stated, "A lit cigarette fell into a four inch conduit full of wires and cable and caught fire. Four or five gallons of water were required to put out the fire. The incident was not reported." This was addressed in section 19 of the inspection report 86-52.

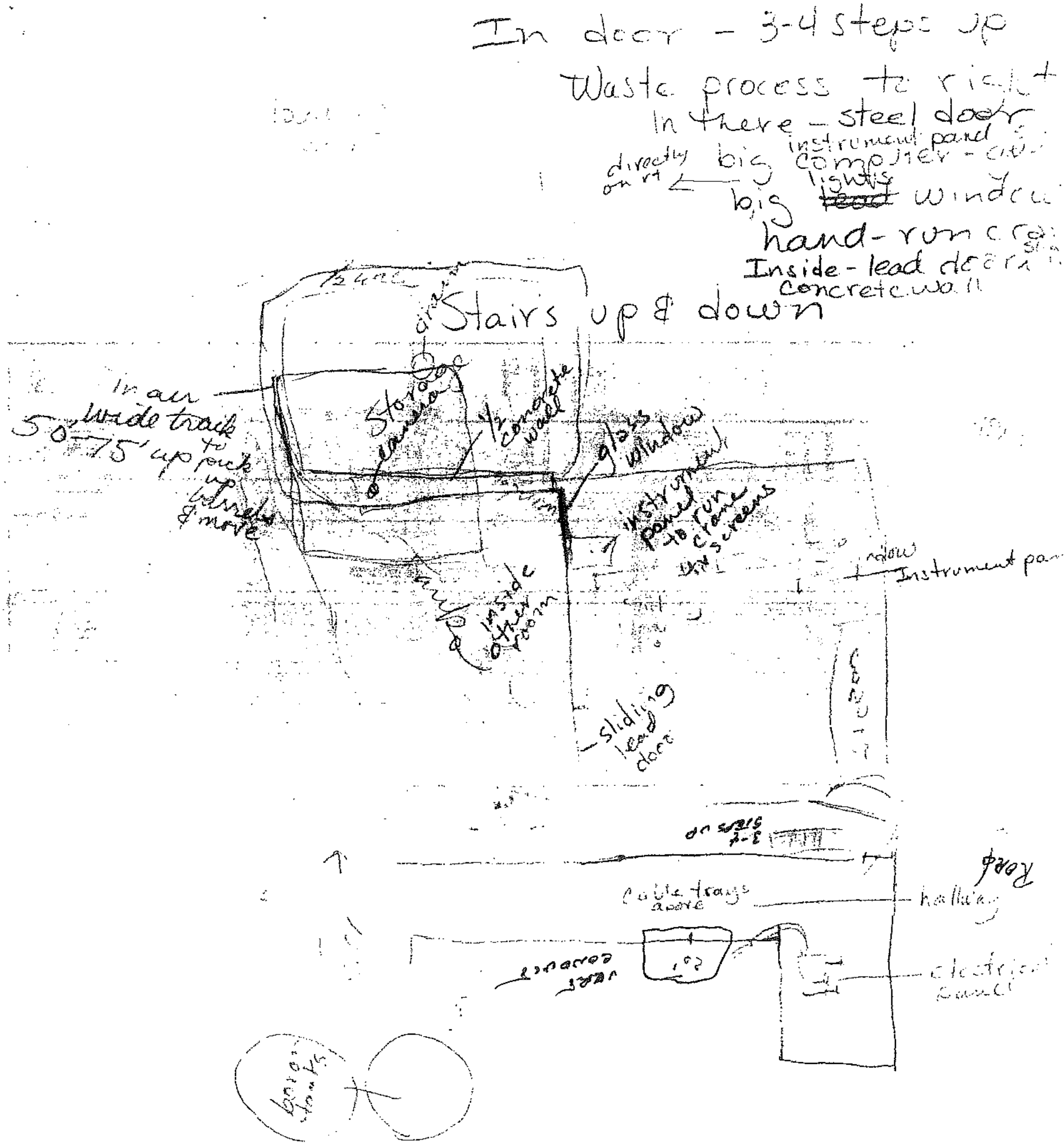
The NRC, provided the ELP with a plant drawing of the area where the alleged fire had occurred to assist the alleger in narrowing the area of concern. The alleger was unable to locate the area where the fire had occurred; however, he drew a sketch of a location that was ultimately identified as the waste processing building. (See Drawings 1 and 2)

The ELP speculated that debris, in addition to the cigarette, may have fallen into the conduit and created a hotter fire than the NRC postulated in the test.

2.14.3 NRC Inspection

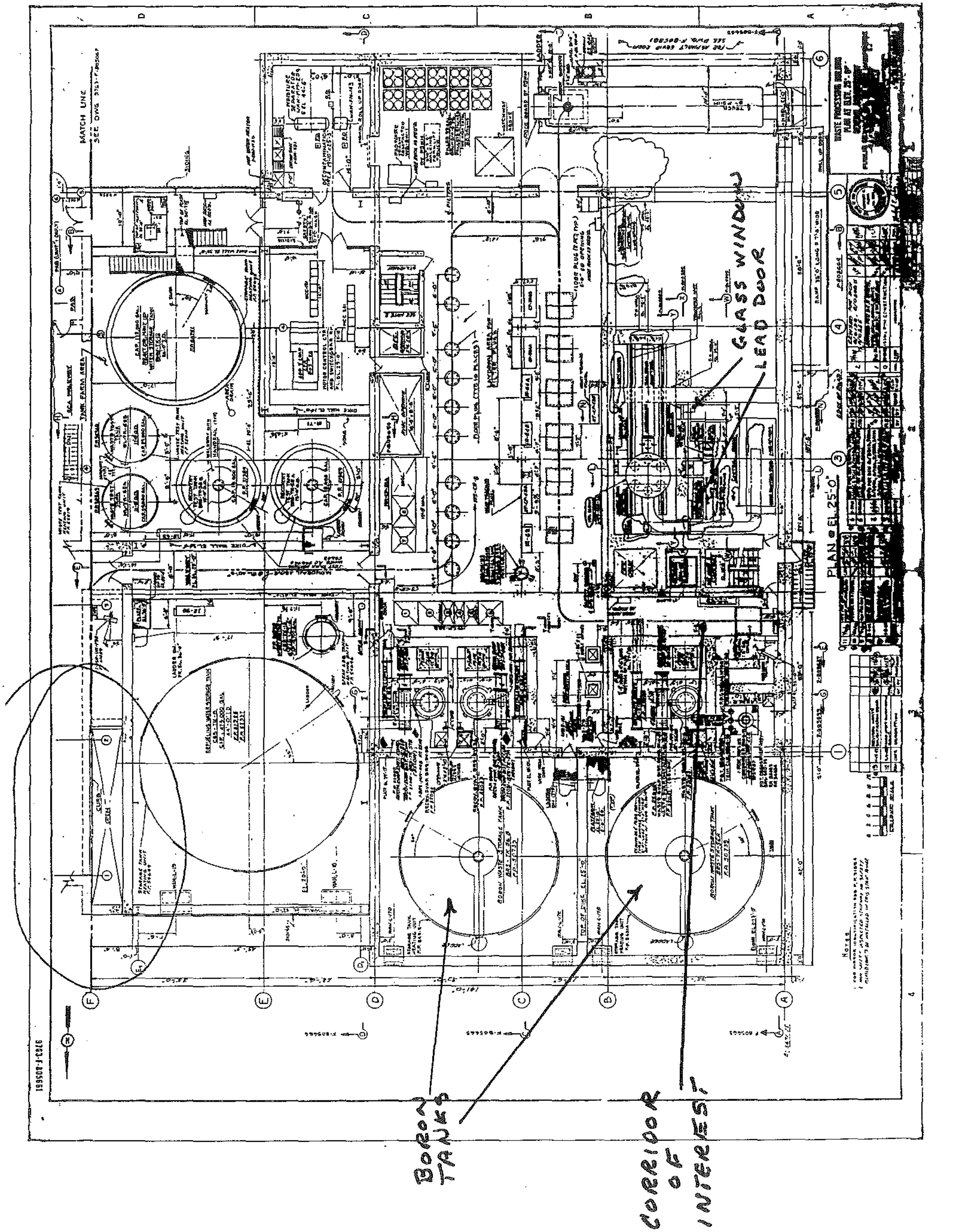
Subsequent to the April 20, 1987 meeting with the ELP, a second hand-drawn sketch of the location of the alleged conduit fire was provided by the alleger through the ELP (See Drawing No. 1). This sketch provided enough detail for the NRC to determine that the location was in the Waste Processing Building, elevation 25 ft., in a corridor running adjacent and parallel to column line 2 (See Drawing No. 2).

Discussions with the ELP disclosed that the subject conduit was vertically oriented and was approximately 20 ft. down the corridor on the left side. There are only two electrical conduits in the area described. These conduits service a nonsafety related lighting transformer ED-X-11C and its associated lighting panel L22 in the Waste Process Building. Photo No. 3 is a view along the waste processing building corridor oriented such that the left wall is the area of interest (Note: Photo is inverted). The area containing the "computer", sliding lead door and the glass window are on the right side in the photo. As can be seen from the photo, no conduits penetrate the floor slab in this area. Also, the distance from the position where the photo was taken to the wall mounted lighting panel is approximately 30 ft.



ALLEGERS' DRAWING

DRAWING NO. 1



PLAN @ EL 25'-0"
WASTE PROCESSING BUILDING PLAN AT ELEV. 25'-0"
MATCH LINE
TANK FARM AREA
MOTOR CONTROL CTR AND SWITCHGEAR RM
REFUELING WATER STORAGE TANK
BORON WASTE STORAGE TANK
FLOOR PLUGS
NOTES
GLASS WINDOW
LEAD DOOR
BORON TANKS
CORRIDOR OF INTEREST

DRAWING NO. 8

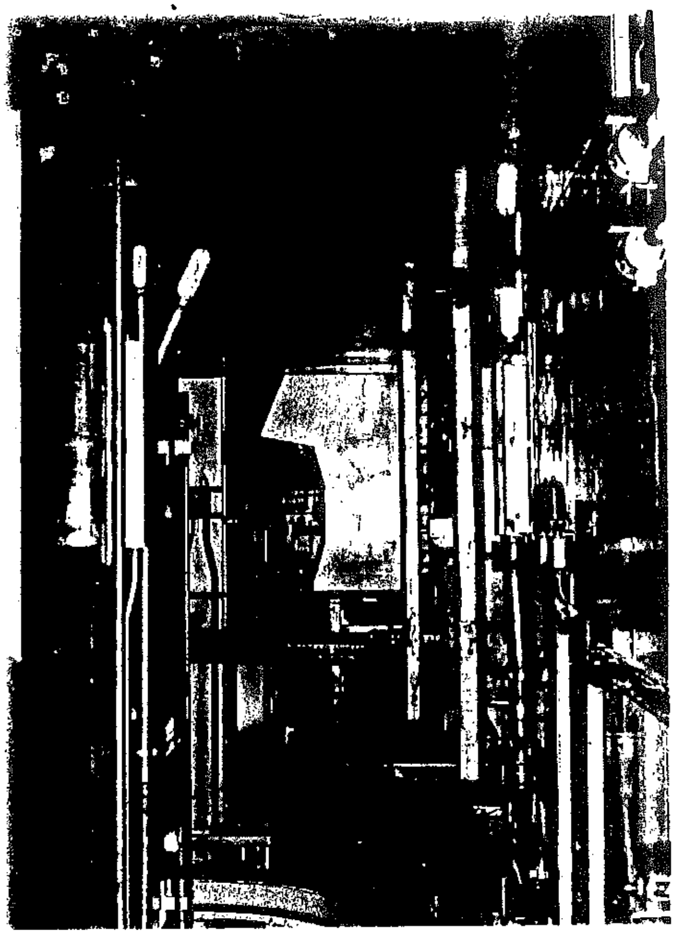


PHOTO NO. 3
VIEW OF CORRIDOR AT COLUMN LINE 2 IN
WASTE PROCESS BUILDING

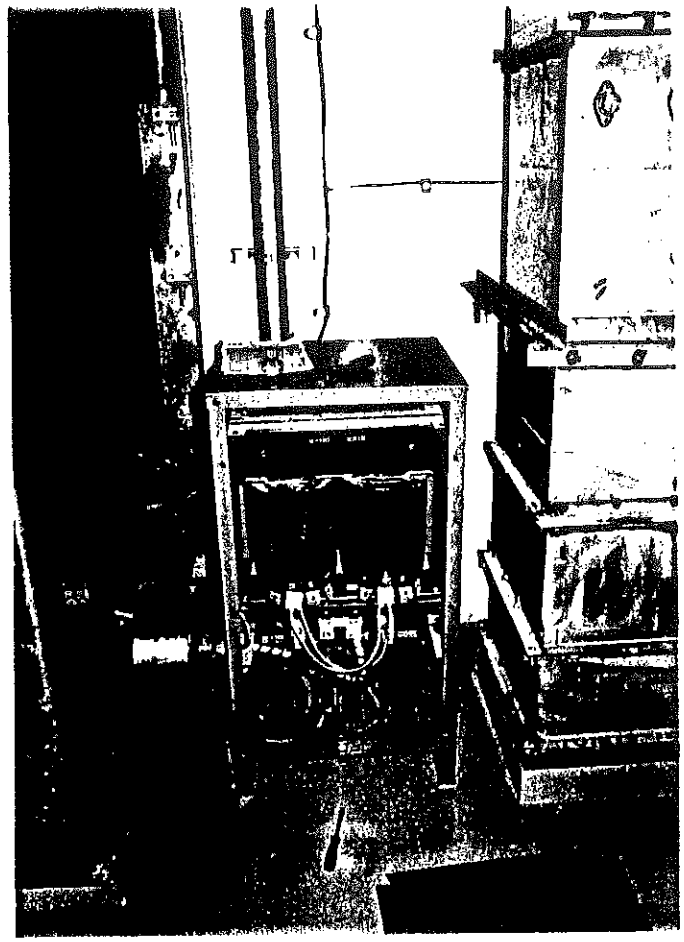
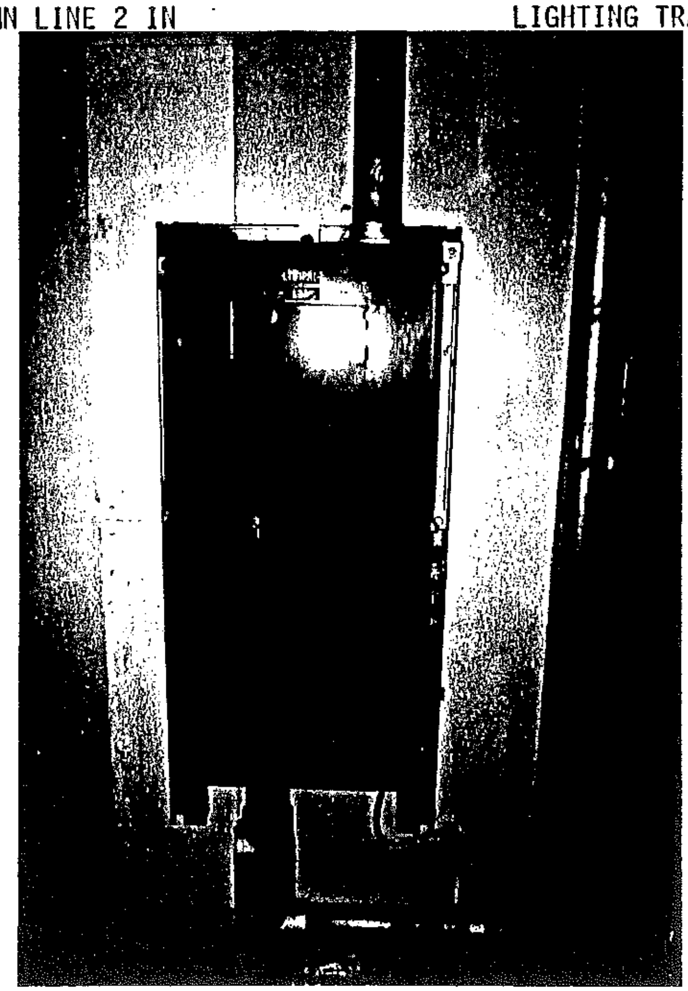


PHOTO NO. 4
LIGHTING TRANSFORMER 1-ED-X-11C



The licensee was requested to perform insulation resistance measurements on the transformer and lighting panel cables located in the two conduits. The test was performed on May 5, 1987 and witnessed by the NRC. The test results indicated that the insulation resistance exceeded the minimum values and was acceptable. Further, the NRC inspector requested that the conduit covers be removed for visual inspection to determine if there was any evidence of a fire. The visual inspection revealed that the conduits were clean and no combustion products were evident.

Lastly the licensee has also performed insulation resistance measurements and continuity checks of cabling and electrical equipment during the preoperational phase of the test program. Any degraded cabling or equipment, if degraded due to fire or any other cause, would have been identified during these tests.

2.14.4 Conclusion:

No physical effect of fire in the conduit is evident now; nor, has any electrical malfunction been identified to support the allegation. Further, all electrical equipment in the immediate location is nonsafety related.

Based on the tests performed on the cables in inspection report 86-52, the fact that all cabling is fire resistant and meets IEEE Standard 383-1974, the field tests of the cables alleged to have sustained fire damage and the licensee's preoperational test program, the NRC has no further concern regarding this matter.

2.15.0 Technical Response

2.15.1 Source of the Issue

The ELP Draft Response on page 24 states: "The alleger was concerned an accident, specifically a fire or a seismic event, could destroy both systems at once...."

2.15.2 Scope of the Issue

The above statement raises a question regarding the main feedwater and emergency feedwater systems. According to the understanding of the alleger both systems run together and are supported by common pipe supports and restraints which make it susceptible to failure from a common fire or earthquake.

2.15.3 NRC Inspection

The NRC staff examined the design basis and logic/flow diagram of the system. The as-installed piping configuration was also inspected. Based on this review, examination, and reinspection, the staff determined the following:

The emergency feedwater system consists of four feed lines supplied by two independent and diverse pump systems of one hundred percent capacity, i.e. each pump is capable of supplying the total feedwater volume required by design for safe shutdown of the plant. One pump is powered by an electric motor and the other with a steam driven turbine; thus, providing diversity in the power source to assure reliability. These pumps are housed in a separate location in the power block, away from the main feedwater pumps, in a noncombustible environment. Two lines from the common header of the emergency feedwater system travel through two different pipe tunnels of reinforced concrete construction. These pipe tunnels are also physically separated by considerable distance and are below grade with a noncombustible environment. The reinforced concrete structure and the carbon steel piping and supports are all classified as noncombustible material and are fire resistant. The NRC staff reviewed the plant's ability to safely shutdown in the event of a fire in this area. This review is discussed in NUREG 0896, Safety Evaluation Report Related to the Operation of Seabrook Station, Section 9.5.1 and examined the fire protection program, safe shutdown capability and control of combustibles. This review considered the proximity of emergency feedwater equipment and a fire that could affect this equipment. The licensee recognized the susceptibility of the emergency feedwater system to loss from a common fire and requested a deviation from the requirement for independence of the alternate shutdown capability for the fire area of concern. In the event of a fire in the emergency feedwater pump room, the startup feedwater pump will be used to supply water to the steam generators. This pump is located remotely from the emergency feedwater pumps and is provided with a class 1E electrical power source. It does, however, utilize the emergency feedwater flow control valves and transmitters which are located in the fire area of concern. The valves and transmitters are separated by sixty feet, are normally open and fail as-is (open) on a loss of power. Therefore, damage from fire is not a credible event for destroying the main, emergency feedwater and the startup feedwater systems.

The main and emergency feedwater piping systems are in the same area in the pipe tunnels, but it is not true that they are throughout supported by common pipe supports. There are some common supports and restraints, but these supports, guides, and restraints together with the pipe tunnel are designed to withstand the design basis earthquake, and remain functional during and after a seismic event.

Furthermore, two lines of the emergency feedwater system in each tunnel are fully capable of providing design flow of feedwater into two steam generators assuring a safe shutdown. This has been analyzed in the plant Final Safety Analysis Report, evaluated and found acceptable by the NRC.

2.15.4 Conclusions

The concerns regarding a fire or seismic event disabling the feedwater and emergency feedwater systems and preventing a safe plant shutdown are not valid. Damage from a fire is not a credible situation; the system is designed and installed to withstand a design basis seismic event.

2.16.0. Technical Response

2.16.1 Source of Issue

The ELP Draft Response on page 26, last paragraph states; "The NRC examined the feedwater heaters, not the heat exchangers, even though the alleger showed the NRC on their own drawings the exact system to which he was referring."

2.16.2 Scope of the Issue

Resolve the discrepancy in inspection report 50-443/86-52, on page 92 which presents findings relating to feedwater heaters when the alleger was concerned with a problem with heat exchangers located near the 50 foot elevation of the turbine building.

2.16.3. NRC Evaluation

The feedwater heaters are a type of heat exchanger and are located near the 50 foot elevation of the turbine building. There are a total of sixteen high and low pressure feedwater heaters in this area of the turbine building. Inspection Report 86-52, on page 92, notes that the high pressure and low pressure feedwaters were found to be installed level in accordance with the installation instructions. Six of the low pressure feedwater heaters CO-E-21 A/B/C and CO-E-22 A/B/C penetrate the condenser shell a distance of approximately 24'. (See Photo No. 6)

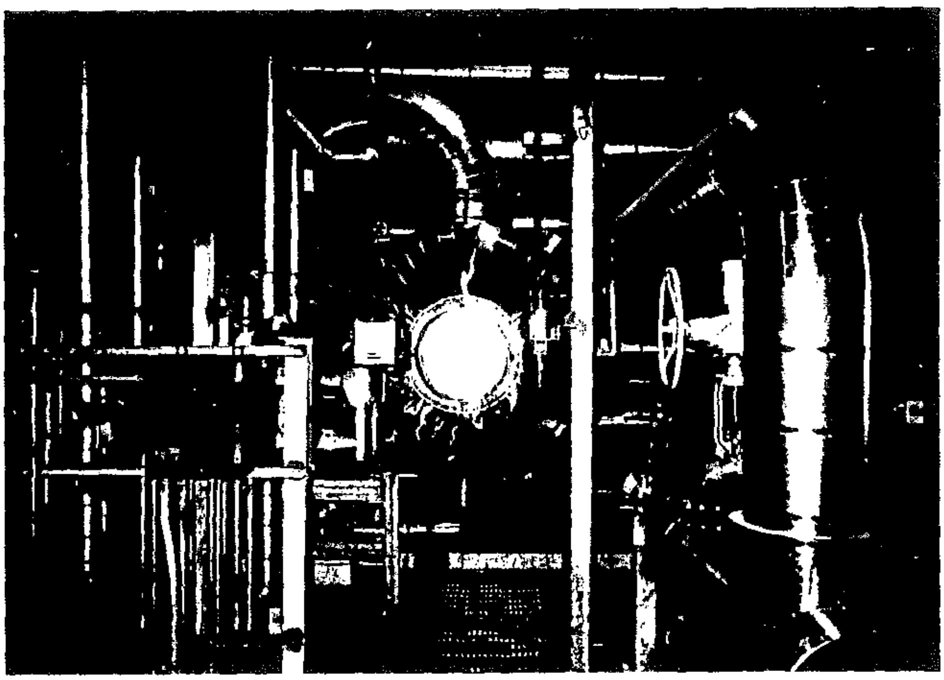


PHOTO NO. 6

TYPICAL FEEDWATER HEATER PARTIALLY
MOUNTED IN CONDENSER/TURBINE
EXHAUST TRUNK

During the interview with the alleger on April 20, 1987, feedwater heater CO-E-21B was identified by the alleger as having appeared to be out of level by 3 or 4 inches. He further stated that he determined the heat exchanger to be out-of-level by placing a liquid level on top of the heat exchanger shell.

On Wednesday April 22, 1987, the NRC Inspector entered the condenser and measured the level condition of each of the six feedwater heaters that penetrate the condenser including CO-E-21B. Measurements were made using a liquid level placed on the heat exchanger shell. Each feedwater heater as measured from inside the condenser was verified to be presently level within tolerance.

2.16.4. Conclusion

The allegation that heat exchangers (feedwater heaters) were installed with a slope when installation instructions required the installation to be level was not substantiated.

2.17.0 Technical Response

2.17.1 Source of the Issue

ELP response, dated April 2, 1987, states: "There are major unresolved cracks in Seabrook Station's Unit I containment, waste process building and equipment vault. The walls shrank or settled, causing cracking. Thus, because they are weak spots, any stresses on the walls are directed toward the cracks. The repairs made so far don't address causes of the cracking, and they merely plug the leak from inside."

2.17.3 NRC Inspection

The cause and effect of concrete cracks in the containment structure and their significance has been evaluated and documented in Paragraph 2.4 of this report. Chapter 3.0 of Seabrook Final Safety Analysis Report (FSAR), and NRC's Safety Evaluation Report (SER) extensively document the design basis and functions of containment. The containment has been tested for integrity and found to be acceptable.

With regard to the waste processing building and equipment vault, the NRC has extensively examined cracking in these buildings. So far, these examinations have not disclosed any major crack which may potentially affect the safety of these buildings.

However, the NRC has requested that the licensee perform additional assessments of the cracking in these structures to ensure the original assessments are valid in light of current conditions and aging. This item is unresolved pending the completion of the licensee's and NRC's reviews. (443/87-07-02)

2.17.4 Conclusions

No immediate safety concern exists in this area.

2.18.0 Technical Response

2.18.1 Source of Issue

The NRC staff performed a detailed evaluation of the concern relating to cold pulling of the safety related piping system during erection. The evaluation was in response to several allegations and statements contained in the ELP draft, supplementary responses, and the transcript of the NRC/ELP meeting of April 20, 1987.

The allegations contained in the ELP's April 2, 1987 supplemental submittal were:

a. Page 6, 3rd paragraph: "In the pipe slot, cold springing was used to fit-up pipes ranging from 24 inch to 42 inch diameters with lengths no longer than 14 feet"

b. Exhibit H, page 3: "In the pipe slot, 'cold spring' was used to achieve fit-up".

The allegations provided to the NRC during the April 20, 1987 meeting were:

a. In the tank farm area, a comealong was used to achieve fit-up between the segment of the CBS piping (4" O.D.) connected to the Refueling Water Storage Tank (RWST) and a valve on the system. The alleger indicated that the observed cold pulling of the piping, an amount in excess of 1", as observed from his location on a staging approximately 40 feet from the RWST. The alleger also indicated that the cold pulling incident took place in 1983 during the time he was employed onsite as a piping fitter/welder.

b. Clamps (Dearman) were used to correct cross-sectional ovality during fit-up of service water piping prior to welding. The alleger considered the practice of using the Dearman clamp as some type of cold spring which induces significant stress to the piping joints' welds.

2.18.2 Scope of Issue

Several statements were made in both ELP tansmittals and during the April 20th meeting which related to concerns involving the issue of cold pull. In order to adequately address and evaluate the significance of these concerns, the NRC staff reviewed references made to this issue which were documented in various transcripts of meetings, ELP transmittals, licensee investigations and NRC inspections of the allegations in this area.

2.18.3 NRC Inspection

The staff's evaluation of these concerns involved a detailed review of the various revisions of specifications and procedures addressing the installation, erection and inspection of safety related piping systems. It also involved the review of licensee and NRC correspondence in this regard throughout construction. Finally, the licensee's technical evaluations and corrective actions regarding identified nonconformances in this area were closely examined and evaluated.

Cold pulling is typically defined as the practice of pulling or jacking of piping to correct misalignment at the closure joint. It is implicit by this definition that, for the closure joint, the other ends of the piping segments are rigidly connected (by welding or bolting) to stiff piping headers, equipment nozzles, or support anchors. In Section III, Articles NB, NC and ND-4000, the ASME code provides the rules for fabrication and installation of piping systems. These, together with the related articles in subsection NCA on Quality Assurance and Inspection, provide the only control on actual construction. Though the code provides explicit guidance on fitting and aligning of piping ends to be joined by welding, it provides no specific rules regarding the issue of cold pulling.

In consideration of the above, the industry has adopted several approaches in addressing cold pulling of piping during construction. In all cases, however, the criteria are based on satisfying the design requirements specified in ASME Section III, Paragraphs NB, NC and ND-3611 which invoke the considerations for acceptability in the design of piping for nuclear power systems.

Thus, an assessment of this concern, and the contractor's activities in this regard, will focus on the acceptability of the installed safety related piping systems in terms of the design requirements setforth by the ASME Code and the design specifications and procedures used in the design process.

The following findings were based on the NRC staff evaluation of the allegations and concerns relating to cold pulling and the assessment of piping installation practices and its impact on the design requirements.

Several concerns were expressed regarding cold springing of piping during construction. Cold springing is defined as the deliberate incorporation of pre-stress in a piping system in order to reduce nozzle or anchor loads or to minimize creep. Cold springing is an acceptable construction practice covered by section III of the ASME code, articles NB, NC and ND-3672 for ASME piping classes 1, 2 and 3, respectively. UE&C addressed the methods for assembling piping spools in systems where cold springing was required by design and which was specified on installation drawings. These requirements were provided in the ASME piping installation procedure No. FPP-2. Since cold springing was not required or specified in any of the piping systems addressed in this allegation, it was apparent to the NRC staff that the allegers were using the terminologies of "cold pulling" and "cold springing" interchangeably without consideration of the technical difference. This apparent confusion was evident in the ELP letter to the NRC on April 2, 1987. The opening statement in the first paragraph under cold pull (page No. 4) stated "Cold pull, also known as cold springing is a prohibited practice, and is defined.... etc." (emphasis added)

Concerns relating to rigging of piping spools and the use of chainfalls, comealongs and jacks during fit-up, prior to welding, were expressed in many cases by the allegers in conjunction with cold pulling. The use of rigging equipment such as chainfalls and jacks to move and position heavy piping spools during erection is a typical construction practice and is acceptable. Only when mechanical devices are used to restrain piping prior to welding or bolting of closure joints, as defined by the explanation of "cold pulling", would the concern of cold pull be of significance. Except for one alleged cold pull incident near the RWST tank, none of the allegations presented by the ELP identified the cold pull as having taken place during fit-up of specific and identified closure joints of affected systems. Based on interviews between the NRC staff and the allegers, and review of numerous Engineering Change Authorizations (ECA's) and Non-Conformance Reports (NCR's) on misalignment of piping assemblies during erection, it was apparent that the construction and QC personnel were exceedingly sensitive about the issue of cold pull as a result of the repeated emphasis, by the contractor, against that practice during and after 1982.

The governing construction documents which addressed the assembly and erection of piping were: (1) UE&C Specification No. 9763-006-248-51 for assembly and erection of piping and mechanical equipment, (2) P-H Procedure No. IX-3 for fabrication and field installation specifications for nuclear power plant components, piping systems and appurtenances, (3) UE&C Procedure No. FPP-2 for ASME piping installations, and (4) P-H Inspection Procedure No. X-9 for inprocess inspections.

Prohibitions against cold pull specifically or the use of mechanical devices which provide restraint to piping joints during welding were stated in all of the above documents. The NRC staff found the above prohibitions to exist as early as 1978 in revision No. 1 of P-H's Inspection Procedure (X-9) above. Further, the staff also verified that the master check list for procedure X-9, which was used by the P-H field quality assurance group, had contained a specific check point for examination of joints before welding to insure the absence of external mechanical forces during fit-up. Review of qualification examination records for level II quality control personnel in 1981 provided evidence of the coverage of cold pull related issues in the training program.

Based on the above, the staff concludes that, from a programmatic view point, the governing construction and inspection procedures for safety related piping systems had provided coverage of the cold pull concern and that the required training of QC personnel in this area was evident.

Concerns were reiterated by the alleger regarding cold pull of service water piping, ranging in diameter between 24" and 42", as it enters the Primary Auxiliary Building (PAB) from the tunnel between the PAB and the Waste Processing Building (WPB). The segment of piping No. 1-SW-1821-1-L1-24"-33 (See Photo No. 7) in the PAB (addressed in NRC inspection report No. 84-12) was referenced by the alleger as an example during the NRC staff interview with alleger on April 20, 1987.

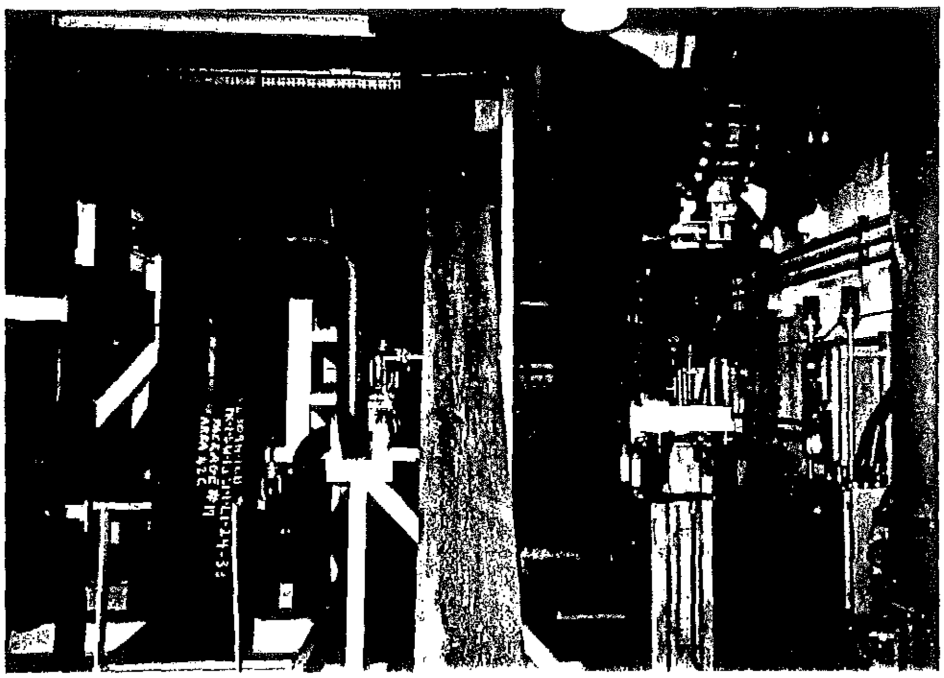


PHOTO NO. 7

PIPE SPOOL 1-SW-1821-1-L1-24"-33
AND
VALVE SW-V74 IN PAB TUNNEL

The NRC staff reviewed the licensee's program, conducted between 1985 and 1986, for the modification of polyurethane lined service water piping spools and valve liner-seats (section 2.19 of this report). The modifications involved disassembling 116 joints during the coating of thirty (30) valves, thirty-two (32) spools and fifty-four (54) flanged ends. The modification included valve No. SW-V74 connecting to the service water piping segment referenced in this allegation. Review of the modification records revealed no evidence of springing of piping spools during the unbolting of affected joints. The above finding provided reasonable assurance that, contrary to the allegation, construction cold pull was not induced into the service water piping during fit-up of closure joints.

A review of an alleged cold pull applied to the Containment Building Spray (CBS) piping No. 1-CBS-1222 from the RWST was documented in NRC's inspection report No. 84-12. The findings from that inspection were based on: (1) ovality checks and ultrasonic wall thickness measurements at several locations along the circumference of the piping, (2) verification of installed piping geometry and radii to the isometric drawings, (3) levelness check to determine if excess pipe roll had occurred, and (4) interview of the construction crew and the supervisor who were involved in the erection of the piping system. The NRC staff concluded, in inspection 84-12, that the piping geometry was in conformance with the design isometric drawings and that the wall thickness and ovality, at piping bend locations, were within the limits of section NC-4223 of the ASME code. During this inspection (50-443/87-07), the NRC staff re-examined the installed piping segment from the RWST No. CBS-TK-8 to the ASME Code break at valve No. CBS-V-0034 and the remainder of the system (B31.1) to the wall penetration. (See Photo Nos. 8&9) The staff also examined P-H's field process sheets for the installation of the piping segment of concern. The weld process sheet for field weld No. F0706, between the piping spool and the valve, indicated that inspection of the joint fit-up and the final weld were completed on March 1983. According to the process sheet for the flanged joint (mark no. 1222-JTR-0801, between the piping spool and the RWST tank, all eight (8) bolts in the joint were replaced and upgraded in July 1985. Records of the joint fit-up revealed no evidence of springing of the pipe spool during the unbolting of the flange.

Based on the previous and current NRC findings, the staff concluded that the allegation of cold pull of the CBS piping was unsubstantiated.

Review of activities related to the use of the "Dearman" clamp in the process of pipe fitting was conducted to determine whether: (1) the practice was consistent with the ASME requirements and described in applicable specifications and procedures; (2) it was used beyond the specified requirements and manufacturer's recommendations; and (3) its use could potentially result in overstress of the piping or weld joints.

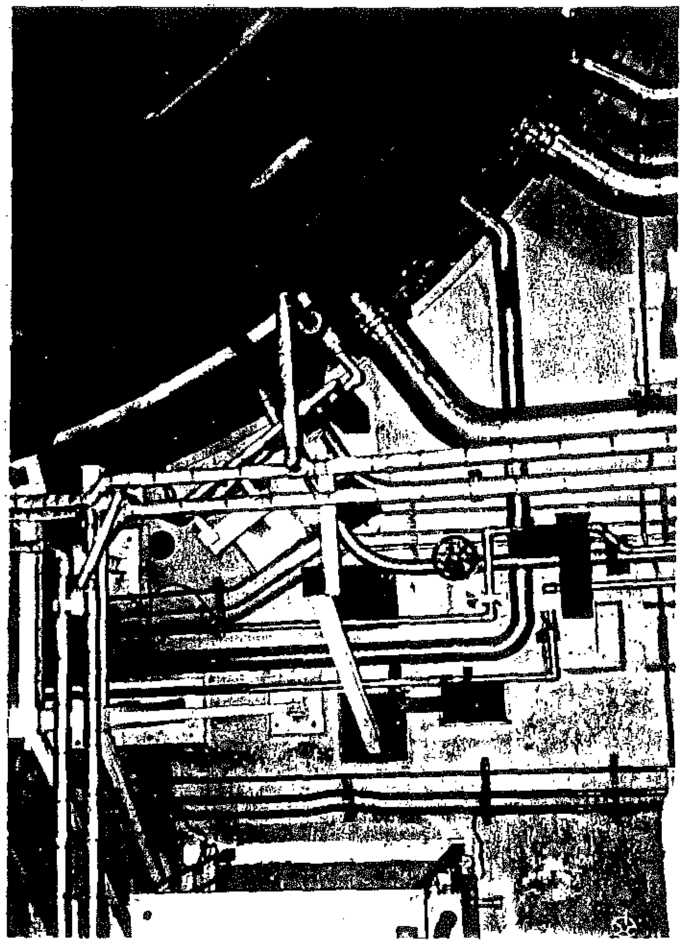


PHOTO NO. 8
VIEW FROM PLATFORM OF RWST-TK-8 PIPE SEGMENT AND VALVE CBS-V-0034

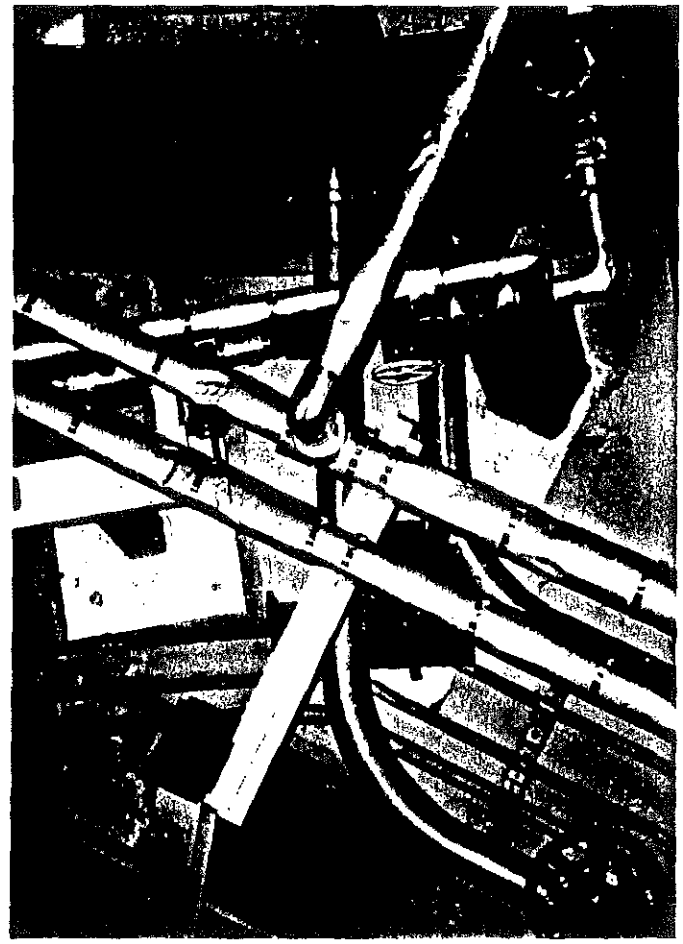


PHOTO NO. 9
CLOSEUP OF PHOTO AT LEFT, SHOWING VALVE CBS-V-0034 IN LOWER RIGHT CORNER

The use of bars, jacks, clamps or temporary attachments in the fitting and aligning of piping joints during welding is permitted by sections NB, NC and ND-4231 of the ASME code. UE&C specification No. 9763-006-248-2 identified the piping material for the service water system (up to 42" O.D) as SA-155 carbon steel. According to the ASME piping material specification, a variation of 1% is permitted due to out-of-roundness of the cross-section. This is equivalent to a maximum of 0.42" mismatch of the adjoining ends of a 42" O.D. piping. A piping spool's conformity to the above tolerance is typically verified during the receipt inspection. The tolerances for piping buttweld mismatch are specified in section ND-4233 of the ASME Code as 1/32" for joints aligned concentrically and 3/32" for joints aligned eccentrically. To meet the latter requirement, the inside diameters of the adjoining joints were typically counterbored or ground to the extent which would not encroach on the minimum specified wall thickness of the piping system. After piping counterbore, inside diameter mismatches exceeding the above limits, were typically adjusted to within tolerance during the fit-up process by the use of the Dearman clamp. Hoop stresses induced as a result of joint fit-up are generally insignificant in comparison to the residual stresses resulting from the shrinkage of the weld metal. Because of the clamp's proximity to the welded joint ends (2" on average), the heat from the welding process further reduces the clamping stresses by approximately 30%. UE&C's specification for assembly and erection (9763-006-248.51), and P-H's procedures for fabrication and installation (IX-3) and in-process inspection (X-9) were found to be consistent with the requirements of sections NB, NC & ND-4231, 4232 & 4233 of the ASME code.

Details depicting the tolerance requirements for buttweld alignment and mismatch were provided in P-H's General Welding Standard (GWS-III) and drawing No. 5000-F-1382 for standard weld end preparation details for piping. Specific details for preparation of cement lined (service water) and non-ferrous piping end joints were provided in UE&C's drawing No. 9763-D-804998.

Based on the details described above, the staff concluded that the practice of using the "Dearman" clamp for fit-up and alignment of piping joints during welding was in conformance with the ASME requirements and typical industry practice. The concerns regarding induced hoop stresses in piping joints due to clamping tools during welding were judged to be insignificant relative to overall piping system stresses resulting from fabrication, installation, and design loads.

The general concern regarding cold pull of safety related piping during erection was initiated as a result of the licensee's investigation of Nonconformance Report (NCR) No. B0749. The NCR pertained to an identified violation of P-H's procedure X-9 during fit-up of field welds F0105 "A" and F0106 "B" in the Main Steam piping. The identified nonconformance led to the issuance of Construction Deficiency Report CDR-32-00-13 which was addressed in the licensee's final submittal to the NRC (SBN-869). The licensee's corrective actions and other NRC inspections in this area were addressed in NRC reports No. 85-25 and 86-52.

An assessment of the requirement established by UE&C regarding the practice of cold pull throughout construction, the significance of the cold pull concern, and subsequent corrective actions undertaken by UE&C and the licensee provided the primary impetus for the NRC activities in this area during this inspection.

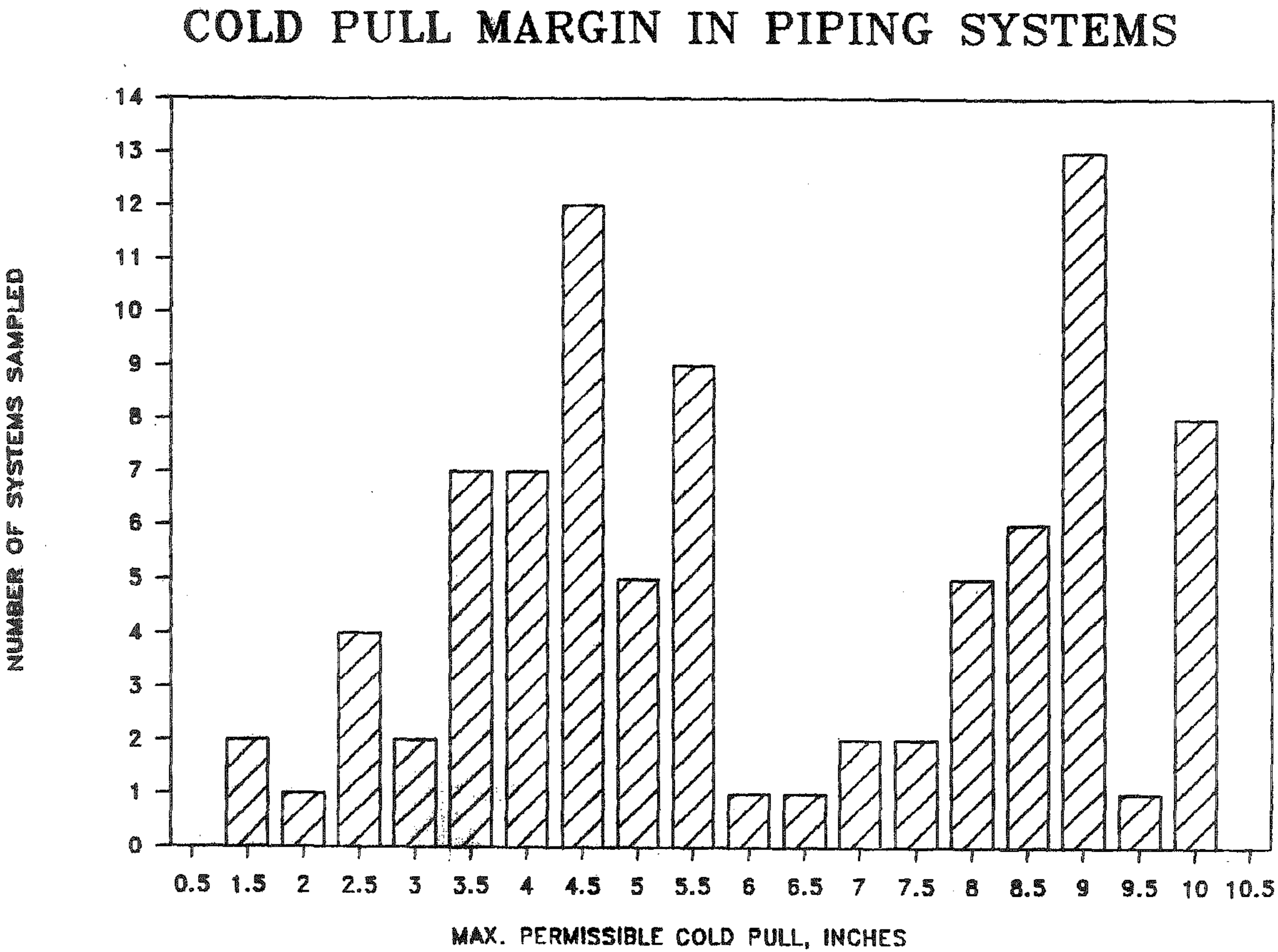
Upon review of the requirements specified by UE&C in the specification for assembly and erection of piping (9763-006-248-51), it was evident that the approach adopted by UE&C was overly restrictive from a practical construction viewpoint. Not until the 19th revision of the specification, in March 1986 did the contractor adopt more realistic requirements regarding cold pull. The 1/8" tolerance specified for gaps and misalignments during fit-up of closure joints was found to be extremely conservative with regard to piping stress requirements and available design margins. Results of an audit performed by INPO of design activities concurred with the NRC's observation. Although in the April 2, 1987 response, ELP asserted that cold pull was a serious problem; the staff found substantial evidence to the contrary. Many of the change documents (ACN's, FCR's & NRC's) issued regarding cold pull and alignment were, for the most part, a result of the above stated overly restrictive construction tolerance requirements. Examination of the revised cold pull requirements specified by UE&C and other approaches documented in several ASME papers and adopted by industry provides evidence regarding the amount of conservatism built-in the original UE&C requirements.
Understanding of the piping assembly and erection process, which relies on the use of temporary gravity supports, it is inituitive that long spans of piping can accommodate cold pull well in excess of the originally specified 1/8" with a minimal amount of induced longitudinal stress.

The sensitivity study performed by UE&C for the evaluation of effects of cold pull on piping systems installed prior to the issuance of CDR-83-00-13, provides some insight regarding the available stress margins in the design of these systems. The study was initiated upon identification of a potential for cold pull of up to 1¼" in piping systems as a result of P-H's misinterpretation of UE&C's specification requirement. The study involved seventy (70) piping subsystems which represented approximately 30% of the piping installed prior to January 1983. Conservative stresses per unit of cold pull displacement were derived on the basis of sampling of relatively stiff piping systems. Based on the existing design stress and the corresponding stress limit allowed by the ASME Code, an allowable displacement for misalignment was calculated for each subsystem.

Review of the calculated allowable displacements indicated a substantially larger cold pull margin in the majority of the lines evaluated, than the maximum potential cold pull of 1 1/4". Only three lines required specific evaluation for determining their allowable cold pull margins. A summary of the sensitivity study is shown in Drawing No. 3.

The above results provided a very high level of confidence (95%) that the potential of cold pull in piping systems installed prior to January 1983 is of no consequence regarding compliance with code design stress limits and overall system reliability.

The licensee's corrective actions regarding prohibition of cold pull, in piping systems erected after January 1983, was considered by the staff to be effective and conservative. The specification for piping assembly and erection by UE&C and the procedure for fabrication and installation by P-H were both revised to


COLD PULL MARGIN IN PIPING SYSTEMS
NUMBER OF SYSTEMS SAMPLED
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
0.5
1.5
2
2.5
3
3.5
4
4.5
5
5.5
6
6.5
7
7.5
8
8.5
9
9.5
10
10.5
MAX. PERMISSIBLE COLD PULL, INCHES

DRAWING NO. 3

include more explicit and restrictive requirements against cold pull and misalignment. These stringent requirements led to an apparent overreaction by P-H's QC personnel which in turn resulted in a large rejection rate and rework of otherwise acceptable welds. To rectify the problem, UE&C developed a weld repair procedure, "Pull Weld", which was subsequently incorporated in P-H's standard weld repair procedure No. JS-1X-14. Finally, in March of 1986, U&EC adopted a more realistic approach on cold pull which involved the specification of permissible cold pull for various piping sizes and spans. The piping lengths were based on minimum distance between closure and fixed (full restrained) joints. The staff found UE&C's formulation for the determination of permissible cold pull to be more conservative than acceptable industry standard.

2.18.4 Conclusion

Based on the above findings, the staff concluded that the allegations regarding cold pull of safety related piping at the Seabrook Station were unsubstantiated. The allegations and concerns were, in almost all cases, non-specific and the result of a lack of understanding of the technical issue. Furthermore, the staff determined that the requirement, regarding cold pull and alignment of safety related piping were generally conservative, in compliance with the design rules of the ASME code, and consistent with accepted industry practice and standards.

Contrary to the ELP's assertions, the staff concluded that the cold pull issue presented no concern regarding the reliability or the ability of safety related piping systems to perform their intended function.

2.19.0. Allegation

2.19.1. Source of Issue

Page 7 of the ELP letter (Tracy to Durr) dated 4/2/87 states;

"...One informant recently said there is grit in the valves of the service water system.

2.19.2. Scope of the Issue

Determine if grit is evident in the service water system and would grit damage the service water system valves.

2.19.3. NRC Inspection

The NRC report 50-443/86-52, on page 29, noted that recent debris removed from service water strainers contained grouting compound fragments but in negligible amounts.

On 4/23/87, during inspection 50-443/86-52, the NRC inspector observed the removal of the service water strainer 1-SW-S-11 and examined the debris in this strainer. The strainer contained 108.75 grams (4 ounces) of wet debris consisting of a few mussel shell pieces, seaweed, one piece of duct tape approximately 2" X 3" and three small fish. Considering the fact that the service water system's source of water is the ocean, some debris, including sand, would be expected. No fragments of cement, grit or grout were found in the strainer.

Considering the flow rate through the strainer to be 10,500 gallons per minute for several months, the amount of debris is insignificant. With respect to the integrity of the service water cement lining it is important to note that no cement fragments were in the strainer.

The service water valve seats are lined with a proprietary molecular polymer elastic (rubber like compound) and as such are not sensitive to wear or erosion by grit. During June 10 - 11, 1987, a specialist inspection was performed of licensee activities in evaluation and repair of the safety related service water 24 inch diameter butterfly valves. On May 18, 1987 the service water valve 1-SW-V15 was noted to not close completely. Examination of the valve showed the presence of wear through the valve body liner elastomer at the area of maximum valve disc to elastomer interference and lift off of the elastomer in this vicinity. The inspection established that the licensee has in place a program to thoroughly evaluate the problem and take corrective actions including elastomer application process control, compensation for interference in the stem area and measurement of loading during valve cycling. Evidence of service water pipe cold spring or significant debris in the service water piping were not observed by the NRC inspector in areas where valves were removed. Observation of the service water valve repairs and related activities will continue to be done by the NRC resident inspectors with engineering specialist assistance as appropriate.

2.19.4. Conclusion:

Strainer examination indicated no grit or cement fragments in the service water system. Grit if present would have minimal effect on the valve sealing surfaces as the valve seats are lined with an elastic material.

2.20.0. Allegation

2.20.1. Source of Issue

Page 7 of the ELP letter (Tracy to Durr) dated April 2, 1987 states;

"A recent problem presented to the ELP describes poor welds in the service water lines due to porosity and mismatch. The problem, described in more detail in Exhibit I..."

"This individual can specifically identify 3 bad welds in the service water system which should not have passed inspection, but which did."

2.20.2. Scope of the Issue

Determine the extent of alleged "bad welds" in the service water piping and evaluate the safety significance of this bad welding.

2.20.3. NRC Evaluation

During the interview of April 20, 1987, the alleger indicated that he did not have the specific weld numbers to identify the three bad welds. However, he did say that these bad welds were in the 24 inch diameter service water lines along the site "50 yard" line outside the Unit 1 waste process building (WPB). The extent of bad welding on each of three welds was identified as a maximum of one foot of circumferential misalignment or offset of pipe edges not exceeding 1/4 inch and pinhole porosity extending 4 inches either side of the overhead or six o'clock position. In the area of pipe misalignment, the ends of the pipe were stated to have been welded together, but not welded to the backing ring, such that the full pipe thickness was welded.

The 24 inch cement lined pipe in this area of the service water piping is SA106, grade "B" STD wall (0.375"), per UE&C specification 9763-006-248-2. The FSAR, table 9.2-2 shows the service water piping design pressure to be 150 psig, with the ASME Code Section III, Class 3 applicable to seismic category 1 sections. The ASME Code for Class 3 piping allows magnetic particle examination (MT) of the weld surface as the nondestructive examination method. The NRC inspector examined field weld process sheets for welds F0201, 03 and 05 on SW 1812-02 and weld F0305 on SW 1801-03. These welds are typical of field welds on 24 inch diameter service water piping along the outside of the WPB. These field weld process sheets show the fitup and tack inspection, welder symbols, root pass welding, completion of welding, visual inspection of welding and MT examination of the completed weld. The process sheets indicate the completion of required QC inspection, nondestructive inspection and ASME Code authorized inspector review and inspection. Each of the service water welds sampled was noted to meet the ASME Code requirements including field inspection at the time of welding. The NRC inspector reviewed radiographs taken of service water piping welding, in particular weld F0307 on Line 1802, noting the presence of one porosity pore that was repaired on NCR 237.

A calculation of wall thickness required due to internal pipe pressure of 150 psi per the ASME Code, paragraph ND 3641, indicated a requirement of 0.149". The actual pipe wall thickness is .375", providing a significant margin of extra thickness. Localized areas of pinhole porosity or misalignment of pipe edges where the pipe is welded through the full thickness are not considered to significantly affect the service life of the piping or present a safety problem provided the piping has passed the ASME Code hydrostatic pressure test.

Further, this allegation or ones very similar in nature were made to the NRC in 1984 by the same individual. This was thoroughly inspected and reported in Inspection Report 50-443/84-12.

2.20.4. Conclusion:

The alleger could not identify specific service water welds with "bad welds". The NRC inspector reviewed field documentation of fitup, welding and inspection performed on service water welds in the area of concern. The actual pipe thickness was compared to the thickness required by the service pressure and the effects of porosity and pipe edge localized misalignment on the potential for piping failure were considered.

The NRC staff concluded that the conditions described by the alleger, if true, would not be cause for further review or evaluation and would not reduce the service life of the service water piping.

2.21.0 Technical Difference

2.21.1 Source of Issues

The ELP supplmental response of April 2, 1987, states: "It is possible then to subject the steel reinforcement bars to continual groundwater contamination, resulting in swelling and oxidation of the rebars...."

"It is important to note that in/inspection 50-443/84-12/01 the concerns relating to possible future changes in ground water chemistry and the changes and affects this may cause in the reinforcing steel bars was still a cause of concern and considered an unresolved item..."

2.21.2 Scope of Issues

The issue here is the potential for corrosion of the reinforcing steel (rebar). The first issue postulates, such severe corrosion that there will be large positive volume changes in the rebar which will degrade the concrete.

2.21.3 NRC Inspection

Prolonged exposure of rebar to groundwater has always been a matter of concern to the NRC. This concern has been documented in several previous reports. However, the NRC has determined that currently changes in groundwater chemistry are not noticeable such that it is a safety-concern. A more detailed evaluation of this concern is documented in item 2.4.3.2 of this report.

2.21.4 Conclusions

None, see section 2.4 of this report.

2.22.0. Technical Response

2.22.1. Source of Issue

Exhibit H, page 3 attached to the ELP letter (Tracy to Durr) dated 4/2/87 stated;

"During my telephone conference with the NRC I referred to the "pipe slot". This is an area between WPB and PAB, not between WPB and DGB. Therefore the inspection was conducted in the wrong location."

2.22.2. Scope of the Issue

Determine where the pipe slot is and verify inspection has been conducted in the proper area.

2.22.3. NRC Inspection

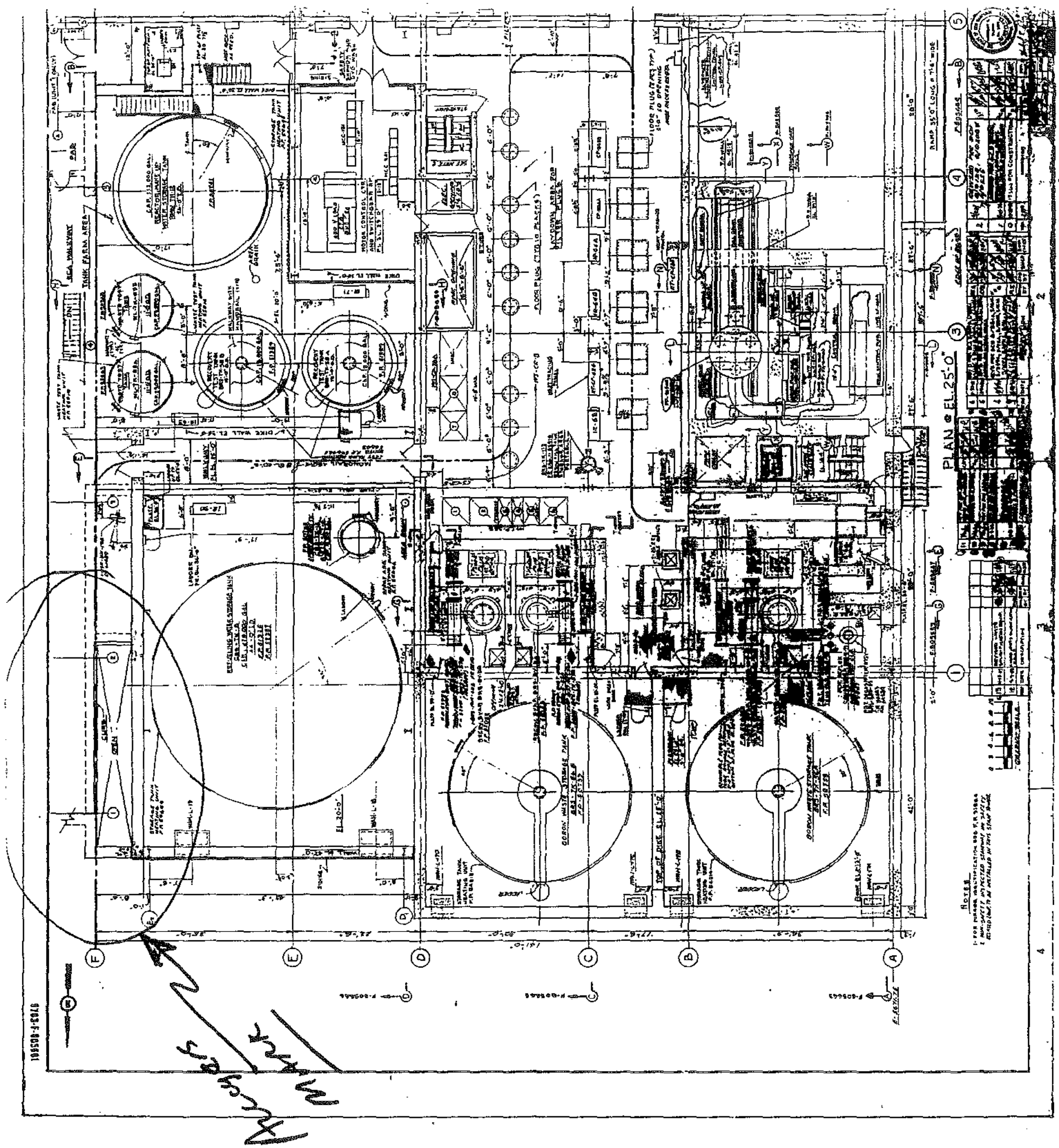
Inspection Report 443/84-12 on page 14 relates the following:

"During a telephone conference with the alleger on August 24, 1984, the alleger referred to the "pipe slot" area as the area between the Waste Processing Building (WPB) and the Diesel Generator Building (DGB)."

The service water piping from the area designated as the pipe slot by the alleger in 1984 enters the plant in an area between the waste process building (WPB) and the primary auxiliary building (PAB) designated the piping tunnel on drawing F-805660. The alleger confirmed during his interview on 4/20/87 that he had a concern with pipe cold pull in the pipe tunnel and designated the specific area on a drawing the NRC staff presented (see drawing no. 4). In particular he said his concern was with the pipe section shown in photograph, figure 5-1 of Inspection Report 84-12. Inspection 84-12 also evaluated cold pull controls on piping in the WPB-DGB pipe slot area as well as in the area shown by the photograph figure 5-1 of Inspection Report 50-443/84-12 (see photo No. 7 of this report). During inspection 87-07 the inspectors verified that numerous flanged joints in the service water system for valves, pipe spools, and other flanged system connections had been unbolted and reassembled further demonstrating that residual stresses from cold pull were not a problem with the service water system.

2.22.4. Conclusion:

Through discussion with the alleger on 4/20/87 and review of inspection documentation the NRC staff has established that inspections were directed toward the areas of concern to the alleger.



DRAWING No.4

2.23.0. Technical Response

2.23.1. Source of Issue

Exhibit H, page 3 attached to the ELP letter (Tracy to Durr) dated 4/2/87 states:

"In Inspection No. 50-443/84-12 the NRC used Centriline Process Booklet on cement lined pipe. A deflection of a 72" X 5/16" pipe with a 1/4" lining was deviated 13" without impairment to the lining. They neglected however, to state how long the spool piece was, and over what distance the deflection was absorbed."

2.23.2 Scope of the Issue

The ELP is requesting information on the length of the deflected test cement lined pipe and the significance of this dimension.

2.23.3. NRC Inspection

Page 18 of the Centriline Process Booklet includes a photograph of the deflection test in progress, the length of the test piece in the photograph appears to be between 3 and 6 feet. However, the actual length is not of significance since the deflection is measured across the pipe cross section. This may be pictured by looking at the end of a cylinder, such as a round tin can, without a top or bottom and pressing the cylinder from round to oval.

2.23.4. Conclusion:

The length of the pipe was not a significant factor in illustrating the resistance of cement lined pipe to cracking or breakage on deformation.

2.24.0 Allegation

2.24.1 Source of the Issue

Allegation as stated by ELP in the April 2, 1987, Update, Exhibit H, Page 5.

"... the portable radiographic sources used in X-ray examinations at Seabrook are incapable of giving decisive renderings of base metal with thicknesses between 3 and 6 inches."

2.24.2 Scope of the Issue

The issue stems from the allegation presented to the NRC and resolved in inspection report 50-443/84-12. The alleger reported a cladding separation in the lower head nozzles of the steam generators. This was examined by the inspection team in 1984 and resolved in paragraph 14.

Subsequently, the alleger expressed a concern that the radioisotope used to radiograph the nozzles was inappropriate.

2.24.3 NRC Inspection

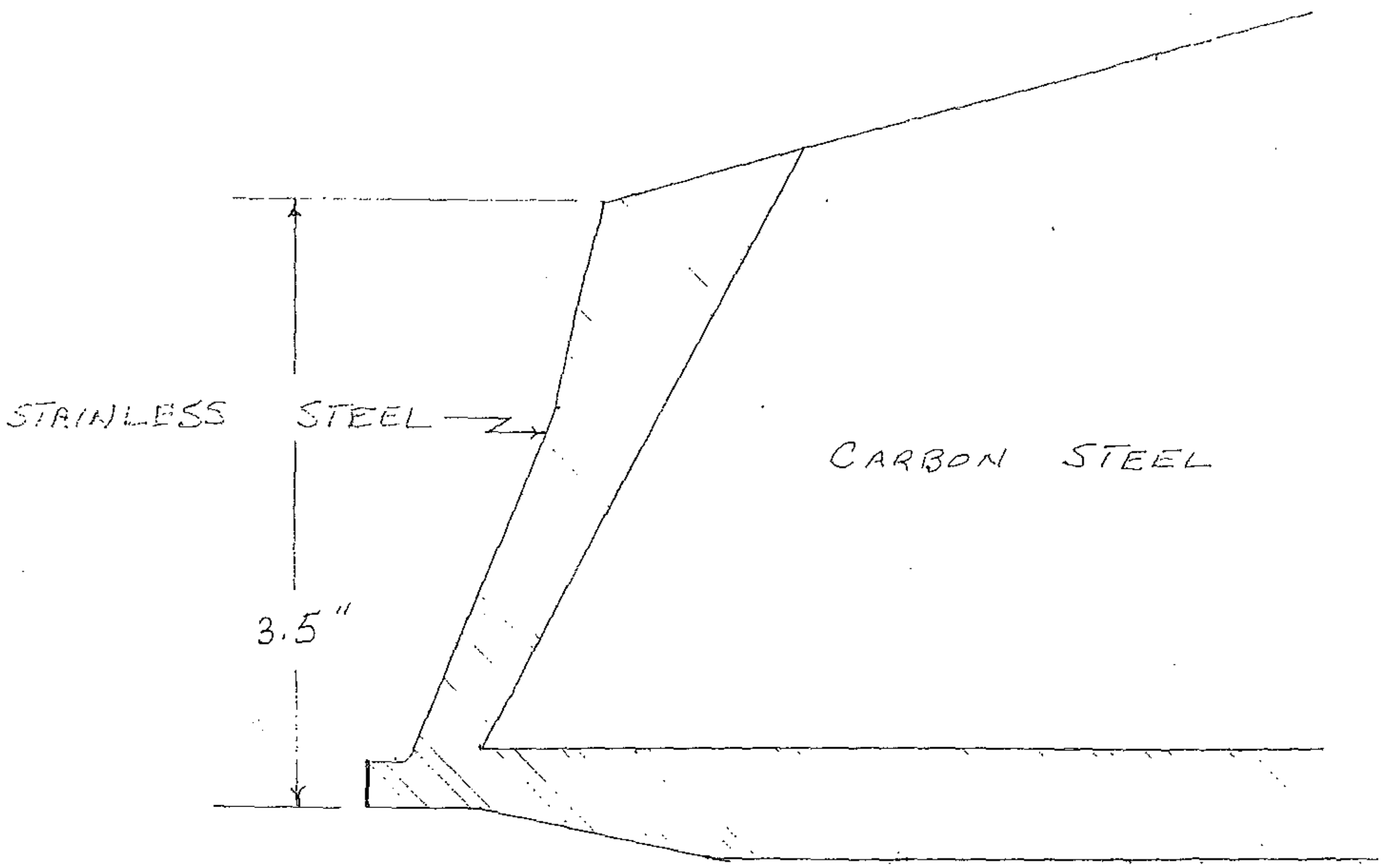
The steam generator nozzles at the location in question are approximately 3.5 inches thick and are carbon steel "buttered" with stainless steel (See Drawing No. 5). The American Society of Mechanical Engineers (ASME), Section III is the governing code as required by the Code of Federal Regulations, 10 CFR 50.55a which controls the fabrication and erection of the reactor coolant pressure boundary. The ASME Code, invokes Section V of the code which describes the application of nondestructive examination techniques.

Article 2, paragraph T-272, states, "The maximum thickness for the use of radioactive isotopes is primarily dictated by exposure time; therefore, upper limits are not shown." This means that for the particular isotope employed, there is no maximum thickness of metal that can be radiographed but as the thickness increases the exposure times become unrealistically long. The choice between Iridium 192 or Cobalt 60 becomes one of time. If regular exposures of thick sections were required then Cobalt 60 offers a significant reduction in exposure time. However, for the general applications of radiography during construction of a nuclear power plant, Iridium 192 is more appropriate for the material thicknesses encountered.

The final radiography records for one steam generator nozzle were randomly selected to determine the technique and isotope used to perform the exposure. The technique sheet for nozzle to head weld "U" indicates that the isotope used was Iridium 192, which is only limited by the minimum thickness of 0.75 inches. A certified Level III, NRC Nondestructive Examiner reviewed the documentation and verified that the technique was acceptable under the ASME Code. Further, during the construction phase, the NRC reviewed the radiographs for seven of the eight steam generator nozzles (see inspection report 50-443/84-12, paragraph 14) and found then to be acceptable.

2.24.2 Conclusion:

The radioisotope used to perform the radiography of the steam generator nozzles is acceptable under the governing rules of the ASME Code, Sections III and V.



STEAM GENERATOR NOZZLE
PRIMARY

DRAWING No 5

2.25.0 Technical Response

2.25.1 Source of Issue

ELP document provided to the NRC during the April 20, 1987, meeting, Attachment D.

"...Due to a misinterpretation in measuring the lengths of pipe leading from the reactor to the steam generators and pumps, the column bases were set 3/4" further from the reactor than design had called for.

When the cross-over piping was installed, the pipe was found to be close to one of the pump columns. The space was less than one inch where six inches of insulation had to be installed....

Since this pipe is quite rigid, most of the stress would, I believe, fall on the welds at the pump and the reactor. This condition would also cause a slight twist in the cross-over piping..."

2.25.2 Scope of Issue

A concern was raised regarding the modification of all four Reactor Coolant Pump (RCP) support structures. The modifications involved relocation of the column lower end closest to the reactor vessel in each support structure (See Photos No. (10&11). They were moved inward toward the reactor vessel to eliminate interference with the cross-over leg piping (between the steam generator and the reactor coolant pump). The concern was related to the possible overstress of both cold leg and cross-over leg piping as a result of the introduced rotation (tilting) of the support structure.

2.25.3 NRC Inspection

The NRC staff reviewed revisions A, B & C of the Engineering Change Authorization (ECA) No. 1557 which addressed the above modification. Other documents reviewed by the NRC staff included:

- Major NSSS Equipment Setting As-built (DWG. No. 9763-F-81559)

- Field instruction for installation of reactor coolant pump vertical columns

- Preliminary guide for erection of RCS equipment support structures and supported components

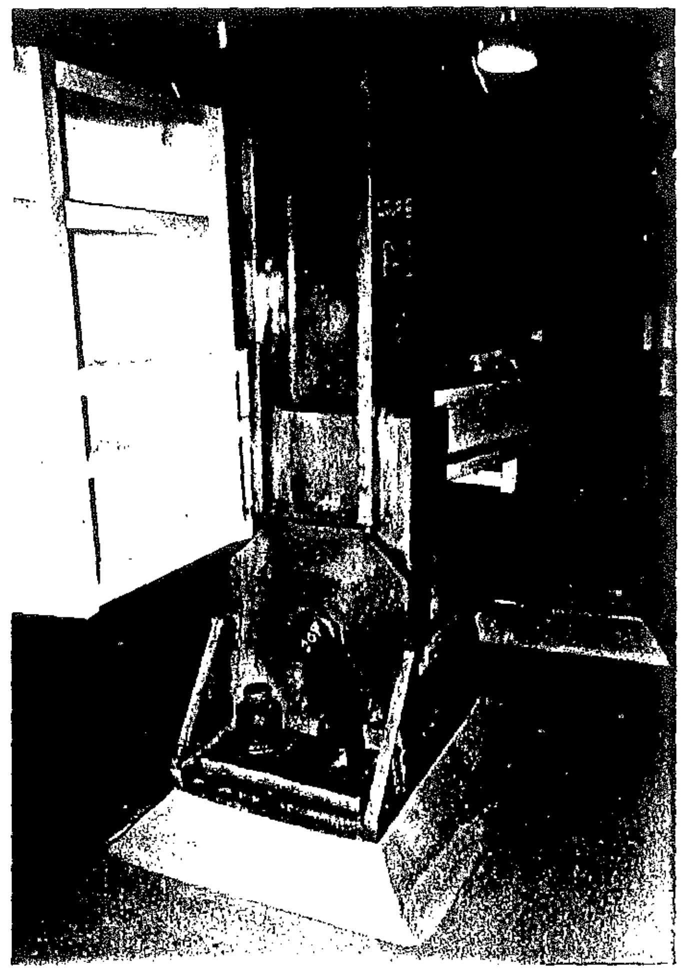


PHOTO NOS. 10 AND 11
REACTOR COOLANT PUMP SUPPORTS

- Suggested guide for setting major NSSS component supports
- QA/QC sign off forms 10A, 10B - I&II for RCP support modification
- Correspondence between UE&C and Westinghouse regarding RCP support modifications
- Work plan (Instruction/Checklist Package for Handling, Installation and Testing) for RCP's 1-RC-P-1C and 1D

The modification resulted in the movement of the lower end of each RCP loop support column by the amount of 2", 2", 1½" and 1 3/4" in loops A, B, C & D, respectively. The final offsets from the vertical position after adjustment of the support columns were 5 3/8", 5 3/8", 4 7/8" and 4 15/16", respectively. The relocation of support columns yielded an angular change with the vertical from approximately 2.05 degrees to approximately 2.33 degrees.

The staff also performed independent measurements of the column offsets in loops A & C using a Plumb Line. The measured offsets were consistent with the above numbers.

A review of the summary of the thermal stress range (ASME equation 12) at several nodes in the RC loop cold leg indicated that actual calculated stresses, in going from ambient to 100% power temperature, were well below the allowable code limit (3 Sm). Also, maximum calculated stresses for ASME equations 9, (upset and faulted) 12 & 13 were within allowable code limits. Nozzle loads at the RCP & RPV ends were also within allowable limits. Additional information provided by Westinghouse indicated that the modifications were acceptable from the standpoint of loop stresses and support loads due to the following:

- none of the cross-over legs was installed when the column modifications were performed
- the RCP is not a design anchor to the connecting piping systems
- maximum postulated pump flange rise at operating transient condition is 100 mils
- maximum permitted rise of RCP support column during modification was 40 mils
- pump flange levelness was monitored at 45° increments during the performance of the column's modifications
- originally installed pump flange levelness was maintained after the column's modifications.

The staff's review of W procedures and guidelines for the original installation did not reveal the requirement of a one mil tolerance during erection of the pump supports. The one mil requirement, referenced in the April 20 meeting, appears to have been required for surveying related measurements.

2.25.4 Conclusion

Based on the above, the staff concluded that the RCP's support structure modifications did not introduce a condition of potential overstress of existing cold leg piping beyond allowable design code limits.

2.26.0 Allegation

2.26.1 Source of Issue

ELP document provided to the NRC on April 20, 1987, during a transcribed interview, Attachment K, paragraph 3, states:

"3. In the Equipment Vault, at elevation - 61, he and his partner were told to put in a bulkhead at the bottom. There are three compartments, front, middle, and back; three cells on one side, three on the other. In the southwest cell, at the very back, there is a $50,000 Westinghouse pump the size of a car. It is an emergency cooling pump. He and his partner were told to build a bulkhead between the motor and the other two cells. This was to separate the motor from the other two cells so when water was forced through the system to clean it, the motor would not get wet.

They built a watertight bulkhead, but someone forgot to close a sumphole in the floor, which is a three-foot-by eight-foot trough. That hole is connected to all cells, so when the system was flooded, so was the motor. He worked in the equipment vault for six months after that incident, and although the millwrights were supposed to strip and clean the motor, they never did, at least not in the six months thereafter."

2.26.2 Scope of the Issue

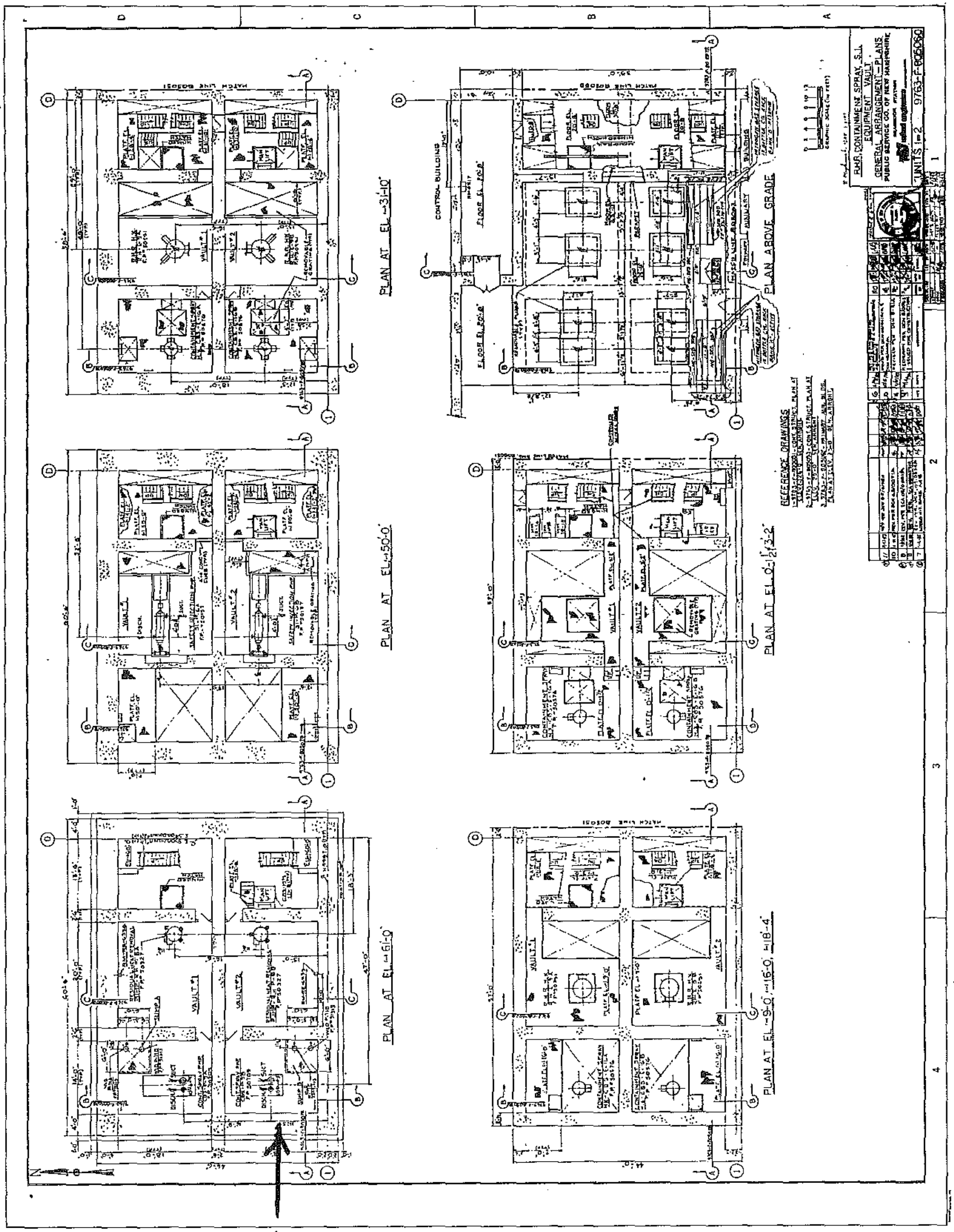
Determine if the flooding incident occurred, if it was properly documented and appropriate corrective actions taken.

2.26.3. NRC Inspection

The inspector toured the equipment vault and located the southwest corner of elevation - 61 feet (refer to Drawing No. 6, line number 1 of the Equipment Vault). The pump in the identified location is the containment spray pump, CBS-P-9B.

A Nonconformance Report (NCR), No. 2109, dated 6/10/83, was issued which describes a flooding incident affecting the CBS pump. The NCR states, "The subject pump was involved in a flood causing the CBS pump skid to be subjected to approx. 2 1/2' of water. Pump skid is located in Unit 1 RHR Vault, South side Elv. - 56."

A Westinghouse technical representative was contracted to inspect the motor on 7/8/83. An insulation resistance check was performed and a measured value of 800 Megohms to ground was recorded. The technician contacted the factory and was informed that the motor should be run to dry the insulation and lubricate the bearings. Subsequently, the pump manufacturer, Bingham-Willamette, Ltd., provided a technician in 1984 to disassemble the pump end and inspect the bearings and seals; damaged parts were replaced and the NCR closed out.



AREA OF INTEREST

DRAWING No. 6

The insulation resistance was tested again on 12/7/84, prior to the Initial Conditions Checklist completion on 12/21/84. This checklist is completed just prior to the initial pump performance test. The record indicates that the pump successfully passed the tests. This test is one of several that requires the pump and motor to be run as a component and later as part of the integrated system.

2.26.4 Conclusion:

The allegation that the pump and motor were flooded is true; however, the flooding incident was properly identified on a Nonconformance Report and appropriate corrective actions were taken. This allegation was not substantiated.

2.27.0 Allegation

2.27.1 Source of Issue

ELP document presented at the April 20, 1987, interview, Attachment K, paragraph 4, stated:

"In the waste process building, at the bottom, there are three cells. There is a two-and-half-foot diameter steel pipe which descends, does a 90-degree turn, then runs flat. In the elbows of the pipe there are GE or Westinghouse 200-horsepower motors to circulate cooling water. There are three such motors on sleds (two-and-a-half-foot by six-foot or eight-foot flat heavy steel beds). The sleds are supported at six points on steel springs. This is so when the pipe expands and pushes the motors' shafts down, the sleds collapse with the pipes' expansion. The sleds did not work properly, and the shafts on all three motors were bent. The millwrights were supposed to replace the sleds, but never did. These motors, which will not run correctly if the shafts are bent, are critical cooling components."

2.27.2 Scope of the Issue

Determine which pumps the alleger is describing and if they are safety related. Determine if the pumps have a design deficiency, that the design deficiency has been identified, and appropriate corrective actions are being taken.

2.27.3 NRC Inspection

A tour of the waste process building disclosed that the pumps being described by the alleger are located at elevation - 3 and are the recovery evaporator reboiler pumps, BRS-P-87A and BRS-P-87B, and the waste evaporator reboiler pump, WL-P-93 (See photos No. 12 & 13 and Drawing No. 7). These pumps have 30" diameter suction piping and 24" diameter discharge piping and the pump baseplates are mounted on spring loaded supports to compensate for expansion. These pumps service the boron recovery and the liquid waste system, respectively, and are not safety related.

The boron recovery system processes reactor coolant and primary drain water for possible reuse as primary grade water and boric acid, or for offsite disposal. The liquid waste system is non-nuclear safety class and designed to provide a central collection point for radioactive liquid waste.

The aforementioned pumps did recently experience a failure of the spring support system and a Request for Engineering Services, dated 12/23/86, and a Design Coordination Report, dated 1/3/87, were initiated. The failures were two adjacent spring mount isolators on pump BRS-P-87A base plate which broke at the relieved area above the threads. The licensee is currently processing a design change to strengthen this area on the supports. At the time of this inspection, the design change had not been completed.

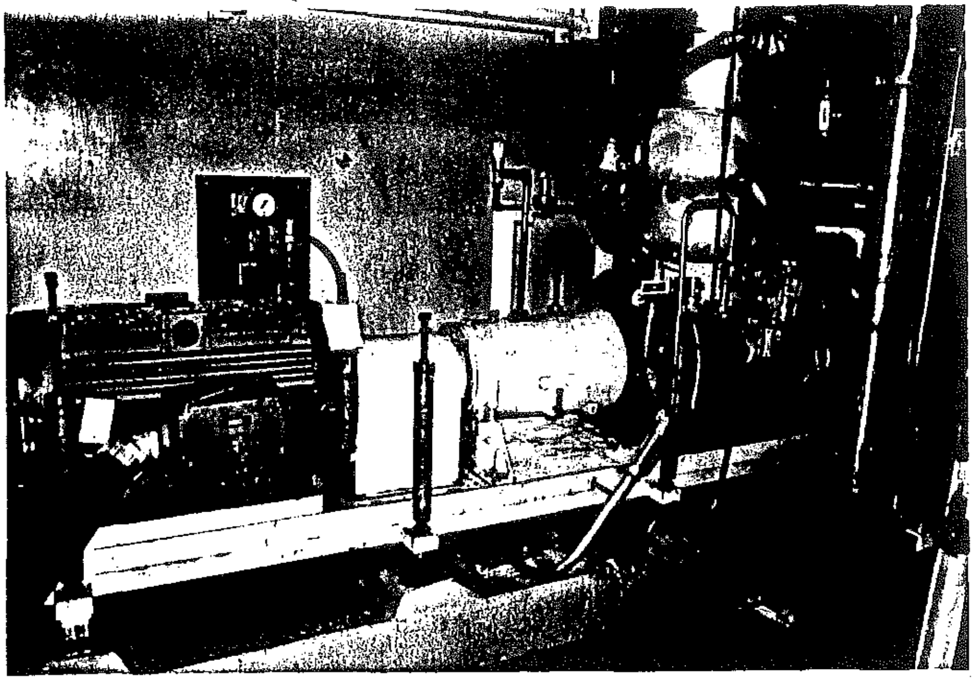


PHOTO NO. 12

TYPICAL BORON RECOVERY AND WASTE EVAPORATOR PUMP INSTALLATION

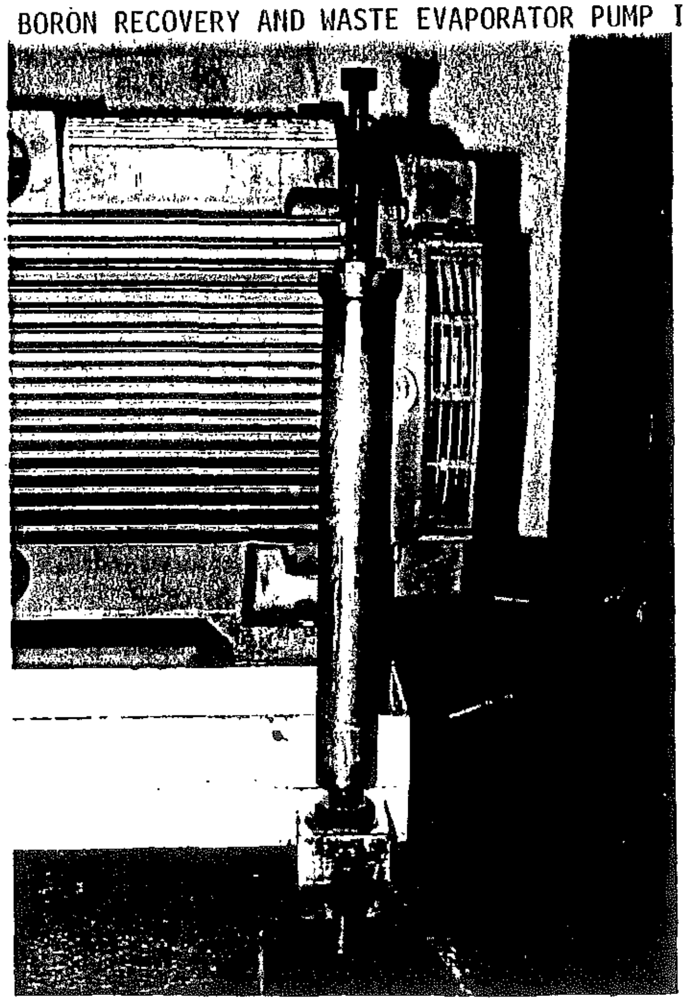
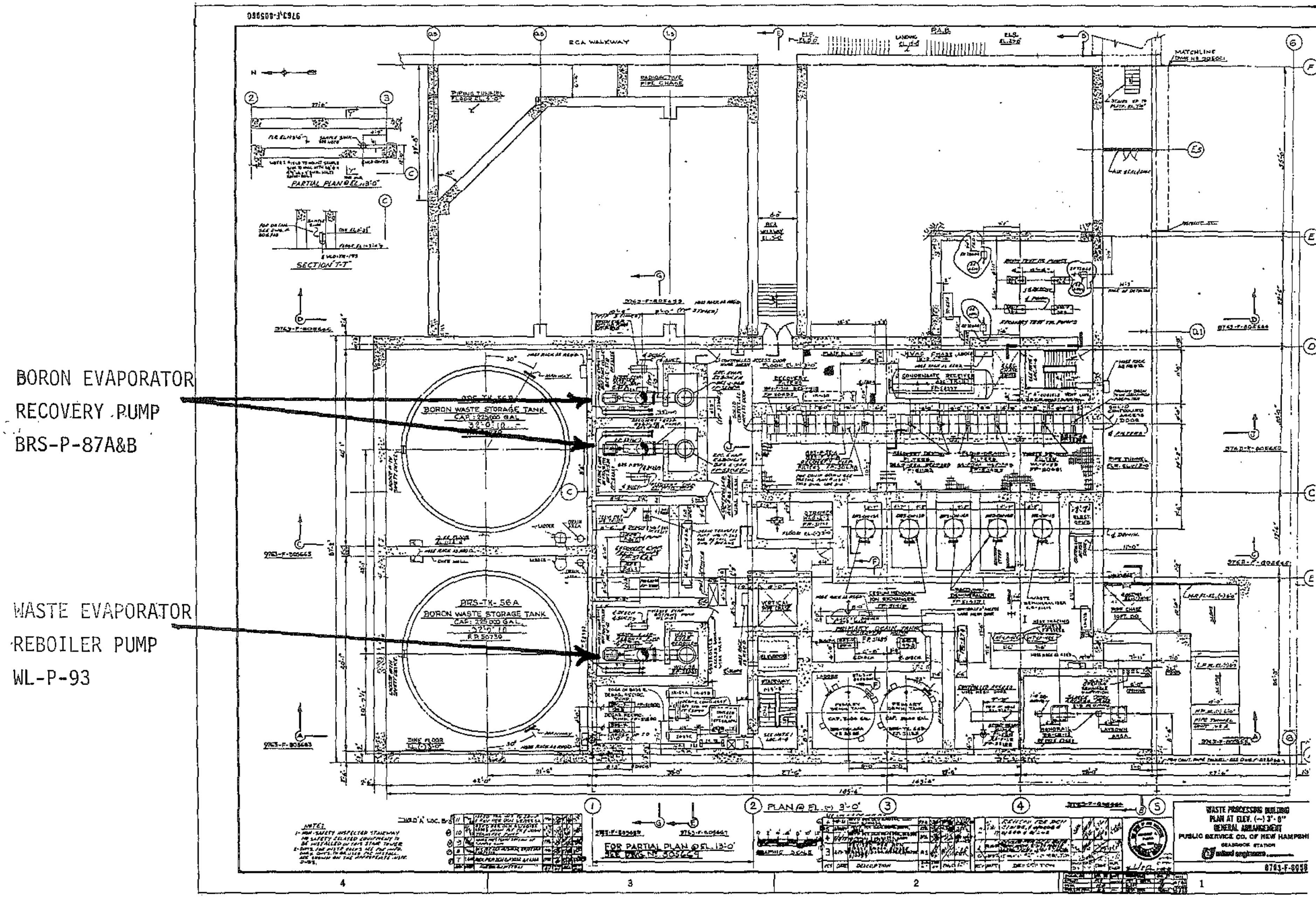


PHOTO NO. 13



BORON EVAPORATOR
RECOVERY PUMP
BRS-P-87A&B
WASTE EVAPORATOR
REBOILER PUMP
WL-P-93
BRS-TK-56B
BORON WASTE STORAGE TANK
BRS-TK-56A
BORON WASTE STORAGE TANK
E.C.A. WALKWAY
PIPING TUNNEL
RADIOACTIVE PIPE CHASE
PARTIAL PLAN @ EL. 3'-0"
SECTION T-T
PRIMARY DRAIN TANK
PLAN @ EL. (-) 3'-0"
FOR PARTIAL PLAN @ EL. 13'-0" SEE DWG. NO. 305669
WASTE PROCESSING BUILDING
PLAN AT ELEV. (-) 3'-0"
GENERAL ARRANGEMENT
PUBLIC SERVICE CO. OF NEW HAMPSHIRE
SEABROOK STATION

DRAWING No.7

2.27.4 Conclusion:

The pumps are not safety related and are not critical to the operation of the nuclear facility. The licensee has identified the deficiency in the design and has initiated corrective action to resolve the deficiency. The allegation that the millwrights were to replace the sleds (pump base plates) could not be substantiated. The sleds (base plates) will not be replaced with the current design change, but the spring supports will be strengthened to accommodate the loads.

2.28.0. Allegation

2.28.1. Source of Issue

Part 6 of an affidavit, dated November 6, 1986 and provided NRC as Attachment C to the ELP information supplied to the NRC on April 20, 1987 states;

"6. One inadequacy that seemed to be rampant was that although pipes were supposed to be capped off when left overnight, or as unfinished work in progress, they were very often not capped at all. All kinds of debris was left in the pipes. One one occasion, a very large pipe wrench was left in a pipe in the waste treatment building. The horizontal pipes which are open ended became a place for workers to rest things and place things, sometimes their own personal articles, sometimes debris. The debris would consist of wood, tools, clothing, and the pipes that were supposed to be capped off would just be open. Often the tradesmen would simply forget that something was left in the pipes when the joints were welded and some tradesmen would put matter in the pipes blatantly because of that "this job sucks" attitude which I can't emphasize enough was an overwhelming attitude throughout the plant."

2.28.2. Scope of the Issue

The alleger is concerned that debris entered the plant piping through pipe ends that were open. The NRC evaluated this concern and reviewed the plant piping flushing and startup testing activities to determine the extent to which debris may have remained in the plant piping.

2.28.3. NRC Inspection

The UE&C specification for assembly and erection of piping and mechanical equipment, number 9763-06-248-51 provides for control of pipe internal cleanliness in part 3.6. These controls include the provision that each piece of equipment including piping be suitably protected against contamination by foreign matter. All assemblies are required to receive a final cleaning after assembly, prior to hydrostatic testing. The inner surfaces of all systems were flushed and rinsed with clean, filtered water. Numerous NRC inspections conducted during piping installation have included observations of the presence of caps or covering on the ends of safety related field pipe runs and pipe sections. A small percentage of these observations have noted pipes without caps or covers. It is the NRC staff observation that during construction most of the pipe ends that should have been covered were covered. The prevention of debris from being introduced into piping is monitored by construction supervision, quality control inspectors, quality assurance personnel and others during construction. The problem was not rampant.

During inspection 87-07, the NRC inspector interviewed startup and test engineers to establish the findings of debris in piping during the pipe flushing and startup testing. Procedure GT-C-04 identifies 62 plant piping systems that required flushing and cleaning procedures to provide for achievement of the designated internal pipe cleanliness level. The pipe systems cleaned included both safety related and non-safety related pipe systems. The inspector reviewed a sample of detailed cleaning packages and the cleaning procedure log. The startup and test engineers described the typical flushing procedure, location of screens and filters in the test systems and described the types of debris found. No more than six small tools (one level, one chipping gun, one ball peen hammer and several tape measures) were flushed from the piping systems. Additional debris included weld purge dams, rags (degraded clothing), wood and solid particles ranging from grit to fine slit.

The pipe flushing was monitored by taking screen (40 mesh) samples to measure the particle sizes and to check for the absence of cloth and wood fibers. The overall pipe flushing and cleaning program was extensive, using approximately 23 people on a two shift basis for 1½ years.

2.28.4. Conclusion

The introduction of debris into plant piping systems was not as severe a problem as presented in the allegation. Maintenance of a clean condition inside piping was a joint effort by construction craftsmen, construction supervision, QC inspectors, site quality assurance, the authorized code inspector and NRC field inspection.

Interviews with startup and test engineers, review of procedures and documentation of pipe cleaning and flushing including valve actuations indicates that plant piping is now internally clean to procedural requirements.

2.29.0. Allegation

2.29.1. Source of Issue

An Attachment to the ELP information provided to the NRC on April 20, 1987 stated.

"The air condition system maintains the temperature in the equipment vault and containment. There are four-by-six-foot ducts which start at -61 elevation in the equipment vault and go to the roof of the Primary Auxiliary Building. They provide cooling for all the buildings around the containment building. It took six to seven months to install the ducts, and everything was sealed with silicon. Just as the very last bolt was sealed, an engineer told the sheetmetal workers the wrong gauge of steel was used for the ducts. It was never changed."

2.29.2. Scope of the Issue

Determine if the heating ventilation and air conditioning ducts (HVAC) in the equipment vault and into the Primary Auxiliary Building (PAB) are installed in accordance with engineering requirements.

2.29.3. NRC Inspection

The NRC inspector examined the installed HVAC ductwork in the North and South Equipment Vaults and the extensions of this ductwork into the PAB. These field observations were compared to the HVAC procedural requirements, applicable drawings and engineering calculations. The HVAC ductwork installed in the PAB is designated to be safety class 2 (10 gauge) material and was confirmed to be as required.

The vertical runs of HVAC ductwork in the North and South Equipment Vaults were installed from the minus 61' elevation to the plus 16' elevation in 1981 as seismic supported category 1. This ductwork was installed as 22 gauge (0.0299" thickness).

By UE&C memorandum dated March 5, 1982 the equipment vault HVAC ductwork was designated to be reviewed by engineering for upgrade to Safety Class III. The engineering calculations set 9763--B-14-10, dated 8/12/82, provide the methodology and applicable calculations for this upgrade. This engineering review established that the equipment vault HVAC ductwork of 22 gauge would meet the Safety Class III requirements of specification 9763-006-226-2 and 3 providing some ductwork stiffening was added. The NRC inspector confirmed the stiffner parameters of B-14-10 were implemented in the field by examination of the ductwork as installed in the equipment vault. Reference drawings UE&C 9763-F-604112 and HAH-SM-604112-1, dated 4/14/83.

2.29.4. Conclusion

Subsequent to the installation of the equipment vault HVAC ductwork in 1981, the safety class designation of this ductwork was upgraded. This required engineering review and field addition of stiffners to the ductwork but not a change to the duct gauge thickness. The allegation states the ductwork was installed with the wrong gauge material; this was not substantiated. The ducting was upgraded and required engineering evaluation and added stiffness.

2.30.0. Allegation

2.30.1. Source of Issue

During the April 20, 1987 interview with ELP one participant said that he had heard that the gate valves in the fire protection system throughout the turbine building were replaced and that there was sedimentation on both the incoming and outgoing lines. The alleger also stated that the sleeves and ball cocks of these valves were replaced because bacteria (MIC) had eaten the sleeves. On April 23, 1987, ELP confirmed that the replacement work of concern had occurred on 8" and 10" gate valves during the March to June 1986 time period.

2.30.2 NRC Inspection

The NRC inspector established that numerous fire protection valves had been disassembled during the time noted by the alleger under work request number 86-004602 in the turbine building. The detailed parts listing for the gate valves included are shown on the Stockham Valve and Fitting Data Sheet page 142 and the NIBCO data sheet, page 103. The valves in question do not have parts itemized as sleeves or ballcocks.

The work request 4602 and attached documentation indicates the actual work scope on the 8" and 10" valves to be bonnet removal, seat cleaning, gland repacking and reassembly. Valve seat cleaning work required inspection of the valve seats with a mirror which included observation of the pipe interior. The responsible fire protection engineer is on record as observing each valve seat and adjacent pipe, finding the pipe to be clean and free of obstructions. The valve seat cleaning was performed to prevent minor water leakage past the valves.

2.30.3. Conclusion

Maintenance work was performed on 8" and 10" gate valves in the turbine building fire protection piping. The alleged problem of sediment in piping or MIC attack of sleeves and ballcocks was not substantiated.

2.31.0 Allegation

2.31.1 Source of Issue

"The Waste Process Building (the tank farm) stores chemicals like boron, etc., to control a reaction. November, 1985, during the hot functional tests, the pressure in containment was brought up to 160 pounds per square inch, and everything expanded. The tank farm building (about 100 feet by 150 feet) has walls of poured concrete and steel. An improper thickness of structural steel was used in the main skeleton. The building wracked about eight inches (as if someone put their hands on opposite corners and twisted). They welded more plates of steel to the girders to reinforce it. During the test, you could see the pipes moving, and there was a lot of banging and popping." (Sic.)

2.31.2 Scope of the Issue

It is claimed that the Waste Processing Building design was inadequate. Due to undersize structural members, the WPB deformed approximately 8 inches when the containment was pressurized to 160 psi during the hot functional test.

2.31.3 NRC Inspection

The NRC staff examined the design of the Waste Processing Building (WPB), and the preoperational test sequence of the containment structure; performed a walkthrough inspection of the WPB/tank farm area, and closely examined the concrete wall on the eastside and the structural steel framing on the other sides of the enclosure. This close visual examination of walls was carried out to detect any evidence of structural distress and/or deformation. Because an eight inch deformation of any structure in any direction or mode must leave some visible evidence of deformation in a rigid structure.

The inspector also interviewed and held discussions with engineering and supervisory personnel responsible for preoperational tests, maintenance, and operation of the plant.

Based on the above review and examination, the inspector determined the following:

The tank farm area in the WPB is essentially a box like structure composed of a reinforced concrete wall on the east side, and braced structural steel frame on the other three sides. Exposed portions of the braced frame are covered by metal siding, and the roof is a concrete slab. The calculations used for the design and development of the mathematical model for analysis are documented in United Engineers Calculation No. SBSAG-SWB. There are approximately 15 feet of fill concrete under the refueling water storage tank and the spray additive tank. Also, the structure is separated from containment and other adjacent structures by a seismic gap of approximately 3".

During the early part of 1984, the NRC performed an extensive in-depth inspection of plant design called the Independent Design Inspection (IDI). The WPB was one of the selected structures for in-depth design review. One of the findings of this inspection was that the seismic analysis model for this structure did not take into account the "as-built" arrangements of the structure. Pursuant to this finding the licensee initiated a reanalysis of the structure, and found that the structure, in fact, needed modification to resist the changed loads and stresses. The NRC staff established by review of United Engineers' letter No. SBU-91680, dated November 27, 1984, that the proposal for the structural modification of the building was underway at that time. The structural modification of the building was started in 1985, and was finished by the middle of 1986. These modifications were going on during the hot-functional test of the plant in November 1985.

Regarding the statement that during the hot functional test the pressure in containment was brought up to 160 psi, the NRC staff believes that this perception of the alleger is based on a complete lack of information or understanding of hot functional testing.

The hot functional test in a PWR steam supply system is performed to verify the operability of Nuclear Steam Supply System (NSSS) portion of the plant without a nuclear heat source. The pressure and temperature in the NSSS is brought up to the operating pressure and temperature of the system, which is approximately 2200 psi and 550 °F, respectively. Also, a PWR containment, during plant operations, functions at atmospheric pressure. The containment system at Seabrook is designed to function at a maximum pressure of 52 psig and is tested for integrity at 60 psig (115% of design pressure). The NRC staff is unaware of any regulatory or technical requirement in which the containment has been pressurized to 160 psi (3 times the service capacity). In any event, pressurization of containment does not affect the WPB structure due to the structural isolation gap (seismic gap) between these structures. During the hot functional test, however, the thermal growth of piping and supports do occur, and there are some "popping" or other sounds (accoustic emissions) from affected piping and supports due to stress readjustments and physical movements in piping and equipment. But, this is one of the purposes of the test to evaluate the behavior of the systems, and make the necessary adjustments in pipe hangers for the hot condition. It, therefore, is conceivable that an uninformed person may consider this phenomenon unusual and be concerned about it.

Regarding the 8 inch of "wracking" (deformation), the staff considers the statement that a rigid structure of reinforced concrete and structural steel construction could deform 8" without any applied load incredible. Also, even if the structure suffered such a load, a deformation of 8" inches (either planner or torsional) would not leave the structure intact without any visible evidence of rupture/yielding is also not credible.

2.31.4 Conclusion

This allegation is not valid, and is apparently based on a lack of understanding of the purpose of structural modifications in the PAB, the intent of the hot functional test, and an imaginary deformation of the building. No safety concern exists.

3.0 Unresolved Items

Unresolved items are matters about which more information is required to determine if the issue is acceptable, a deviation or a violation. Unresolved items are discussed in paragraphs 2.4.3.2 and 2.17.3.

APPENDIX A
PERSONS CONTACTED

NEW HAMPSHIRE YANKEE

B. Brown, Engineer
S. P. Buchwald, Quality Assurance Supervisor
R. M. Cooney, Technical Project Manager
T. Feigenbaum, Vice-President Engineering and Quality Programs
W. Gagon, EAR Program
W. J. Hall, Regulatory Services Manager
D. G. McLain, Projection Services Manager
D. Moody, Station Manager
D. W. Perkins, Licensing
S. B. Sadosky, EAR Program
W. J. Temple, Licensing Coordinator

The above listed personnel were present at the Exit Interview on May 8, 1987. Other craftsmen, engineers and quality control/quality assurance personnel were contacted as the inspection interfaced with their work.