CP&L

Carolina Power & Light Company

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R. B. STARKEY, JR.
Vice President
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SERIAL: NLS-92-136

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
MASONRY BLOCK WALLS

REFERENCES:

1. Letter from Mr. Steven A. Varga (USNRC) to Mr. R. A. Watson (CP&L) dated April 9, 1992, "Masonry Block Walls At Brunswick Steam Electric Plant, Units 1 and 2."

2. Letter from Mr. R. B. Starkey, Jr. (CP&L) to Document Control Desk (USNRC) dated April 15, 1992 (Serial: NLS-92-118), "Masonry Block Walls."

3. Letter from Mr. Steven A. Varga (USNRC) to Mr. R. A. Watson (CP&L) dated April 27, 1992, "Masonry Block Walls At Brunswick Steam Electric Plant, Units 1 and 2."

4. Letter from Mr. R. A. Watson (CP&L) to Document Control Desk (USNRC) dated May 29, 1992 (Serial: NLS-92-148), "Corrective Action Plans."

Gentlemen:

On April 9, 1992 (Reference 1), the Nuclear Regulatory Commission requested information concerning masonry block walls at the Brunswick Steam Electric Plant, Units 1 and 2. The Company's responses were provided in a letter dated April 15, 1992 (Reference 2). In a letter dated April 27, 1992 (Reference 3), the NRC requested further information concerning the anchor bolt deficiencies and the Company's proposed corrective actions. Subsequently, on May 12, 1992, a meeting was held between Carolina Power & Light Company and the Nuclear Regulatory Commission to discuss these structural and seismic issues. By letter dated May 29, 1992 (Reference 4), Carolina Power & Light Company documented commitments made during the May 12, 1992 meeting with regard to major work items to be completed prior to start-up of the Brunswick Plant, Units 1 and 2.

The purpose of this letter is to: (1) identify the commitments that were made during the May 12, 1992 meeting for implementation for each unit prior to start-up from their next scheduled refueling outages, and (2) respond to the NRC Staff letter dated April 27, 1992 concerning masonry block walls at the Brunswick Steam Electric Plant, Units 1 and 2. The April 27, 1992 NRC letter contained a list of questions and issues to be addressed as part of a meeting between CP&L and the NRC. The NRC questions, along with Carolina Power & Light Company's responses which were discussed with the NRC on May 12, 1992, are provided in Enclosure 1 of this letter. The information provided in the enclosed responses supplements the information from our May 12,

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1992 meeting. Where possible, references to the May 12, 1992 meeting presentation slides have been included in the responses.

In the April 15, 1992 response, CP&L committed to develop by May 1, 1992 a sampling plan and schedule for physical examination of raceway support anchor installation and building steel support anchor installation. The sampling plan and schedule information is provided in Enclosure 2 of this letter.

Information addressing questions concerning frozen studs identified as part of CP&L's IE Bulletin 79-02 review is provided in Enclosure 3.

A summary of the commitments contained in the responses to the April 27, 1992 questions (Enclosure 1) is provided in Enclosure 4. If the action was also provided in our May 29, 1992 letter, this has been noted.

Please refer any questions regarding this submittal to Mr. D. C. McCarthy at (919) 546-6901.

Yours very truly,

R. B. Starkey, Jr.

WRM/wrm (wallltr.wpf)

Enclosures

cc: Mr. S. D. Ebneter
Mr. R. H. Lo
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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKET NOS. 50-325 & 50-324
OPERATING LICENSE NOS. DPR-71 & DPR-62
MASONRY BLOCK WALLS

NRC QUESTION I.A:

Discuss the causes of the apparent lack of timeliness of corrective actions for Emergency Diesel Generator (EDG) masonry wall bolting and service water pumps.

CP&L RESPONSE:

Diesel Generator Walls:

In late 1986 or early 1987, a Brunswick Plant Technical Support organization engineer investigated a report from the maintenance instrumentation and control organization that diesel generator building anchor bolts in angle frames were not penetrating into the column. The Technical Support organization addressed the concern in a site memorandum (BPE-5306) to the Brunswick Engineering Support Unit (BESU) dated February 13, 1987. Due to reasons that could not be determined, a response which proposed sampling of the bolts was not made by the Brunswick site engineering unit until January 26, 1988. An approval for bolt sampling was sent to the Brunswick Engineering Support Unit via site memorandum BPE-6083 dated April 8, 1988. This site memorandum apparently was either lost or not received by the Brunswick Engineering Support Unit. Although the Brunswick Engineering Support Unit engineer involved in the response to memorandum BPE-5306 was transferred to another organization, his follow-up eventually led to the discovery that the memorandum was lost. The Technical Support organization transmitted another site memorandum (BPE-6884) dated September 15, 1989 to the Brunswick Engineering Support Unit requesting a response to the prior sampling approval. Also during the time period the memorandum was lost, re-assignments within the Technical Support organization changed the engineer responsible for handling the wall deficiency.

An engineer from the Nuclear Engineering Department investigated the condition at the Brunswick Plant in November 1989. During the next year, the bolts in the diesel generator building wall angles were as-built verified using a feeler gauge. The data was used to analyze the diesel generator building walls. On November 15, 1990, a short term structural integrity calculation (calculation number 0-15-34A-270) was completed. Based on the calculation, the Company concluded that the walls met short term structural integrity criteria for the design basis loading. A permanent long term modification design was initially issued to the Brunswick Plant site construction organization on April 8, 1991. This design was reviewed by plant organizations and was revised due to field constructability concerns. The design was re-issued on February 19, 1992.

As discussed in our enforcement conference on May 12, 1992, the primary causes were:

1. Inadequate Technical Support and Engineering interface
2. Scope and impact of the wall deficiency was not recognized by personnel
3. Perception of resource limitations dominated scope of effort
4. No standards existed on acceptable time limits for short term conditions

5. Lack of a comprehensive line-driven corrective action program

Reference: See pages 17-18 and page 27 of the May 12, 1992 CP&L Technical Presentation package.

Service Water Pumps:

During 1979, service water pump deterioration was noted and an upgrade of the service water pumps was initiated. In 1981, during the engineering process, an error in the original seismic calculations performed for the original Peerless pumps was discovered. This calculation erroneously concluded that the pump was classified as a rigid structure due to an error in the pumps' natural frequency determination. This condition was reported to the Nuclear Regulatory Commission and evaluated in letters dated April 10, 1981 (Serial: BSEP 81-0791) and July 16, 1981 (Serial: BSEP 81-01359). Brunswick Plant Modifications 81-207 and 81-208 (issued in 1982) installed new Johnson Pump components and upgraded the seismic qualification of the pumps to meet the long term seismic requirements of the pump specification.

In 1989, an additional seismic issue developed. Inappropriate response spectra curves were found in an equipment specification for butterfly dampers. A non-conformance report (NCR 89-8) was issued to address the problem's generic implications. In June 1989, during engineering for a service water pump upgrade to install a product lubrication enhancement, an incorrect response spectra was found in the service water pump specification. At that time, Engineering Evaluation Report 89-237 evaluated the upgraded pumps as capable of meeting their design functions. The modifications for the product lubrication upgrade were promptly changed to address the seismic response spectra error and were re-released in April 1990.

Concurrent with this effort, the original motor thrust bearings were evaluated as part of the system hydraulic review and found to require an excessive pump minimum flow, so as not to overload the pump motor bearing. Because this minimum flow requirement detracted from available system flow margin, a study was done to evaluate the best approach to resolve this issue. This study was completed in June 1990 and concluded the upgrade of service water pump motors was the best alternative. New pump motor bearings were designed and, through this process, it was determined that it would be possible to entirely eliminate the service water lubrication system. Plant modifications were revised to incorporate the deletion of the lubrication water system and modification of the motors. These modifications were approved in 1991 and a purchase order for pumps issued in September 1991.

Resolution of the seismic response spectra issue was integrated with other emerging service water system design issues relating to system hydraulics and single failure. As a result, the current service water pump design addresses seismic upgrades, elimination of the lubrication water system, minimum flow requirements for the pumps, and changes to minimize pump maintenance.

Carolina Power & Light Company's current schedule is to begin replacement of the service water pumps starting in 1993. The service water pump upgrades are scheduled to be completed November 30, 1994. The Company has expedited vendor support for equipment deliveries needed to achieve the schedule. The Company has determined that during the interim period, the service water pumps are capable of withstanding a design basis earthquake. Carolina Power & Light Company is conducting additional field inspections to provide assurance that these calculations are valid. This inspection will be completed prior to start-up of either Brunswick Plant unit. Based on the information known and the actions taken by CP&L to address service water system issues, the Company believes that the actions to address the service water system seismic qualification issue have been timely.

Reference: See pages 38-43 of the May 12, 1992 CP&L Technical Presentation package.

NRC QUESTION I.B:

Present the results of masonry wall bolt inspections, and provide the basis for the 25 percent sampling program for masonry wall bolts for walls other than those in the EDG Building.

CP&L RESPONSE:

The results of the wall inspections are as follows:

1. Plated block shield walls (10 walls) (diesel generator building, elevation 23 foot)
   - Approximately 60 percent of self drilling expansion anchor bolts missing
   - Approximately 7 percent of through bolts missing
2. Block walls
   - Diesel generator building, elevation 23 foot - one block wall missing 50 percent of anchor bolts
   - Diesel generator building, elevation 50 foot - 2 walls, smaller diameter sleeve anchor bolts (5/8-inch in lieu of 3/4-inch) than designed
3. Reinforced concrete walls (diesel generator building, elevation 23 foot)
   - Approximately 85 percent of anchor bolts missing

The deficiencies were the result of original construction work. The walls in the diesel generator building having structural angle restraints with expansion anchors were 100 percent inspected by a combination of ultrasonic examination and/or anchor/nut removal and re-installation. In addition, the walls with IE Bulletin 80-11 modifications in both the control building and diesel generator building were reviewed by confirming quality assurance records with field reviews to ensure installations matched the drawings. No deficiencies were found in any of the modification work performed in response to IE Bulletin 80-11.

In addition, a group of seismic walls outside the diesel generator building (i.e., the control building and reactor building) were inspected. Of the seismic walls, 11 walls were restrained by anchor bolted angles (6 walls by original design and construction). These six walls were 100 percent inspected. In a few cases, anchors were discovered to be 5/8-inch sleeve anchors in lieu of 3/4-inch anchors, but all met IE Bulletin 80-11 requirements and in no cases were fake bolts encountered. The remaining five walls were post IE Bulletin 80-11 modified and found to be in conformance with design. No deficiencies were found.

Six walls in the control building (elevation 49 foot) were determined to be required to be in place post earthquake for control room habitability requirements, but were deemed to be non-safety by the 1980 IE Bulletin 80-11 reviews. These walls were declared inoperable. As committed in our letter dated May 29, 1992 (Reference 4), Carolina Power & Light Company will complete repairs upgrading to seismic classification walls in the control building (elevation 49 foot) that have been determined to be required post-earthquake for control room habitability requirements. Also as indicated in our letter dated May 29, 1992 (Reference 4), Carolina Power & Light Company will perform a design review and a field inspection review of non-safety masonry walls to verify the walls are appropriately classified as non-safety.

Reference: See pages 13, 22, and 24-25 of the May 12, 1992 CP&L Technical Presentation package.

<u>NRC QUESTION I.C</u>:

Explain why you are inspecting less than 100 percent of through-wall bolts. Are non-functioning bolts to be removed?

<u>CP&L RESPONSE</u>:

One hundred percent of the through bolts in the diesel generator building block walls have been ultrasonically examined with a total of approximately seven (7) percent determined to be improperly installed. As indicated in our letter dated May 29, 1992 (Reference 4), accessible non-functional through-bolts will be removed and cover plates installed over the holes prior to start-up of the two Brunswick Plant units.

Reference: See page 21 of the May 12, 1992 CP&L Technical Presentation package.

<u>NRC QUESTION I.D</u>:

Describe your program for inspection and analysis of reinforced concrete walls.

<u>CP&L RESPONSE</u>:

Five reinforced concrete non-load bearing wall panels were poured in the diesel generator building. The Company has reviewed the balance of the plant and determined that the diesel generator building was the only location in the plant where this type of wall was utilized. These five walls are currently undergoing repair to restore them to their design configuration. As indicated in our May 29, 1992 letter (Reference 4), these repairs will be completed prior to start-up of the two Brunswick Plant units.

Reference: See page 23 of the May 12, 1992 CP&L Technical Presentation package.

NRC QUESTION I.E:

With regard to pipe supports, you stated that the sampling technique and frequency of expansion anchor bolt inspections were in accordance with the requirements of IE Bulletin 79-02. You also stated that "out of a total 433 anchors that were examined, 156 anchors could not be fully evaluated because the stud (rod, bolt) or leveling nut was, for unknown reason, 'frozen' or seized." If the bolts were frozen, that means those bolts could not have been backed out for measurements of bolt thread engagement length, anchor sleeves embedment lengths, and the anchor torquing could not have been verified. Explain how the sampling technique and inspection frequency used could have met the requirements of IE Bulletin 79-02, as stated above.

CP&L RESPONSE:

During the IE Bulletin 79-02 effort completed in the early 1980's, the sampling program chosen involved testing of one anchor per base plate as stated in our response to the NRC dated July 12, 1979 (Serial: GD-79-1739). The data showed, however, essentially all anchors were tested where possible. When the April 1992 CP&L audit team reviewed this data, it was clear the original IE Bulletin 79-02 sampling exceeded the one anchor per base plate requirement of the Bulletin. This fact formed the basis for the Company's conclusion that the sampling technique met IE Bulletin 79-02 requirements.

With respect to the number of frozen anchors, Carolina Power & Light Company's letter to the Nuclear Regulatory Commission dated July 26, 1982 (Serial: BSEP/82-1616) provides the summary data cited in our April 15, 1992 response (Reference 2). A copy of the July 27, 1982 letter is enclosed (Attachment 1). On page 5 of the letter, the total number of anchors tested is 433, and the number of anchors not tested due to frozen studs (117) or frozen leveling nuts (39) is a total of 156. The frozen anchors were not considered test failures or passes. For purposes of IE Bulletin 79-02, these anchors were considered as anchors not completely tested. As noted on page 4 of the July 27, 1982 letter, those anchors with frozen studs were load tested, but could not be checked for thread engagement or embedment. Most of the 156 frozen anchors had grouted base plates. The grout was removed, providing access to the underside of the base plate. Any fraudulent installation practices, such as tack welds or cut off anchors, would generally have been visible at the time the base plate grout was removed.

Thus, the large number of anchors tested and the detailed data sheets documenting the generally good test results lead Carolina Power & Light Company to the conclusion that the Brown & Root installed expansion anchors on safety-related pipe supports did not involve deficient installation practices.

The CP&L audit team leader discussed these issues with a member of the NRC Staff on April 30, 1992 at the Brunswick Plant site. The Company's understanding was that all NRC questions concerning this issue were addressed satisfactorily.

NRC QUESTION I.F:

With respect to Design Guide II.20, "Design Guide For Civil/Structural Operability Reviews" (DG), for piping and piping supports, the staff finds that the DG does not address or inadequately addresses the following attributes in the operability determination criteria:

(1) How the comprehensive loading combinations for both normal and faulted conditions are considered in the criteria.

(2) What damping values and response spectra are to be used, as well as a comprehensive methodology and analysis procedure similar to Ft. Calhoun's and Dresden/Quad Cities', which have been accepted by the staff.

(3) How other occasional loads, including water hammer or steam hammer, as well as secondary loads, are to be used.

(4) The appropriateness of using the "Structural Review Panel," in lieu of a comprehensive evaluation provided in the DG.

Explain how these issues are addressed in your operability evaluations for piping and piping supports.

CP&L RESPONSE:

Part 1:

Additional documents applicable to the Brunswick Plant provide the specific load case combination information. Study Report M-020, "Criteria for Evaluating and Performing Computerized Piping Analyses of Existing Systems with Minor Modification," Appendix A, page 3 of 5, provides load combinations and stress limits for piping. Study Report M-021, "Evaluation Criteria for Existing Pipe Supports Associated with NRC Bulletins IE79-02, 79-07, and 79-14," provides load combinations for pipe supports. Copies of the two study reports referenced are enclosed (Attachments 2 and 3). Additional structural load combinations are provided in the applicable sections of the Updated Final Safety Analysis Report.

Part 2:

The response spectra critical damping ratio used for Brunswick Plant short-term evaluations is PVRC N-411 damping. The use of N-411 damping for the Brunswick Plant was approved in an August 28, 1985 NRC letter to Carolina Power & Light Company. In some cases, a higher damped curve will be invoked to provide a "first cut" at the operability assessment. Final assessment would be based on time history or gap evaluation.

Part 3:

The specific Brunswick Plant short term structural integrity (STSI) load case combinations specified in Study Reports M-20 (Piping) and M-21 (Supports) require that any applicable system transients be considered in an operability evaluation. Secondary (i.e., self-limiting) loads are not considered in operability pipe stress evaluation; however, secondary loads are considered in the pipe support load combinations.

Part 4:

The use of a "Structural Review Panel" concept was based on the following:

- The use of experienced structural engineers to make field assessments of nonconforming conditions.
- Use of previously performed, similar structural evaluations.
- Encouragement of "hands on" engineering in lieu of drawing only review.
- Use of calculations as backup to reinforce judgements where appropriate.
- Issue plant instructions to repair conditions.

Experienced engineers use this process to provide hand calculations, study analysis, or document judgement to accept existing items that require modification for full compliance. This allows the engineer to quickly move to the redesign phase and focus efforts on providing a rigorous long-term calculation and applicable modification drawings. These criteria are not used for long-term documentation of structure acceptability. The long-term documentation calculations currently performed provide a complete, comprehensive, and detailed evaluation.

It should be noted that the Brunswick Plant piping effort now underway is a design basis reconstitution effort to address as-built errors discovered in 1986. Between 1979 and 1986, a complete IE Bulletin 79-14 program was performed, resulting in the analysis of identified safety lines and the modification of over 2000 supports (both units). The reconstitution effort now has re-inspected approximately 3500 supports since 1986. Since original calculations and modifications were previously performed, the "Structural Review Panel" operability review would consist of a documented review of the as-built differences identified by the re-inspection against the existing IE Bulletin 79-14 analysis to ensure operability. The piping would then be re-analyzed in the updated, as-built condition and any "long-term" fixes identified and issued. This method of review of as-built impact on existing analyses has been used for operability determinations at the Brunswick Plant.

In addition, DG II.20 is being revised to incorporate Brunswick-specific piping and support criteria, as well as update requirements for short term operability in accordance with Generic Letter 91-18.

Reference: See pages 46-48 and page 50 of the May 12, 1992 CP&L Technical Presentation package.

NRC QUESTION I.G:

Your root cause response provided a discussion of the paper work that you had reviewed but reached no conclusion. Discuss the progress you have made with respect to determining the root cause.

CP' _PESPONSE:

See the response to Question I.A above.

NRC QUESTION I.H:

Discuss the progress you have made on your pla _ to inspect and correct identified deficiencies.

CP&L RESPONSE:

All masonry walls that were considered seismic in the original IE Bulletin 80-11 walkdown with structural angle restraints attached with expansion anchors that are located outside of the diesel generator building have been reviewed. The results of these inspections are summarized in the response to question I.B.

NRC QUESTION II.A:

Address how you determined there was an issue, why it was overlooked in your earlier response to our letter, and what actions you took to evaluate and correct the deficiencies.

CP&L RESPONSE:

Cosmetically applied anchor bolt heads in structural angle restraints attached to reinforced concrete walls in the diesel generator building were found to be an issue when engineers supporting repairs of the masonry wall missile barriers questioned the installation of similar structural angle restraints on local reinforced concrete walls, even though their design basis in a reinforced concrete wall was not immediately apparent. These walls were not addressed in CP&L's April 15, 1992 response because they were found afterward. Please note that the responses provided in the Company's April 15, 1992 letter (Reference 2) were preliminary and based on the best available information.

NRC QUESTION II.B:

In light of your recent identification that the anchor bolt deficiencies were more widespread in the DG building than originally anticipated, discuss your plans for validating the original conclusions resulting from your IE Bulletin 80-11 program reviews.

CP&L RESPONSE:

An overall review of the IE Bulletin 80-11 program is underway for the Brunswick Plant. The review will address existing masonry wall functions, including missile barrier, tornado barrier, ventilation barrier, or other functions for which it is not analyzed.

NRC QUESTION III.A:

Characterize the type, number and safety significance of the backlog of items qualified under your short term structural integrity program.

CP&L RESPONSE:

The current short term structural integrity list is tracked by a total of 48 outstanding items. The items can be characterized as follows:

ITEMS	CONTENTS	IMPACT	COMPLETION SCHEDULE
41	217 Pipe Support Design Turnovers Identified	Minor Repairs Short Term Qualified	Reload 8 (Unit 1)* Reload 10 (Unit 2)*
1	Service Water Pumps	Short Term Qualified	11/30/94
1	Diesel Generator Building Walls	Fixed (Long Term Qualified)	Prior to Start-up
1	Air Tubing Supports	Short Term Qualified	Reload 8 (Unit 1) Reload 10 (Unit 2)
1	RWCU Supports	Short Term Qualified	Reload 8 (Unit 1) Reload 10 (Unit 2)
1	Fuel Oil Small Bore	Short Term Qualified	1995
1	Main Steam Radiation Monitor	Short Term Qualified	Reload 8 (Unit 1) Reload 10 (Unit 2)
1	Diesel Generator Exhaust	Short Term Qualified	Prior to Start-up

* Exceptions: - Diesel Generator Service Water Supply and Return - Unit 2 Reload 11
- Service Water Lubrication Water Supports - Unit 1 Reload 10 and Unit 2 Reload 11

These items are tied to existing modification schedules.
The Brunswick Unit 1 Reload 8 outage is currently anticipated to begin March 1993. The Brunswick Unit 2 Reload 10 outage is currently scheduled to begin September 1993.

Of the total of 48 items listed, engineering for long-term fixes is complete on thirty-four. In general, the repairs are typically the addition of gussets, beam stiffeners, adjusting U-bolts, additional welding, component replacement, and the removal of snubbers. Because CP&L is reviewing, reconstituting, or redoing calculations done by our architect-engineer in the 1980 to 1986 time frame for piping and supports, the type of repairs are not as significant as would be found in a first-time IE Bulletin 79-14 review. The Company has adopted more conservative criteria in accordance with up-to-date analysis methods as the work has progressed. Efforts are underway to ensure a complete short term structural integrity program is outlined and documented, noting both items under review as well as those with fixes issued. The Company has initiated a third-party review of the short term structural integrity program which may result in the identification and scheduling of additional r[illegible]. The short term structural integrity list above will be reviewed for completeness and any additional items will be added, if necessary.

The above items were characterized for safety significance by evaluating them in a limited scope seismic probabilistic risk assessment of the Brunswick Plant and then by verifying the results and assumptions with plant walkdowns. From the list of short term structural integrity items, only the service water pumps and the diesel generator building walls were found to be significant contributors to core damage frequency. The pipe supports were evaluated as not significant and the remaining items were screened out based on lack of applicability to safe shutdown as determined in the probabilistic risk assessment model.

Seismic interactions from the service water pumps and the diesel generator building walls were incorporated into a limited scope seismic probabilistic risk assessment to examine the safety significance of the as-built condition versus the as-designed condition of these two short term structural integrity items. Seismic fragilities for the pumps and the walls were developed for the as-built and as-designed conditions. The core damage frequency in the as-built condition increased by a factor of approximately 2.3 over the as-designed condition. However, in absolute terms, the core damage frequencies were small (i.e., on the order of 1E-5 per year for both cases).

The pipe supports were found not to be a significant safety concern. Plant isometric drawings showing the pipe supports in question were reviewed and evaluated using the probabilistic risk assessment model of the plant to determine if they were concentrated near safety significant components such that multiple safe shutdown paths would become unavailable during a seismic event. Effects such as "unzipping" of a series of supports and seismic-induced flooding were also considered. The plant walkdowns confirmed these findings and also verified that other potential pipe or valve interactions not identified in the drawing review would not significantly affect the estimated core damage frequency. Overall, the supports were found to be of high quality material and good construction.

It is concluded that these as-found conditions resulted in an increased estimate of core damage frequency from seismic events. However, the value of this increased estimate is well within the range of core damage frequencies from other seismic probabilistic risk assessments and is not considered to be a significant risk.

NRC QUESTION III.B:

Discuss the schedule for correcting these items and the reason more timely corrective action was not taken.

CP&L RESPONSE:

The schedule for correcting the backlog of short term structural integrity program items is provided in the response to Question III.A above. The primary reason more timely corrective action was not taken with regard to these items was the lack of standards establishing acceptable time limits for short term conditions to exist.

Reference: See page 27 of the May 12, 1992 CP&L Technical Presentation package.

NRC QUESTION III.C:

Provide the basis for assumed validity of existing analyses for short term structural integrity in view of deficiencies found recently in analysis for CBEAF (Control Building Emergency Air Filters) supports and masonry wall bolting.

CP&L RESPONSE:

As discussed in the preceding questions, the overall short term structural integrity list is dominated by pipe support repairs resulting from the piping design turnover efforts. Most repairs are somewhat minor in nature. However, to ensure the proper evaluation of the items addressed, Carolina Power & Light Company has initiated a third-party review of the short term structural integrity program. The review will address evaluation techniques, field validation of critical assumptions, as well as a review of other communications from the Technical Support organization to the Engineering organization. This review began the week of May 11, 1992 and is expected to be complete by July 31, 1992.

Reference: See page 52 of the May 12, 1992 CP&L Technical Presentation package.

NRC QUESTION III.D:

Provide the basis for design values assumed in masonry wall analyses (i.e. bolt, mortar, rebar and grout strength).

CP&L RESPONSE:

Bolts:

Manufacturers' allowables for Phillips Redheads were used in the original construction. These allowables use a safety factor of five (5). Subsequent research by the Seismic Qualification Utility Group (SQUG) for USI A-46 confirms these allowables. The allowables for the Phillips bolts used in the short term evaluations were below the SQUG or vendor allowables.

Mortar, Block, Grout Strength:

Allowable stresses for both flexible members and shear walls have been established based on tests for shear walls.

Two major test programs have evaluated the shear strength of concrete block masonry walls. The first test was performed by Schneider; his test results were used as the basis for developing the UBC, NCMA, and ACI code allowable stresses for reinforced masonry. A subsequent test program was performed at the University of California - Berkeley. These test results were used as a comparison with the code allowables. Therefore, code allowables are generally accepted and specifically used at the Brunswick plant.

NCMA Code Method 2 was used to establish concrete masonry strength. This method requires testing of concrete masonry units. The allowable compressive stress (f'm) is determined from the test results for various mortar types. Specific test results for concrete masonry units at the Brunswick Plant are attached (Attachment 5).

No tests were performed on the mortar; however, Specification 9527-01-29-1 required that mortar adhere to the following ASTM Standards: ASTM C91, ASTM C144, ASTM C270, ASTM C476, and ASTM C780.

Rebar:

Vertical reinforcement bars are deformed bars, ASTM 615-68 Grade 60 for sizes Number 6 to Number 11 and Grade 40 for smaller sizes. Horizontal joint reinforcing is standard Dur-O-Wal galvanized collar joint reinforcing (vertical) is welded wire fabric. Presence of rebar in both masonry and concrete walls were determined by magnetic rebar scanner.

Reference: See page 33 of the May 12, 1992 CP&L Technical Presentation package.

NRC QUESTION III.E:

Describe the quality controls applied to verify bolt torque values during recent masonry wall work.

CP&L RESPONSE:

Permanent construction work at the Brunswick Plant is performed under approved Quality Control (QC) procedures. W 'l repair work underway is being conducted with the appropriate QC verification of work and materials.

The verification of anchor existence during the week of April 6, 1992 was performed to obtain information to determine the operability status of the plated missile shield walls. The 3/4-inch Phillips Red Head self drilling anchors were backed out, checked for length, and re-installed to a "snug tight" condition. This work was directly observed, supervised, and documented by Nuclear Engineering Department site engineering staff personnel.

NRC QUESTION III.F:

Explain the non-uniformity in the use of steel angles on masonry walls in the switchgear rooms in the EDG building, and in the use of steel bracings for the stairwell enclosures in the same rooms.

CP&L RESPONSE:

A floor plan for the diesel generator building showing the interior wall designations for elevation 23 foot is provided in Attachment 4.

In the switchgear rooms, three types of block walls exist:

- Plated block walls which are part of the standard missile shield wall typically between the diesel generators.
- Unplated block walls between the switchgear.
- Unplated block walls which form an enclosure around the stairwell and serve as a fire barrier between floors.

The typical detail for plated block walls required steel angles on the top, bottom, and sides of the wall due to seismic and missile loading. These angles were installed except where penetrations or other obstructions exist.

The typical detail for unplated walls between switchgear requires steel angles only on the top of the wall. The sides are restrained by mortar joints with dove-tail anchors (as required), and the bottom is restrained by a fully bedded first course. The reinforced concrete itself serves as the missile barrier.

The typical detail (original construction) for the stairwells uses no steel angles.

Steel bracing for the stairwell enclosures was installed as an IE Bulletin 80-11 fix for Wall 9a. This wall was classified as safety-related due to proximity to safety-related equipment. The mirror image of this wall on the north end of the building is Wall 9a. Wall 9a is classified as nonsafety-related because the potential target (safety-related equipment) does not exist in this end of the building. Wall 9a was downgraded to non-safety in later IE Bulletin 80-11 submittals and, therefore, steel bracing was not installed. However, recent reviews have determined that wall 9a is required to support wall 9c, which is safety-related. Therefore, additional analysis is in progress to determine if steel bracing is required for wall 9a.

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKET NOS. 50-325 & 50-324
OPERATING LICENSE NOS. DPR-71 & DPR-62
MASONRY BLOCK WALLS
PLAN AND SCHEDULE FOR PHYSICAL EXAMINATIONS

Scope:

Carolina Power & Light Company plans to select, by means of walkdowns and design document research, a substantial raceway run, a substantial building steel sample, a substantial HVAC duct run , and a sample of various equipment foundations with accessible Red Head self drilling snap-off anchor bolts that are original plant construction in each of the following buildings for each Brunswick Plant unit: the control building, the reactor building, the diesel generator building, and the service water building. The drywell is excluded from the initial sample in order to maintain personnel exposure as low as reasonably achievable. The drywell may be included in an expanded sample if results of the above sampling demonstrate the need.

Inspection:

For the anchors selected, the inspections will primarily be conducted by ultrasonic testing with the option of removing anchors for inspection in order to identify deficiencies such as missing, cosmetically applied, or faked anchors. During the walkdowns and inspections, personnel will also observe for any bolt heads that appear to be welded to their plate that may or may not be part of the sample. The need for sample expansion will be determined by CP&L management based on the results of the inspections.

Schedule:

The walkdowns and design reviews are currently in progress to identify the structures to be inspected. Completion of the inspections is expected to be completed by July 31, 1992.

Documentation:

Inspection results will be documented, including the deficiencies identified. Documentation will be adequate to permit the inspection results to be verified later. Deficiencies identified will also be documented in accordance with applicable plant procedures and Nuclear Engineering Department guidelines. The need for sample expansion will be determined by CP&L management based on the results of the inspections.

ENCLOSURE 3

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKET NOS. 50-325 & 50-324
OPERATING LICENSE NOS. DPR-71 & DPR-62
MASONRY BLOCK WALLS

IE BULLETIN 79-02 (SP 79-22) REVIEW

The purpose of this information is to address questions concerning the fourteen (14) frozen studs in surface/flush mounted attachments reported in Carolina Power & Light Company's letter dated April 15, 1992 (Serial: NLS-92-118). This information was discussed in CP&L's response to NRC Question 2, Response Item A.3, "Frozen Studs," on page E1-4.

Ten (10) of these were threaded studs and four (4) were hex head bolts. Although frozen, the threaded studs were still properly and successfully tested for design load capacity, meeting the requirements of Special Test Procedure (SP) 79-22 ("Proof Load Test"). The hex head bolts could not be properly tested since it was impossible to provide a gap behind the baseplate for insertion of 1/4-inch shims as required for the "Proof Load Test." Test results were inconclusive for these four hex head bolts. The Company's review of the installations using these hex head bolts (drawing numbers L-02260-1372 and L-02260-5230, Sht. 1) resulted in the following conclusions:

1. L-03360-1372, Mark No. PS-1372 (Iso. D-02846, Sht. 178A-D.P.3)
   This support is installed on line number 2-RNA-222-1-170 which is Quality Class D (non-safety related). The support anchors on this support will be reworked prior to start-up of either Brunswick unit.

2. L-02260-5230, Sht. 1, Mark No. PS-5230 (Iso. D-02846, Sht. 166A, D.P.560)
   Based on the Special Procedure 79-22 documentation, only one bolt of this two bolt installation was frozen. The other bolt passed inspection, but could not be properly tested due to the frozen bolt. This support was re-inspected under the IE Bulletin 79-14 closeout program. Both bolts were reworked under trouble ticket 88-ARUU1 (the bolts tightened and torques to 110 foot-pounds). As a result of line re-analysis, this support is now also non-Q. Therefore, the Company believes that no further action is needed.

ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKET NOS. 50-325 & 50-324
OPERATING LICENSE NOS. DPR-71 & DPR-62
MASONRY BLOCK WALLS

SUMMARY OF COMMITMENTS

As discussed in the information provided in Enclosure 1 of this letter, Carolina Power & Light Company commits to complete the following activities:

1. Carolina Power & Light Company's current schedule is to begin replacement of the service water pumps starting in 1993. The service water pump upgrades will be completed by November 30, 1994. The service water lubrication water system will be removed as part of this replacement.

2. Carolina Power & Light Company will, prior to start-up of either Brunswick Plant unit, complete additional field inspections to provide assurance that calculations supporting interim seismic operability of the service water pumps are valid.

3. Carolina Power & Light Company will complete repairs upgrading seismic classification walls in the control building (elevation 49 foot) that have been determined to be required post-earthquake for control room habitability requirements. (This action was included in CP&L's May 29, 1992 letter, Reference 4).

4. Carolina Power & Light Company will perform a design review and a field inspection review of non-safety masonry walls to verify the walls are appropriately classified as non-safety. (This action was included in CP&L's May 29, 1992 letter, Reference 4).

5. Carolina Power & Light Company will remove accessible non-functional through-bolts and install cover plates over the holes. (This action was included in CP&L's May 29, 1992 letter, Reference 4).

6. Carolina Power & Light Company will complete repair of five reinforced non-load bearing wall panels in the emergency diesel generator building to restore the walls to their design configuration. These repairs will be completed prior to start-up of the two Brunswick Plant units. (This action was included in CP&L's May 29, 1992 letter, Reference 4).

7. Carolina Power & Light Company is revising Design Guide II.20 to incorporate Brunswick-specific piping and support criteria and to update requirements for short term operability in accordance with NRC Generic Letter 91-18.

8. Carolina Power & Light Company will perform a review of IE Bulletin 80-11 program for the Brunswick Plant. The review will address existing masonry wall functions including missile barrier, tornado barrier, ventilation barrier, or other functions for which it is not analyzed.

9. Carolina Power & Light Company will complete long term qualification of the 217 identified pipe support items prior to start-up following the Unit 1 Reload 8 outage and the Unit 2 Reload 10 outage.

10. Carolina Power & Light Company will complete long term qualification of the instrument air system tubing supports prior to start-up following the Unit 1 Reload 8 outage and the Unit 2 Reload 10 outage.

11. Carolina Power & Light Company will complete long term qualification of reactor water cleanup system supports prior to start-up following the Unit 1 Reload 8 outage and the Unit 2 Reload 10 outage.

12. Carolina Power & Light Company will complete long term qualification of diesel fuel oil small bore lines prior to start-up following the Unit 1 Reload 8 outage and the Unit 2 Reload 10 outage.

13. Carolina Power & Light Company will complete long term qualification of the main steam line radiation monitor supports prior to start-up following the Unit 1 Reload 8 outage and the Unit 2 Reload 10 outage.

14. Carolina Power & Light Company will complete long term qualification of the emergency diesel generator exhaust supports prior to start-up from the current outage.

15. Carolina Power & Light Company will complete long term qualification of the diesel generator service water supply and return line supports prior to start-up following the Unit 2 Reload 11 outage.

16. Carolina Power & Light Company will, following the installation of new service water pumps in 1994 which do not require an external lubrication water system, decommission the remaining portions of the existing service water lubrication water system and its supports. This work will be completed by the end of the Unit 1 Reload 10 and Unit 2 Reload 11 outages.

17. Carolina Power & Light Company will ensure a complete short term structural integrity program is outlined, including items under review and items with fixes issued.

18. The short term structural integrity list will be reviewed for completeness and any additional items will be added, if necessary.

19. Carolina Power & Light Company will perform a third-party review of the short term structural integrity program to address evaluation techniques, field validation of critical assumptions, and a review of communications from the Technical Support organization to the Engineering organization. The third-party review is expected to be completed by July 31, 1992.

ATTACHMENT 1

CP&L LETTER DATED JULY 26, 1982 (SERIAL: BSEP/82-1616)
SUPPLEMENTAL RESPONSE TO IE BULLETINS 79-02, 79-07, AND 79-14

Carolina Power & Light Company

COP

Extra copy

Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429

July 26, 1982

FILE: B09-13510C
SERIAL: BSEP/82-1616

Mr. James P. O'Reilly, Director
U. S. Nuclear Regulatory Commission
Region II, Suite 3100
101 Marietta Street N.W.
Atlanta, GA 30303

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
SUPPLEMENTAL RESPONSE TO IE BULLETINS 79-02,
79-07, AND 79-14

Dear Mr. O'Reilly:

In our letter (BSEP/81-0440) dated February 25, 1981, we committed to complete the Phase I and Phase II portion of the seismic reanalysis of the plant to satisfy the requirements of IE Bulletins 79-02, 79-07 and 79-14 by July 31, 1981, and March 31, 1982, respectively. This letter is to report the Phase I, or generic analysis, and Phase II, the individual analysis, have been completed in accordance with these dates.

The Phase I and Phase II programs did not include as-built evaluation of inaccessible isometrics as that work required a unit outage for access to complete. These isometrics on Unit No. 2 have been as-built and reanalyzed during the current outage. Two inaccessible isometrics in Unit No. 1 remain. These will be as-built and evaluated during this year's outage.

In the February 25, 1981, letter, we provided a list of potential problem areas and inconsistencies that were discovered during our review program, together with intended resolution and schedules. We will address the status of these areas in the same order as listed previously.

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A. Lines Originally Seismically Analyzed, But Not Included in IE Bulletin 79-07 Efforts

Upon the completion of the seismic line review, 38 isometrics remained to be analyzed. Twenty-one of these isometrics were addressed under the February 25, 1981 letter. The remaining 17 (isometrics) were analyzed under the Phase I program. There were no short-term fixes required based on this analysis. All long-term fixes associated with these isometrics have been issued and are in the process of being installed. We plan to complete the fixes associated with these isometrics on both units by the end of the next Unit No. 1 refueling outage. This outage is presently scheduled to start in September 1982. For Unit No. 2, approximately 35 inaccessible (during power operation) fixes may not be completed during the current outage, however, due to insufficient outage time. If necessary, these few remaining supports will be completed during the next available outage of sufficient duration, and no later than the end of the next Unit No. 2 refueling outage.

B. Vents, Drains, Instrument Connections

These connections were not covered by the original computer analysis so they did not fall under the scope of IE Bulletins 79-07 and 79-14. It was determined they should be evaluated to give reasonable assurance that they did not significantly affect the process piping. A generic analysis of these connections showed a negligible effect for large bore piping. The remaining small bore piping was handled by a sampling program. Approximately half of these connections were analyzed with no cases of overstress or the process pipe. It was thus concluded that no significant impact on the analysis of the parent lines existed.

C. Unanalyzed Loads Due to Valve Eccentricity

In our letter of February 25, 1982, approximately 25 motor-operated valves were cited as not having been analyzed for eccentric loadings. All but four have been evaluated based on UE&C estimated valve and operator weights and centers of gravity. Efforts to verify the assumed values with vendors have indicated that the estimated values are as accurate (± 10 percent) as any values which could be supplied by the manufacturers. Since the analyses will not be significantly affected by a 10 percent variance in weights and the vendor's estimates will not improve the accuracy of the analysis, the vendor verification program was terminated. The remaining four valves were not originally computer analyzed and, therefore, are not encompassed by IE Bulletin 79-07. However, a generic analysis was performed on these lines which verified that the piping stresses are within ANSI B31.1 limits.

D. Verification of Acceptable Containment Penetration Nozzle Loads

All penetration nozzle loads have been verified as acceptable per the requirements of IE Bulletins 79-07 and 79-14.

E. Vendor Supplied and Vendor - A/E Interface Piping

After a review of vendor documentation, we have concluded that these lines were not computer analyzed. IE Bulletins 79-07 and 79-14 thus do not apply.

F. Small Nozzle Loads on Safety-Related Components

The only lines encompassed in this category are the vent and drain lines off the HPCI, RCIC and core spray pumps, which were analyzed.

The IE Bulletin 79-14 condition has been reviewed and no short-term fixes were required. Long-term fixes are scheduled on the same basis discussed for Item A.

G. Seismic Requirements Inconsistencies

Only two lines under this category were found to require analysis; one is the surge line in the Diesel Fresh Water Cooling System, the other is a drain line in the Standby Liquid Control System. These lines are small and were not originally computer analyzed. Therefore, the 79-07 Bulletin is not applicable. However, in order to completely close out all outstanding items, these lines were as-built and evaluated as part of Item A.

H. CRD System Baseplate Flexure Analysis

In regard to our bulletin requirements for the CRD System supports, we stated in our February 25, 1981 letter, "Completion of baseplate flexure analysis on CRD piping not essential to safe shutdown is scheduled for completion as part of the Phase II Program." CP&L has determined that nonessential portions of the CRD System are not safety related or seismically qualified; therefore, this analysis was not required.

I. Anchor Bolt Testing

As stated in our February 25, 1981 letter, the scheduled anchor bolt testing per IEB-79-02 of all the additional supports identified for testing is now complete for Unit No. 2.

The testing of self-drilling anchors included application of a torque representing a pull out load equal to or greater than the allowable design load for the anchor. Concrete embedment and thread engagement were also measured whenever it was possible to remove the bolt/stud from the anchor. It must be noted here that this phase of test program covered many floor mounted supports employing self-drilling anchors with all thread rod studs and grout. Because of moisture conditions during plant operation several studs were found to be frozen in anchors and could not be removed for measurement of depth. All of these anchors, however, either passed the preload test or were replaced.

All supports that did not meet the test acceptance criteria were conservatively evaluated for the load values generated by IE 79-07 reanalysis effort. Repairs were made to deficient supports and the frozen studs broken during test. Rusted self-drilling anchors in service water intake structure were replaced by stainless steel wedge anchors.

A total of 163 baseplates containing 433 anchors were included under this phase of the program. All baseplates and anchor bolts were tested to the extent as was reasonably possible. The primary test verifying adequate preload was performed on 88 percent of all anchor bolts and on at least one anchor bolt on all of the baseplates except two. One of these baseplates had a seismic load of one pound and the other a safety factor of 20. These loads are sufficiently low that the satisfactory inspections of their condition when testing was attempted was adequate to assure their reliability. The preload test demonstrated the actual ability of each anchor bolt to withstand its design load. The failure rate for this test was 2.4 percent. The inability to back off the leveling nut was the predominant reason for not testing all of the anchors. A stuck leveling nut does not indicate any structural deficiency with an anchor, it just prevented any meaningful testing. The low failure rate and the extensiveness of the test program for both baseplates and anchors provides a high confidence in the ability of the existing anchor bolts to accommodate the required loads.

Tests for proper installation were performed on 59 percent of the anchor bolts. A failure rate of 1.6 percent was obtained for improper engagement and 1.6 percent for inadequate embedment. Problems with anchor bolts or studs which could not be removed (27 percent of all anchors), in addition to the previously mentioned frozen leveling nut problems (9 percent of all anchors), were the overriding reasons preventing full testing. All of the anchors with unremovable bolts or studs were successfully tested for preload, however, demonstrating the load capability of the anchors. This satisfactory demonstration and the low failure rates indicate there is no concern for inadequate embedment and engagement. In addition, 31 percent of these anchor bolts which were not fully tested were subsequently replaced for other reasons further reducing the number of not fully verified anchors.

serial BSEP/82-1616

An overall failure rate of 5.6 percent was obtained from the testing program on Unit No. 2. The extensiveness of the preload testing, the low failure rates from the tests, and the number of anchors which were replaced, leave a small opportunity for inadequate baseplates.

Anchor Bolt Testing Results Summary on Unit No. 2 ← (81-83 Program

Total number of baseplates	163
Number of baseplates tested	161
Total number of anchors	433
Number of anchors tested for preload	380
Number of anchors failed preload	9
Preload test failure rate	2.4%
Number of anchors not tested for preload	53
Number not tested due to frozen leveling nut	39
Number tested for other reasons	14
Number of anchors tested for embedment	254
Number of anchors with inadequate embedment	4
Embedment test failure rate	1.6%
Number of anchors not tested for embedment	179
Number not tested due to frozen leveling nut	39
Number not tested due to frozen stud	117
Number not tested for other reasons	23
Number of anchors tested for engagement	256
Number of anchors with inadequate engagement	4
Engagement test failure rate	1.6%
Number of anchors not tested for engagement	177
Number not tested due to frozen leveling nut	39
Number not tested due to frozen stud	117
Number not tested for other reasons	21
Total failure rate	5.6%

As required by IE Bulletin 79-02, CP&L has completed the test program for Unit No. 2. Unit No. 1 testing is essentially complete. The results are being tabulated and checked to assure no identified supports remain untested. During the upcoming Unit No. 1 outage scheduled to start in September 1982, any supports not yet tested in the primary containment will be tested and results transmitted to your office.

total # of tests 380 + 254 + 256 = 890

total failures 9 + 4 4 = 17

In our February 25, 1981 letter, we committed to performing a weld verification sampling program for seismic pipe supports as part of the Phase I program. We have completed this sampling program with greater than a 95 percent confidence level that the original QC inspection program was adequate. This 95 percent confidence level is consistent with that required for the IEB 79-02 sampling programs and thus we believe our pipe support welds are acceptable.

In conclusion, upon the completion of the long-term fixes discussed previously, the Pipe Stress Analysis Summary Tables will be updated to indicate completion of the field modifications. This update will signify our completion of work and compliance with the above bulletins. We anticipate this milestone will occur in mid-1983, at which time you will be notified in writing.

Very truly yours,

ORIGINAL SIGNED BY
C. R. DIETZ

C. R. Dietz, General Manager
Brunswick Steam Electric Plant

JSB/dg

cc: Mr. R. C. DeYoung

bcc: Mr. D. L. Bensinger/File: BC/A-4
Mr. F. R. Coburn
Mr. A. B. Cutter
Dr. T. S. Elleman
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