UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report No.: 50-400/87-41

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket No.: 50-400 License No.: NPF-63

Facility Name: Harris 1

Inspection Conducted: November 16 - December 11, 1987

Inspectors: J. J. Lenahan — Date Signed 2/22/88

R. P. Carrion — Date Signed 2/22/88

Approved by: J. Blake, Chief — Date Signed 2/22/88
Materials and Processes Section
Division of Reactor Safety

SUMMARY

Scope: This special announced inspection was in the areas of followup on previous inspection findings and on concerns pertaining to design activities.

Results: One violation was identified - Uncontrolled Change to Design Input, paragraph 6.b.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

P. L. Brady, Civil Engineer
V. Cox, Project Specialist, Electrical
*R. Knott, Structural Engineer
A. Lewis, Civil Engineer
**L. I. Loflin, Manager, Harris Plant Engineering Section
***J. W. McKay, Resident Civil Engineer
G. A. Meyer, Manager, Modifications Projects
J. A. Nevill, Manager, Nuclear Engineering Department
*R. Panelle, Senior Structural Engineer
M. Sealey, Civil Engineer
*V. Stephenson, Senior Structural Engineer
**D. Tibbitts, Manager, Licensing
J. Turner, Civil Engineer
*E. Wagner, Manager, Plant Engineering
**M. G. Wallace, Regulatory Compliance Specialist
R. A. Watson, Vice President, Harris Plant
***H. L. Williams, Principal Engineer

Other licensee employees contacted included four civil engineers, and one electrical specialist.

NRC Resident Inspectors

S. P. Burris
G. Maxwell

*Attended November 20 exit interview
**Attended December 11 exit interview
***Attended both exit interviews

2. Exit Interview

The inspection scope and findings were summarized on November 20 and December 11, 1987, with those persons indicated in paragraph 1. The inspector described the areas inspected, and discussed in detail the inspection findings listed below. Dissenting comments were not received from the licensee. The following new items were identified during this inspection:

- Inspector Followup item 400/87-41-01, Add ANSI N690-1984 to List of Applicable Codes, Standards and Specifications in FSAR Section 3.8.3.2.

- Violation 400/87-41-02, Uncontrolled Change to Design Input

The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Enforcement Matters

a. (Closed) Violation item 400/86-69-01, Failure to Protect Permanent Plant Equipment. This violation involved craftsmen using safety related cable trays as scaffolding and attaching scaffolding to permanent safety-related equipment. The licensee's corrective actions for this violation are stated in their December 11, 1986 response to NRC Region II for Inspection Report No. 50-400/86-69.

The cause of this violation was that the two craft personnel who were using cable tray L1901 SR4 as scaffolding failed to follow Work Procedure WP-48. To correct this problem, personnel and construction materials were removed from the cable tray and the tray and installed cables were inspected for damage. No damage was noted. The two craft personnel and their foreman were reprimanded and the CP&L construction manager issued a memorandum dated September 8, 1986 to the project craft supervisor reiterating the requirements of WP-48.

Regarding the scaffolding which was attached (wired) to an instrument tubing track and was in contact with an HVAC valve actuator, the licensee's corrective actions included relocation of the scaffolding and inspection of the track and actuator for damage. No damage was noted. The foreman responsible for erection of the scaffolding was reprimanded.

The inspectors examined CP&L procedure number CMP-4, Rigging and Temporary Loads, and procedure number MMM-022, Rigging Loads from Permanent Plant Components. These procedures were written to replace procedure WP-48 which was a construction procedure and control the attachment of rigging and temporary loads to permanent plant equipment. Violation 400/86-69-01 is closed.

b. (Closed) Unresolved Item 400/86-69-03, Evaluation of Containment Building Load Reversal Design Error as Nonconformance and for Reportability to NRC. During the special inspection conducted September 4-11, 1986, documented in Inspection Report Number 50-400/86-69, licensee design engineers informed the inspector that errors were discovered in the original EBASCO design for the containment building (CB) platform steel. Unresolved Item 400/86-69-03 was identified by the inspectors as a result of the failure of the licensee to write an NCR to document and disposition this problem. Subsequent to inspection 86-69, the inspector held further discussions with licensee engineers pertaining to the final design verification program for the CB platform steel. These discussions disclosed that the final design verification program was

a systematic in-depth review of the EBASCO design and was being conducted to include all final attachment loads (e.g. piping, equipment, conduit cable trays, etc.) which were not in the original analysis. The purpose of this review was to verify that the "as-built" CB platform steel structures conformed to FSAR and NRC design requirements, and to modify the CB steel structures when necessary to restore the original FSAR design margin. The discussions also disclosed that the licensee only suspected the problem concerning the load reversals during inspection 86-69. The extent of the problem was not identified until late September 1986, after conferring with the EBASCO design engineers, and performing some additional design verification studies. At that time, NCR 86-600 was written to disposition the problem. The CB steel design verification program and disposition of NCR 86-600 was tied to an oral commitment to re-inspect welds on the connections for the 92 heaviest loaded beams on the CB platform. This commitment was made by a licensee representative during the September 4-11, 1986 inspection. These beams were defined as those loaded to 0.85 or greater of the code allowable stresses (Note: In Inspection Report Number 50-400/86-69, this definition was incorrectly written that the heavily loaded beams were those loaded at 0.75 or greater of the code allowable stresses. The figure of 0.75 is in error. The correct figure should have been 0.85.) During the licensee's in-depth review of the CB platform steel design, licensee engineers identified several questions regarding design methodology. These questions were documented and resolved in CP&L memoranda and various design calculation books. The inspector reviewed portions of the CB structural steel design calculations. The calculations reviewed are discussed in paragraphs 5.b and 5.c below.

Based on the discussions with licensee engineers, review of various design calculations, and other documentation, and review of the sequence of events, the inspectors concluded that NCR 86-600 was not reportable under 10 CFR 50.55(e), and that the licensee acted properly in delaying initiation of an NCR until late September when NCR 86-600 was issued. Unresolved Item 400/86-69-03 is closed.

c. (Open) Violation Item 400/86-77-01, Failure to Implement Adequate Design Control Measures. The licensee's corrective actions for this violation are stated in their July 2, 1987, August 4, 1987, and October 30, 1987 responses to NRC for Inspection Report No. 50-400/86-77. The August 4, 1987 response corrected some statements in the July 2, 1987 response. The October 30, 1987 response extended the date for completion of corrective action from November 1987 to March 31, 1988. The cause of the violation was due to errors in design assumptions made for distribution of weld stresses under specific connection types and loading conditions. In addition, a supervisor was involved in the design verification process for the calculation for FCR-AS-10360 in violation of the licensee's design control procedures. The licensee's actions to

correct this violation involved review of the specific calculations, i.e., Detail G, and FCR AS-10360, which contained the errors in design assumptions identified in the violation. This review was accomplished during the inspection and resulted in issuance of NRC Nos. OP-86-0183 and OP-86-0185 to document and disposition these design errors. The specific errors were corrected through issuance of field modifications for Detail G and a plant change request for the spent fuel rack support design (FCR AS-10360). Further corrective action included review of civil-structural calculations for the CB platform steel, the RAB 248 platform steel and steam generator lower lateral supports. This review has been completed within the corporate Nuclear Engineer Department, Civil-Structural Design Section. However, due to other concerns identified by individuals within the CP&L organization regarding these same design calculations, the licensee has decided to conduct an independent review of these calculations and others which involve structural steel design at the Harris Plant. NCR Nos. OP-86-0183 and 0185 are still open pending completion of this independent review. The scope of the independent review is outlined in a CP&L memorandum from R. A. Watson to A. B. Cutter and H. R. Banks dated September 15, 1987, Subject: Independent Review of Structural Steel Design Issues. This memorandum summarized the areas of concern expressed by various individuals within the CP&L organization. The concerns are as follows:

(1) Potential errors in RAB 248 platform steel calculations identified in CP&L Letter No. MS-876288(E), dated June 22, 1987.

(2) Effect of addition of attachments to the steam generator lower lateral supports.

(3) Resolution of previously identified design deficiencies in the CB platform steel calculations. Also, the resolution of questions raised by HPES engineers during review of the platform steel design. The questions regarded the original EBASCO design.

(4) Possible discrepancies in the HPES civil program for cataloging and analyzing the effects of additional attachments to various steel structures.

(5) Possible improper FCR/FM justifications performed by the HPES civil unit.

The inspectors discussed the independent design review program with the Manager of the Nuclear Engineering Department. These discussions disclosed that an outside consultant will be retained by CP&L to perform an overall review of the structural steel design to assess proper methodology in relation to AISC Code compliance. In addition, a detailed review will be conducted by two senior structural

engineers who were not involved in the Harris structural steel design work. These individuals are independent and do not work under the supervision of the principal engineer who was formerly the Manager of the HPES civil unit.

The results of the Independent Review of the Harris Structural Steel design issues will be examined by NRC Region II. Pending completion of this review, and closeout of NCR OP-86-0183 and 0185, Violation Item 400/86-77-01 will remain open.

d. (Open) Violation Item 50-400/86-77-02 Undersized Welds on Cable Tray Supports. The licensee's corrective actions for this violation are stated in their July 1, 1987 response to NRC for Inspection Report 50-400/86-77. The cause of the violation was due to failure of the welders to provide acceptable weld profiles on the identified Detail G welds and failure of the QC inspectors to identify the undersized welds. The licensee issued Nonconformance Report numbers OP-86-0149 and OP-86-163 to document and disposition the undersized Detail G welds. During construction of the Harris Plant, the licensee's QA staff issued several NCRs regarding undersized welds. These NCR were examined by a Region II QA specialist during the September 4-11, 1986 inspection. (Note: For a listing of these NCRs see Inspection Report Number 50-400/86-69.) An indepth review of the licensee corrective actions pertaining to identification and correction of undersized structural steel welds will be conducted by a Region II welding specialist (i.e. Metallurgical Engineer or an NDE specialist). Pending completion of this review, Violation Item 86-77-02 will remain open.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Case RII-87-A-0086

a. Background

An individual, hereinafter referred to as the alleger, contacted NRC Region II and expressed several concerns relating to undersized welds on structural steel, structural steel design methodology and design of welded pipe attachments. Followup on these allegations is discussed in paragraphs 5.b through 5.d below.

b. Deficiencies in Structural Steel Design

(1) Concern

Design deficiencies were previously identified by NRC on major structural calculations. Other structural steel design deficiencies were also identified by licensee engineers in an

internal CP&L memorandum dated June 22, 1987. In addition, there are other questions regarding design of the containment building structural steel. These concerns are listed in the table below.

TABLE

CB Platform Steel Design Concerns/Contentions

Concern Number	Contention
1	Connections were modeled in the computer analysis as pinned connections when in fact some were as built as rigid (fixed) connections.
2	Stress calculations were not performed for angles and plates in some connections.
3	Load reversals not summed correctly.
4	Eccentricity not considered in connection or weld design.
5	Wrong load case chosen as most critical.
6	Torsional loads not considered.
7	Whip restraint loads not considered.
8	Expansion joints used to limit thermal stresses are not functional.
9	The value for accident thermal temperature, i.e. $\Delta T = 148°F$, was not justified.
10	Use of 1.1 interaction factors were not justified.
11	Reduced horizontal "g" values were applied to vertical loads in seismic analysis.
12	Supplemental steel was added to CB platform to stiffen member and reduce weak areas stresses. However, the loads from the supplemental steel were not included in analysis.
13	Hanger foot print loads were not resolved to beam centroids.

14	Effect of DBE loads from pipe supports on platform steel not checked.
15	Use of rigid plate theory used to check embeds is not justified.

(2) Discussion

The design deficiencies discussed by the alleger which were identified by NRC are those associated with Violation Item 400/86-77-01. These deficiencies involved the Detail G connection on Drawing number CAR 2168-G-251-501 (containment building cable tray frames) and incorrect application of the AISC Ultimate Strength Method for welds in design of connections for the new fuel pool racks covered by FCR AS-10360. As stated in paragraph 3.c above, the licensee issued NCR numbers OP-86-0183 and OP-86-0185 to document and disposition these errors.

The RAB 248 platform steel concerns are documented in CP&L Letter No. MS-876288(E), dated June 22, 1987. This letter is discussed in Paragraph 3.c above.

The 15 specific concern/contentions regarding the CB platform steel were identified by CP&L structural engineers employed in the Harris Project Engineering Staff Civil Structural Unit in Fall 1986, during the final design verification of the CB platform steel design. These specific questions were addressed to the Harris project Architect-Engineer, EBASCO in a hand written telecopy transmitted to EBASCO at 4:03 p.m. on October 20, 1986. The inspectors reviewed a copy of this telecopy and noted that the alleger's concerns are verbatim as those listed in the telecopy. The EBASCO response to the CP&L questions is documented in an EBASCO memo to Lee Williams/Ron Knott (CP&L HPES) from T. McCarthy (EBASCO) dated January 6, 1987, Subject: Harris RCB Platform Supplementary Analysis. Additional responses to address these concerns were prepared by the CP&L Design Engineers to cross reference responses to the concern/contentions to specific EBASCO or CP&L design calculations. This cross reference is outlined as an undated, unlabeled listing of 16 contentions and specific response (including referenced calculation) which addressed the contentions. The cross reference is included in the CB structural steel calculation books.

In response to this concern, a review of the calculations for the containment platforms was conducted by a Region II structural engineer to make a more detailed evaluation concerning the adequacy of the CB platform steel design. Specifically, several highly-stressed beams (as determined by

CP&L and EBASCO to have Interaction Equation values exceeding 0.85) were checked to verify design procedures, including assumptions, load combinations and factors, allowable stresses, and structural configuration. Also, the calculations for the connections, including welds, clip angles, and embedded plates, of the respective beams were examined. The beams checked included 514, 515, 517 of the Load Verification Calculation, LV54, Volume 4. In addition, calculation number CAS15, "Containment Building Platform Static Analysis - Revised Stresses," which is a compilation of beams which EBASCO was unable to qualify when using the original design load assumptions, was reviewed.

The preliminary conclusion reached upon completion of this evaluation is that the calculations have been performed in a manner consistent with professional standards of practice and satisfy design and FSAR requirements. However, additional review of the structural steel design concerns will be conducted by NRC Region II in closeout of Violation Item 400/86-77-01, following completion of the licensee's independent review of structural steel design concerns discussed in paragraph 3.b.

(3) Findings

The concerns expressed by the alleger had been previously documented by the licensee in various internal CP&L documents, or had been previously identified by NRC. These concerns will be further evaluated by NRC following completion of the independent review being conducted by the licensee.

During review of calculation LV-54, the inspectors noted that thermal loads were evaluated using methods specified in ANSI N690-1984, Steel Structure - Design. This design is based on steel ductility and local effects of thermal stresses. During review of the FSAR, the inspectors noted that this ANSI standard was not referenced. The inspectors discussed with licensee engineers the need for referencing this standard in the appropriate FSAR Section (Section 3.8.3.2) when the next FSAR amendment is submitted to NRC the Licensee. This was identified to the licensee as Inspector Followup Item 87-41-01, Add ANSI N690-1984 to List of Applicable Codes, Standards and Specification in FSAR Section 3.8.3.2.

c. Undersized Welds on Structural Steel Connections

(1) Concern

Compounding the concerns discussed in Paragraph 5.b., above, there are an unknown number of undersized welds on structural steel connections. As an example, the alleger stated that

undersized welds were found in 126 of 187 connections examined on the CB platform steel. The connections had to be modified because of the undersized welds.

(2) Discussion

The licensee examined 92 of the highest stressed beams for possible effect of undersized welds. This was in response to concerns expressed by NRC Region II during the September 4-11, 1986 inspection which are documented in Inspection Report Number 50-400/86-69. In addition to these 184 connections (92 beams x 2 connections per beam = 184), the licensee evaluated three other individual connections which were considered to be critical and highly stressed (Note: Highly Stressed is defined as a beam loaded to greater than 0.85 fy). In order to perform the evaluation, a detailed as-built sketch was prepared of each weld on each connections. These weld sketches were attached to NCR number 86-0542. Review of weld sketches showed that at least one weld was undersized on 126 of the connections. However, in all but a few cases, the effect of the undersized welds was offset by oversized welds in other parts of the connection.

The inspectors examined the design document (design change notices, field change requests, and modification packages) listed below which were issued to modify the CB platform steel connections. The inspectors also examined the calculations associated with the modification packages. A summary of the design documents reviewed by the inspectors is listed below.

- DCN 650-960, issued on March 19, 1986, by EBASCO to modify connections on CB steel platforms due to increases in loads from attachments to platforms. This DCN involved the more highly stressed beams which had been identified by EBASCO prior to the time the structural steel design was transferred from EBASCO to the CP&L onsite HPES unit.

- FCR AS-10481 issued on May 28, 1986 to revise details on DCN 650-960 for modifications to connections on elevation 261 CB platform at azimuths 5°-9', 266°, 30', and 345°. The FCR was required due to field conditions and interferences.

- FCR-AS-10546, issued on June 12, 1986, to revise details on DCN 650-960 for modification to connections on elevation 286 at azimuths 330° and 345°. The FCR was required due to field conditions and interferences.

- Field Modification (FM) C 6525, issued October 5, 1986, to modify four connections to increase capacity of connections to carry additional vertical loads.

- FM C 6526, issued October 4, 1986, to modify five connections due to torsional loads.

- FM C 6527, issued October 5, 1986, to modify three connections due to loads from the main steam line whip restraints. One of these connections number 236-C3822N-188 had been undersized. The specified weld size (per the original design drawing) was 0.3125 inches. The "as-built" weld size was 0.281 inches. The required weld size, due to the increased loads was 0.345 inches.

- FM C 6529, issued October 4, 1986, to modify one connection by installation of two 7/8 inch diameter high strength bolts. Modification was due to increase in loads acting on CB platform due to attachments.

- FM C 6531, issued October 7, 1986 to modify five connections due to increased loads from attachments to the CB structural steel platforms. The modifications involved increasing weld sizes (beyond those originally specified), or addition of high strength bolts to the connections

- FM C 6535, issued October 8, 1986, to modify two connections due to torsional loads and to add seat plate to one connection due to increased vertical loads due to attachments. The original designed weld on this connection was overstressed due to new attachment loads.

- FM C 6539, issued October 9, 1986, to modify two connections due to torsional loads.

- FM C 6542, issued October 30, 1986, to modify two connections due to increased loads, and to remove the modifications installed under DCN 650-960 on one connection. The removal consisted of cutting the plates attached to the flanges on Connection Number 261-C3822N-575 so the connection would behave as pinned connection rather than a fixed connection after the plates were installed. The fixed connection was undesirable since high stresses were transferred from the beam into the embed plate at this connection.

- FM C 6543 modified once connection due to construction problem (nuts were not tightened on bolts and were inaccessible due to interferences added since the bolt were installed).

Review of the calculations and modifications listed above showed that the welds were modified (enlarged) due to changes in loads during final platform steel design verification program, not because of undersized welds.

(3) Findings

The concern was not substantiated. Although undersized welds existed on some connections, modifications to the welds/connections were not required because of the undersized welds, but as a result of additional loads from new attachments to the structural steel platforms. The analysis of the additional attachment loads were part of a planned systematic design program.

d. Design of Welded Pipe Attachments

(1) Concern

Problems were identified with design of welded pipe attachment. The criteria used to design the attachments (e.g., lugs, trunnions, etc) was inconsistent and was not properly documented.

(2) Discussion

This is similar to the concern expressed in Quality Check 9950. An evaluation of welded pipe attachment (WPA) calculations disclosed inconsistencies in use of various coefficients (e.g. Beta factors and load coefficients). To respond to the Quality Check, licensee engineers conducted a review of 125 pipe hanger designs which included a welded pipe attachment. This review disclosed that there were inconsistencies in selection of load coefficients and Beta factors. However, all calculations reviewed were found to have used acceptable load coefficients, although in some cases they result in higher calculated stresses. The review disclosed that in some calculations very conservative coefficients were used in calculating load stresses in the pipe wall at the WPA, while in other cases, the correct value listed in various Beta and/or load coefficient tables was used in the calculation. Use of the conservative load coefficient values resulted in calculation of higher local stresses in the pipe welds. For example, since pipe stress is based on the load coefficient or Beta factor multiplied by the applied loads acting on the pipe at the WPA, use of a Beta factor or load coefficient of 0.4 when the correct value was 0.3 would result in a 33 percent increase is calculated stresses in the pipe (0.4/0.3 = 1.33). Therefore even though an incorrect factor was used in the analysis, its use had no safety significant since the errors were always in the conservative direction.

During review of the 125 calculation packages, minor errors were found in eight supports calculations. These errors were not associated with the Beta factors or load coefficients. The

licensee issued NCR 87-096 to document and disposition these minor errors. Correction of the problems resulted in minor revisions to the calculation packages. No work was required to modify the hangers affected by the minor calculation errors.

In order to assure that consistent values for load coefficients would be used in future design of welded pipe attachments, the licensee issued NED Design Guide No. DG-II.12, Design and Analysis of Welded Pipe Attachments for the Harris Project. The inspector reviewed this design guide and found that it clearly establishes requirements to be used in calculation of local stresses in pipe walls at the WPA.

(3) Finding

The concern was substantiated in that criteria used in design of welded pipe attachments was inconsistent. However this problem had been previously identified and resolved under the licensee's Quality Check Program. Although inconsistencies were identified in some calculations, these inconsistencies did not result in nonconservative piping designs. The inconsistent use of load coefficients resulted in more conservative calculated stresses and thus did not affect any hardware. Thus, this substantiated concern has no safety significance.

6. Previously Identified Inspector Follow-up Items

a. (Closed) IFI 400/86-69-02, Reinspection of Shared Electrical Supports for Deleted Conduit

During the September 4-11, 1986 NRC inspection, a Region II inspector examined the shared conduit support problems identified on NCR 86-0444. The NCR documented discrepancies identified by the licensee during walkdown inspections of non-nuclear safety-related (NNS) conduits which disclosed that installation of NNS conduit supports that share supports with safety-related raceway had not been inspected for seismic attributes. The cause of this problem was due to insufficient instructions in CP&L Procedure TP-51, Inspection of Non-Safety Related Electrical Raceways, to assure that installation of NNS conduits that shared supports with safety-related (SR) conduit would be inspected to verify that seismic installation requirements were completed as noted on Sheet 14 of Drawing number CAR 2166-B-060. In order to correct this problem, the licensee amended Revision 7 of procedure TP-51 with a procedure change notice (PCN) to clearly specify that NNS conduit that shares supports with SR conduit are to be inspected to assure installation conforms to seismic requirements. The NNS conduits listed in NCR 86-0444 were reinspected. Conduit found with unacceptable seismic installations were reworked and reinspected. Some of the rework involved removal of temporary conduits that had been installed for construction, and other conduit

no longer required due to design changes and thus were scheduled to be removed. These conduits are referred to as "deleted conduit". During the September 4-11, 1986 inspection, an additional conduit number 17129K, on a shared support was identified which lacked inspection records to verify compliance with seismic attributes. This problem was documented on NCR 86-0555. To correct this problem, conduit 17129K was inspected for seismic installation attributes. The installation was found to be acceptable. The licensee's corrective actions to resolve the shared support problems included a detailed walkdown program to identify all NNS and "deleted" conduits which share supports with SR conduit, review of inspection records to verify NNS conduit on shared supports had been inspected, necessary rework to correct shared support problem identified during inspection, removal of "deleted" conduit where required, and documentation of inspection of NNS conduit on shared supports when records were found to be incomplete. In addition, the Regulatory Guide 1.29 walkdown program, discussed in paragraph 6.d below, was another program conducted independent of the resolution of NCR 86-0444 and 86-0555 to identify potentially inadequately supported non-safety related equipment which could affect SR equipment. The inspectors examined documentation supporting closeout of NCR 86-0444 and -0555 and the following memorandums which summarize corrective actions to resolve shared support problems:

- Memorandum to A. H. Rager from V. Cox/J. Martin dated September 22, 1986, Subject: Deleted Conduits

- Memorandum to M. Holveck from V. Cox, dated October 3, 1986, Subject: Response to Concern Related to Deleted Conduits Installed on Shared Supports

- Memorandum to V. Cox from J. Frantz, dated October 28, 1986, Subject: Shared Support Walkdown

- Memorandum to File HXDE-2XX-XXX-516 from J. Martin / M. Bodnar, dated November 24, 1987, Subject: Deleted Conduit

Subsequent to closeout of NCR 86-0444 and -0555, NCR OP-86-0121 was issued on October 24, 1986, regarding identification of two additional conduits, numbers 19567K and 16231U, which had not been inspected for seismic attributes during original installation. The conduits were inspected for seismic attributes to resolve this problem. Conduit 16231U was acceptable while conduit 19567K was found to be unacceptable and required further evaluation. After evaluation by HPES, this conduit was found to be acceptable. Further review by the licensee resulted in cancellation of NCR OP-86-0121 since the walkdown inspection program being conducted to identify these problems was in progress. The inspector concurs with cancellation of this NCR. Based on review of the licensee's program for reinspection of shared supports, IFI 400/86-69-02 is closed.

b. (Closed) IFI 86-77-04, Resolution of Cable Tray Riser Design Concerns

An engineer, designated Individual E in Inspection Report Number 50-400/86-77, involved with the analysis of the containment building cable tray riser structures resigned his position at the Harris site to accept other employment. Since he had only partially completed the cable tray riser analysis, Individual E prepared a list of his pending concerns regarding the analysis and submitted this list to his supervisor. The licensee addressed these concerns in calculation number LV-66, Cable Tray Riser Frame (Final Verification). The inspectors reviewed calculation LV-66 to determine if the design engineer's concerns were resolved. During review of this calculation, the inspectors noted that the change in temperature (ΔT) used in calculation of thermal stress in the structure was assumed to be 60°F not the value of 148°F used in the CB structural steel platform analysis. This error appeared on page 67 of the calculation and was used in recalculation of thermal stresses after use of the correct ΔT value of 148°F resulted in an interaction equation of 2.47, which considerably exceeds the acceptable value of 1.0.

The change in the ΔT value from 148° to 60°F constitutes a change in a specified design input, specifically, the accident temperature used to calculate thermal stresses. This change was not properly identified, approved, documented or controlled in accordance with the licensee's approved QA program or 10 CFR 50, Appendix B, Criterion III. This was identified to the licensee as Violation Item 400/87-41-02, Uncontrolled Change to Design Input.

After the inspector identified the problem regarding the change in design input for ΔT, licensee engineers conducted a review of calculation LV-66. This review disclosed that incorrect allowable stresses had been used in some portions of the calculations to qualify some members for the operating base earthquake design loads. These problems, the incorrect ΔT and the incorrect allowable stresses, were documented on NCR 87-51. During resolution of the NCR, which involved an extensive revision to calculate LV-66, licensee engineers discovered some errors in geometry in the computer model of the riser structure analyzed in the original calculation LV-66. As a result, licensee engineers conducted an extensive design re-evaluation of the riser structure. This re-evaluation was still ongoing at the conclusion of the inspection.

Based on review of calculation LV-66, the inspectors determined that Individual E's concerns regarding the cable tray riser frame analysis were addressed in calculation LV-66, although the errors discussed above will result in revision of the calculation. A summary of Individual E's concerns and their resolutions follows below:

(1) Concern

No frequency calculations for attaching conduit and instrument supports were performed.

Resolution

A frequency analysis was performed and considered in analysis of attachment of conduit and instrument supports to the riser frame. The attachments were checked for a reduced frequency determined from an updated static analysis, based on actual column properties.

(2) Concern

The column properties input in the EBASCO analysis were incorrect. The moment of inertia (I) input in the EBASCO analysis assumed the 2 C9 x 15 members were adjacent to each other and behaved as a beam without consideration of actual riser configuration (Channels spaced approximately 15 inches center-center with cross-braces consisting of horizontal tie plates connecting the channels). The same condition also applies to the internal bracing (built-up 4 x 4 x 3/8 angles).

Resolution

The actual "as-built" riser configuration of the riser structure (channels connected by the plates) was modeled in the analysis and an equivalent moment of inertia was computed for these members. The equivalent moment of inertia was used in the analysis. The internal bracing was also analyzed using the correct moment of inertia.

(3) Concern

There were no calculations performed to determine stresses in welds on internal connections.

Resolution

Calculation were performed to check weld stresses in connections on tie plates, double angle cross braces, and the W8 braces.

(4) Concern

There were no stress calculations performed for the tie pipes. The main concern here was that the material grade used for the pipe had a yield stress of 25 Ksi (ASTM A53, Type F, Grade B) which was substituted for the original pipe material which specified a yield stress of 36 ksi.

Resolution

The pipe material originally specified was not available in either ASTM 501 or ASTM A53, Type E or S, Grade B (yield stress of 36 ksi). Therefore, the material with the 25 ksi yield stress was substituted. Stress calculations were performed for 1½-inch diameter pipes with a 25 ksi yield stress. The analysis showed that the materials used were acceptable.

(5) Concern

Additional attachments, e.g. nonscheduled conduit, to the riser structure were not considered in the analysis.

Resolution

Loads from additional attachments were considered in the reanalysis (calculation LV-66).

(6) Concern

The model used in the original analysis had bracing in rear of structure at azimuth 217°-30' when actual structure has "x" type bracing in front. The incorrect modeling may reduce calculated loads to the structure, and may also affect the platform steel.

Resolution

The structure was remodeled using the correct configuration for the bracing. The effect of the loads from the riser structure on the CB platform steel was included in the analysis for the CB platform (See paragraph 5.b).

The inspectors will re-examine calculation LV-66 as part of follow-up on corrective action for Violation Item 400/87-41-02. Since Individuals E's concerns were resolved, IFI 400/86-77-04 is considered closed.

c. (Open) IFI 400/86-77-05, Painting of Restricted Embeds

During construction of the Harris plant, the licensee used adhesive tags to identify restricted embed plates. The restricted plates were those to which no new attachments could be made to the embed without the explicit approval of design engineering. Restricted embeds were identified on field change requests (FCRs). At the end of construction, licensee engineering personnel conducted a review of the restricted embed program and concluded that the use of the adhesive tags to identify restricted plates was ineffective. The licensee conducted an indepth review, which included a field walkdown and design evaluation of restricted embeds, to ascertain that the

restricted embeds were not loaded in excess of their design capacity. The licensee also decided to paint the restricted embeds with red paint to permanently identify them. During the inspection documented in Inspection Report Number 50-400/86-77, the inspector noted during a field inspection of restricted embeds that there was apparently some confusion regarding which embeds required painting to identify them as restricted embeds. Licensee engineers indicated to the inspector that they were coordinating the restricted embed painting program with craft personnel to assure that all the restricted embeds would be painted. During the current inspection, licensee engineers indicated that all restricted embeds had been painted. The inspector selected the following FCRs which restrict embeds: FCR AS-6340 (R1), 7066 (R1), 7067, 7068, 7392, 8665, 8666, 9345 (R1). Examination of the embeds restricted by these FCRs disclosed the following problems.

(1) The description of the location of the embed restricted by FCR AS-7067 was incorrect on page 1 of 4. The description stated that support/embed in question was located 7'-0" east of column line Fv, when in fact, correct location is 7'0" west of column line Fv.

(2) The as-built sketch on Sheet 7 of 7 of FCR AS 9345 (R1) shows a conduit support attached to embed shown on top of page, when in fact a cable tray support is attached in this location.

(3) The incorrect area was painted for embed shown on FCR AS-7392. The as-built sketch of the embed was also incorrect.

Following completion of the inspector's walkdown, a licensee engineer examined embeds restricted on 12 additional FCRs. During the engineer's walkdown, he identified an embed which had an attachment not shown on the as-built sketch attached to the FCR. Based on these reviews, the licensee concluded that an additional indepth examination of the restricted embeds was necessary. Pending the outcome of the findings of this reinspection, and determination of the safety significance of any discrepancies identified, IFI 400/86-77-05 will remain open.

d. (Closed) IFI 400/86-77-06, Review of Discrepancies Identified in R.G. 1.29 Walkdown Program

The inspector examined the discrepancies in the licensee's R.G. 1.29 walkdown program identified by an NRC consultant during the inspection documented in Inspection Report Number 50-400/86-77. These discrepancies and their corrective actions are summarized below:

population size of the affected anchor using CP&L procedure CQA-7, Evaluation of Program Effectiveness. The inspector reviewed the results of the licensee evaluation of these anchors which are documented in CP&L Letter Number MS-876316(E) dated July 17, 1987, Subject: CQA-7 Evaluation of Silver Anchors Used in Engineering Evaluations of Reg. Guide 1.29 Interactions. The results of the CQA-7 program showed that the non-Q anchor program had proficiency greater than 97%, identifying only 26 unacceptable anchors in the 945 anchors inspected. The licensee concluded that their non-Q anchor program was acceptable. The inspector concurs with licensee's conclusions.

(3) Discrepancy

Oversized 2-bolt clamps were used on conduit for Item 69/70 in Area A-1-190-1. In addition, some clamps were installed with only one bolt due to interference with structural steel supports for stairway platforms.

Corrective Action

Normal practice in installing conduit supports (clamps) is that the ID of the clamp is equal in size to the OD of the conduit. This attaches the conduit rigidly to the structure. However, review of the package for this item showed that oversized conduit clamps were installed intentionally to prevent the conduit from interacting with (falling on) safety related equipment in case of a seismic event. Review of the sketches detailing installation of the clamps showed that some of the clamps were intended to have only one anchor. The anchor placement reports (APR) attached to the package reviewed by the NRC Consultant were incomplete. Review of the completed APRs for this item showed that the number of anchors installed agreed with the as-built conditions in the field. No rework was required.

(4) Discrepancy

Incorrect span lengths were used in analysis of supports for Item 38 in Area Package F-2-236-1. Also, additional conduits were not considered in analysis.

Corrective Action

The supports were re-evaluated by licensee engineers using the correct span lengths and considering all attachments. The inspector examined the calculations and verified that they had been corrected. No rework was necessary.

(5) Summary

The licensee corrected the specific problems identified by the NRC consultant. No rework was necessary to correct these problems. The licensee also evaluated the generic aspects of these specific problems and conducted further inspections and evaluations to determine if problems existed with the R.G. 1:29 walkdown program which had been completed to date and revised procedures to prevent these same problems from occurring again. Some discrepancies were identified in other R.G. 1.29 packages which required rework to correct. The inspector examined the procedures which controlled the R.G. 1.29 program and conducted field walkdown inspections to examine selected R.G. 1.29 packages. The results of these inspections are discussed below.

(a) Review of R.G. 1.29 Procedures

The inspector examined procedures which controlled the R.G. 1.29 evaluation program. These procedures were:

- HPES Manual of Instructions (MOI) 7.1.F, Guidelines for Evaluation of Reg. Guide 1.29 Problem Identification Reports

  HPES MOI 7.1.G, Guidelines for Evaluation of Interdisciplinary Clearance Problem Identified Reports

- HPES MOI 7.6.B, Reg. Guide 1.29

- HPES MOI 7.1.A, General Design Guidelines for Civil/Structural Engineering Unit

Review of the above procedures disclosed that procedure 7.1.F was revised as a result of the NRC consultant's finding to emphasize the need to accurately determine span lengths to be used in calculations, and to assure that required hardware was installed. Procedure 7.6B was revised to require evaluation of field conditions by HPES engineering personnel.

(b) Review of R.G. 1.29 Generic Calculations

The inspector examined calculation number MOI 7.1.G, Interdisciplinary Clearance Guidelines, and Calculation number MOI 7.1.F, Reg. Guide 1.29 Evaluation Guidelines. These calculation included the following.

- Basis for acceptance of electrical boxes, wall mounted transformers, power panels, and communication boxes and speakers

- Allowable conduit loads on B-Line supports
- Fire extinguisher bracket design
- Notes on Non -Q expansion anchors
- Emergency light box support calculations
- Basis for acceptance of one-inch diameter air line supports
- Basis and review of trapeze supports

(c) Field Walkdown Inspection of R.G. 1.29 Packages

The inspector performed a walkdown inspection and examined selected R.G. 1.29 area packages/case numbers. The packages examined were those that the licensee had reverified due to findings of the NRC consultant. The inspectors also reviewed the verification calculations which formed the basis for acceptance of the R.G. 1.29 interactions. Packages/Items examined are listed in the Table below.

TABLE

R.G. 1.29 Verification Walkdown
Problems Examined

Area Package Numbers	Case Numbers
A-1-236-1	3 and 4
A-1-236-1	25
A-2-236-2	51
A-1-261-1	8
A-2-261-1	5 and 6
A-6-261-1	25
A-1-286-1	71
A-1-286-1	89
A-1-286-1	133
A-2-286-1	17
A-2-305-1	26
A-2-305-1	96
A-2-305-1	97
A-2-305-2	6

Examination of the above items disclosed some minor discrepancies in three of the packages. Two of the errors involved use of incorrect span lengths in calculation of the

loads on the supports and the other involved an error in the sketch of the support configuration in the walkdown sketches which resulted in an error in calculation of the loads on one support. The inspector determined that the errors were minor and did not affect the results of the walkdown verification. The licensee revised the calculation to correct the minor errors. These errors had no safety significance.

(6) Conclusions

The licensee's R.G. 1.29 walkdown verification met NRC requirements. The licensee's R. G. 1.29 program was sufficient to identify interactions between safety and non-safety related equipment, and assure that the non-safety related equipment would not collapse on or interfere with safety-related equipment during a seismic event. Although some minor discrepancies were identified by the inspector during this inspection, these discrepancies had no safety significance. The inspectors concluded that the licensee's R.G. 1.29 walkdown verification was thorough and conservative. IFI 400/86-77-06 is closed.

e. (Closed) IFI 400/86-77-07, Follow-up on Justification for FCR AS-2381

Review of the justification for FCR AS-2381 showed that the calculations for analyzing the baseplates included shear and axial loads only, no moments. The interaction equation for these loads (shear and tension) equaled 0.99. The concern was that since attachment of the structural members to the baseplate was with clip angles welded on three sides, the actual loads on the baseplate anchors would be in excess of the allowable due to moments carried through the welded connections (i.e. the interaction equation would exceed 1.0).

The inspector examined the calculation titled "Addendum to RAB 315.5 Calculation Book." Review of this calculation showed that the licensee reanalyzed the platform using updated seismic coefficients for 4 percent and 7 percent damping, and the as-built loads acting on the platform. The analysis was performed by assuming the end condition for the connections to the baseplates were both rigid (fully fixed) and semi-rigid (one-third fixity). These assumptions resulted in moment transfer into the baseplate; however, the frequency of the platform was also increased due to the fixity which permitted transfer of moments. The overall result was a reduction in the axial and shear stresses acting on the baseplate. The calculation showed that the maximum bolt interaction was 0.314 and the maximum principal stresses in the plate were 1085 psi versus allowable of 27,000 psi. The revised calculation showed that the original justification for FCR AS-2381 was based on conservative assumptions and that the baseplate design was acceptable. IFI 400/86-77-07 is closed.